NEI Public Meeting w/NRC
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Construction Inspection Program and ITAAC Verification Topics

The ITAAC for RVH2 – AP1000 ITAAC 2.1.3.3			
Requirement	Inspection, Test or Analysis	Acceptance Criteria	ITAAC Determination Basis
3. The components identified in Table 2.1.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.1.3-1 as ASME Code Section III.	ASME III Code Data Package for the as-built components identified in Table 2.1.3-1.

2.1.3 Reactor System

Design Description

The reactor system (RXS) generates heat by a controlled nuclear reaction and transfers the heat generated to the reactor coolant, provides a barrier that prevents the release of fission products to the atmosphere and a means to insert negative reactivity into the reactor core and to shutdown the reactor core.

The reactor core contains a matrix of fuel rods assembled into fuel assemblies using structural elements. Rod cluster control assemblies (RCCAs) are positioned and held within the fuel assemblies by control rod drive mechanisms (CRDMs). The CRDMs unlatch upon termination of electrical power to the CRDM thereby releasing the RCCAs. The fuel assemblies and RCCAs are designed in accordance with the principal design requirements.

The RXS is operated during normal modes of plant operation, including startup, power operation, cooldown, shutdown and refueling.

The component locations of the RXS are as shown in Table 2.1.3-3.

1. The functional arrangement of the RXS is as described in the Design Description of this Section 2.1.3.
2. a) The reactor upper internals rod guide arrangement is as shown in Figure 2.1.3-1.

   b) The rod cluster control and drive rod arrangement is as shown in Figure 2.1.3-2.

   c) The reactor vessel arrangement is as shown in Figure 2.1.3-3.
3. The components identified in Table 2.1.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.
4. Pressure boundary welds in components identified in Table 2.1.3-1 as ASME Code Section III meet ASME Code Section III requirements.
5. The pressure boundary components (reactor vessel [RV], control rod drive mechanisms [CRDMs], incore instrument guide tubes) identified in Table 2.1.3-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.
6. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.
7. The reactor internals will withstand the effects of flow induced vibration.
8. The reactor vessel direct injection nozzle limits the blowdown of the reactor coolant system (RCS) following the break of a direct vessel injection line.
9. a) The Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

b) The Class 1E components identified in Table 2.1.3-1 are powered from their respective Class 1E division.

c) Separation is provided between RXS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

10. The reactor lower internals assembly is equipped with holders for at least eight capsules for storing material surveillance specimens.

11. The reactor pressure vessel (RPV) beltline material has a Charpy upper-shelf energy of no less than 75 ft-lb.

12. Safety-related displays of the parameters identified in Table 2.1.3-1 can be retrieved in the main control room (MCR).

13. The fuel assemblies and rod control cluster assemblies intended for initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements.

14. A top-of-the-head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle, can be performed.

Inspections, Tests, Analysis, and Acceptance Criteria

Table 2.1.3-2 specifies the inspections, tests, analysis, and associated acceptance criteria for the RXS.

Table 2.1.3-1					
Equipment Name	Tag No.	ASME Code Section III Classification	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
RV	RXS-MV-01	Yes	Yes	-	-
Reactor Upper Internals Assembly	RX-MI-01	Yes	Yes	-	-
Reactor Lower Internals Assembly	RXS-MI-02	Yes	Yes	-	-
Fuel Assemblies (157 locations)	RXS-FA-A07/A08/A09/B05/B06/B07/B08/ B09/B10/B11/C04/C05/C06/C07/C08/C09/C10/ C11/C12/D03/D04/D05/D06/D07/D08/D09/ D10/D11/D12/D13/E02/E03/E04/E05/E06/E07/ E08/E09/E10/E11/E12/E13/E14/F02/F03/F04/ F05/F06/F07/F08/F09/F10/F11/F12/F13/F14/ G01/G02/G03/G04/G05/G06/G07/G08/G09/ G10/G11/G12/G13/G14/G15/H01/H02/H03/ H04/H05/H06/H07/H08/H09/H10/H11/H12/ H13/H14/H15/J01/J02/J03/J04/J05/J06/J07/J08/ J09/J10/J11/J12/J13/J14/J15/K02/K03/K04/ K05/K06/K07/K08/K09/K10/K11/K12/K13/ K14/L02/L03/L04/L05/L06/L07/L08/L09/L10/ L11/L12/L13/L14/M03/M04/M05/M06/M07/ M08/M09/M10/M11/M12/M13/N04/N05/N06/ N07/N08/N09/N10/N11/N12/P05/P06/P07/P08/ P09/P10/P11/ R07/R08/R09	No[1]	Yes	-	-

Note: Dash (-) indicates not applicable.

1. Manufacture standard, but uses ASME Section III guidelines

Table 2.1.3-1 (cont.)					
Equipment Name	Tag No.	ASME Code Section III Classification	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
Rod Cluster Control Assemblies (RCCAs) (minimum 53 locations)	RXS-FR-B06/B10/C05/C07/C09/C11/D06/ D08/D10/E03/E05/E07/E09/E11/E13/F02/F04/ F12/F14/G03/G05/G07/G09/G11/G13/H04/ H08/H12/J03/J05/J07/J09/J11/J13/K02/K04/ K12/K14/L03/L05/L07/L09/L11/L13/M06/ M08/M10/N05/N07/N09/N11/P06/P10	No(1)	Yes	-	-
Grey Rod Control Assemblies (GRCAs) (16 locations)	RXS-FG-A07/C03/C11/E05/E07/E09/G01/G05/ G09/G13/ J05/J07/J09/L03/L11/N07	No(1)	Yes	-	-
Control Rod Drive Mechanisms (CRDMs) (69 Locations)	RXS-MV-11B06/11B08/11B10/11C05/11C07/ 11C09/11C11/11D04/11D06/11D08/11D10/ 11D12/11E03/11E05/11E07/11E09/11E11/ 11E13/11F02/11F04/11F06/11F08/11F10/ 11F12/11F14/11G03/11G05/11G07/11G09/ 11G11/11G13/11H02/11H04/11H06/11H08/ 11H10/11H12/11H14/11J03/11J05/11J07/ 11J09/11J11/11J13/11K02/11K04/11K06/ 11K08/11K10/11K12/11K14/11L03/11L05/ 11L07/11L09/11L11/11L13/11M04/11M06/ 11M08/11M10/11M12/11N05/11N07/11N09/ 11N11/11P06/11P08/11P10	Yes	Yes	No/No	No
Incore Instrument Guide Tubes (42 Core Locations)	IIS-JT-G01 through G42	Yes	-	-	-

Note: Dash (-) indicates not applicable.

1. Manufacture standard, but uses ASME Section III guidelines

Table 2.1.3-1 (cont.)					
Equipment Name	Tag No.	ASME Code Section III Classification	Seismic Cat. I	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display
Source Range Detectors (4)	RXS-JE-NE001A/NE001B/NE001C/NE001D	-	Yes	Yes/Yes	No
Intermediate Range Detectors (4)	RXS-JE-NE002A/NE002B/NE002C/NE002D	-	Yes	Yes/Yes	Yes
Power Range Detectors – Lower (4)	RXS-JE-NE003A/NE003B/NE003C/NE003D	-	Yes	Yes/Yes	No
Power Range Detectors – Upper (4)	RXS-JE-NE004A/NE004B/NE004C/NE004D	-	Yes	Yes/Yes	No

Note: Dash (-) indicates not applicable.

The ITAAC for RVH3 – ABWR ITAAC 2.1.1.7.			
Requirement	Inspection, Test or Analysis	Acceptance Criteria	ITAAC Determination Basis
7. The RPV internals with stand the effects of FIV.	7. A vibration type test will be conducted on the prototype RPV internals of an ABWR.	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.	Vibration type test report concluding that the prototype RPV internals have no damage or loose parts as a result of the vibration type test.
	A flow test and post-test inspections will be conducted on the as-built RPV internals	The as-built RPV internals have no damage or loose parts.	Inspection report documenting that the as-built RPV internals experienced no damage of loose parts after the flow test.

Table 2.1.3-2 Inspections, Tests, Analysis, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analysis**	**Acceptance Criteria**
1. The functional arrangement of the RXS is as described in the Design Description of this Section 2.1.3.	Inspection of the as-built system will be performed.	The as-built RXS conforms with the functional arrangement as described in the Design Description of this Section 2.1.3.
2.a) The reactor upper internals rod guide arrangement is as shown in Figure 2.3.1-1.	Inspection of the as-built system will be performed.	The as-built RXS will accommodate the fuel assembly and control rod drive mechanism pattern shown in Figure 2.3.1-1.
2.b) The control assemblies (rod cluster and grey rod) and drive rod arrangement is as shown in Figure 2.1.3-2.	Inspection of the as-built system will be performed.	The as-built RXS will accommodate the control assemblies (rod cluster and grey rod) and drive rod arrangement shown in Figure 2.1.3-2.
2.c) The reactor vessel arrangement is as shown in Figure 2.1.3-3.	Inspection of the as-built system will be performed.	The as-built RXS will accommodate the reactor vessel arrangement shown in Figure 2.1.3-3.
3. The components identified in Table 2.1.3-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.1.3-1 as ASME Code Section III.
4. Pressure boundary welds in components identified in Table 2.1.3-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
5. The pressure boundary components (RV, CRDMs, incore instrument guide tubes) retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components of the RXS required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the pressure boundary components (RV, CRDM's, incore instrument guide tubes) conform with the requirements of the ASME Code Section III.

Table 2.1.3-2 (cont.) Inspections, Tests, Analysis, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analysis**	**Acceptance Criteria**
6. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.1.3-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.1.3-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
7. The reactor internals will withstand the effects of flow induced vibration.	i) A vibration type test will be conducted on the reactor internals representative of AP1000.	i) A report exists and concludes that the prototype reactor internals have no observable damage or loose parts as a result of the vibration type test.
	ii) A pre-test inspection, a flow test and a post-test inspection will be conducted on the first as-built reactor internals.	ii) The first as-built reactor internals have no observable damage or loose parts.
8. The reactor vessel direct vessel injection nozzle limits the blowdown of the RCS following the break of a direct vessel injection line.	An inspection will be conducted to verify the flow area of the flow limiting venturi within each direct vessel injection nozzle.	The throat area of the direct vessel injection line nozzle flow limiting venturi is less than or equal to 12.57 in^2.

Table 2.1.3-2 (cont.) Inspections, Tests, Analysis, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analysis**	**Acceptance Criteria**
9.a) The Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analysis, or a combination of type tests and analysis will be performed on Class 1E equipment located in a harsh environment.	A report exists and concludes that the Class 1E equipment identified in Table 2.1.3-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
9.b) The Class 1E components identified in Table 2.1.3-1 are powered from their respective Class 1E division.	Testing will be performed by providing simulated test signals in each Class 1E division.	A simulated test signal exists for Class 1E equipment identified in Table 2.1.3-1 when the assigned Class 1E division is provided the test signal.
9.c) Separation is provided between RXS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See Tier 1 Material, Section 3.3, Nuclear Island Buildings.	See Tier 1 Material, Section 3.3, Nuclear Island Buildings.
10. The reactor lower internals assembly is equipped with holders for at least eight capsules for storing material surveillance specimens.	Inspection of the reactor lower internals assembly for the presence of capsules will be performed.	At least eight capsules are in the reactor lower internals assembly.
11. The RPV beltline material has a Charpy upper-shelf energy of no less than 75 ft-lb.	Testing of the Charpy V-Notch specimen of the RPV beltline material will be performed.	A report exists and concludes that the initial RPV beltline Charpy upper-shelf energy is no less than 75 ft-lb.
12. Safety-related displays of the parameters identified in Table 2.1.3-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.1.3-1 can be retrieved in the MCR.

Table 2.1.3-2 (cont.) Inspections, Tests, Analysis, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analysis**	**Acceptance Criteria**
13. The fuel assemblies and rod control cluster assemblies intended for initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements.	An analysis is performed of the reactor core design.	A report exists and concludes that the fuel assemblies and rod cluster control rod assemblies intended for the initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements.
14. A top-of-the-head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle, can be performed.	A preservice visual examination of the reactor vessel head top surface and penetration nozzles will be performed.	A report exists that documents the results of the top-of-the-head visual inspection, including 360 degrees around each reactor vessel head penetration nozzle.

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 and the RIP motor casings 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 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 flooder (LPFL) spargers for the Residual Heat Removal (RHR) System,

high pressure core flooder (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.

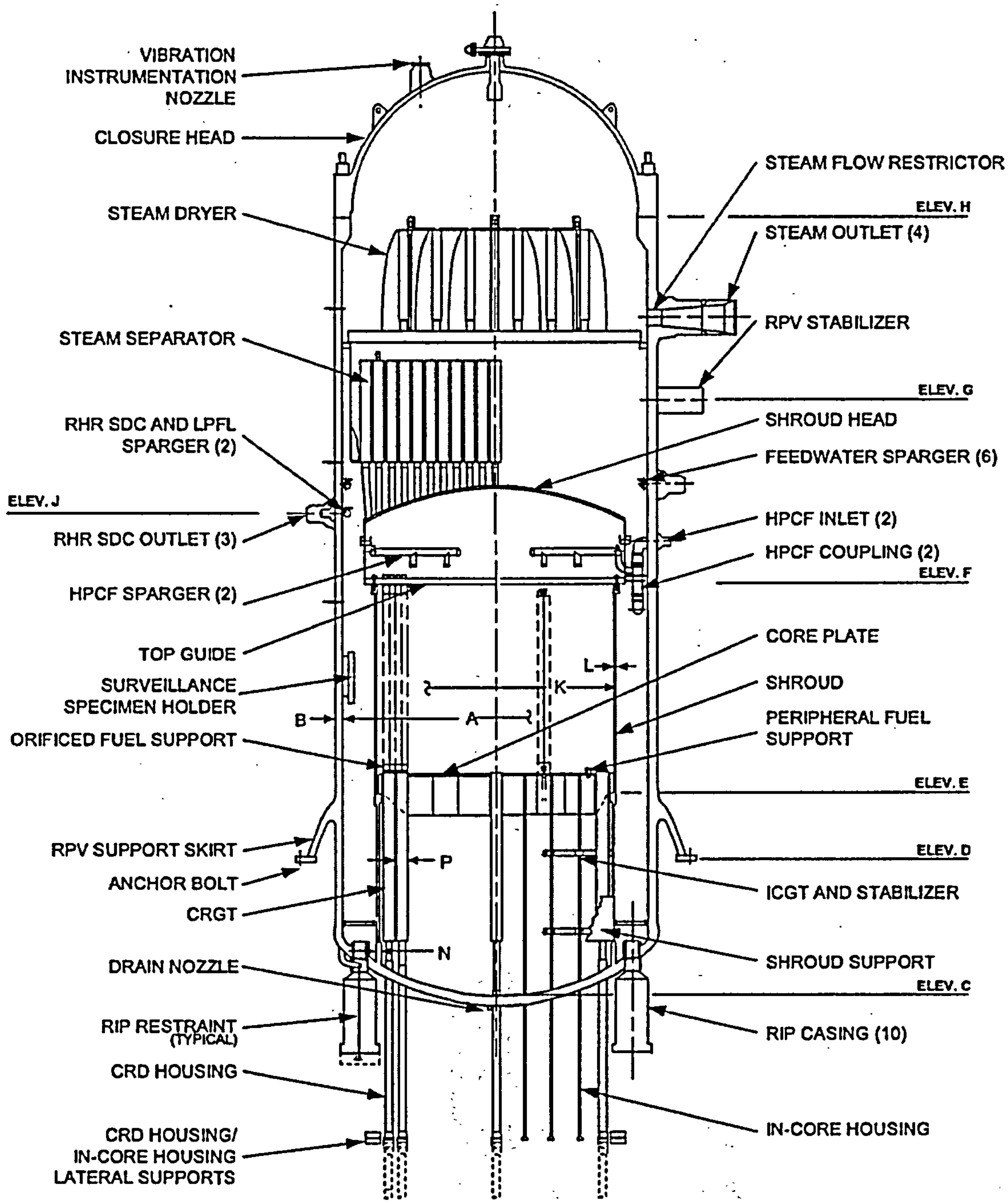


Figure 2.1.1a Reactor Pressure Vessel System Key Features

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.

The ITAAC for CVS5 – AP1000 ITAAC 2.3.2-4.11.a)			
Requirement	Inspection, Test or Analysis	Acceptance Criteria	ITAAC Determination Basis
11.a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed that demonstrate the capability of the valve to operate under its design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.3.2-1 under design conditions.	System startup test results concluding that each motor-operated valve changes position as indicated in Table 2.3.2-1 under design conditions.
	ii) Inspection will be performed for the existence of a report verifying that the as-installed motor-operated valves are bounded by the tested conditions.	ii) A report exists and concludes that the as-installed motor-operated valves are bounded by the tests or type tests.	System design calculations concluding that the as-installed motor-operated valves are bounded by the tests or type tests.
	iii) Tests of the as-installed motor-operated valves will be performed under pre-operational flow, differential pressure, and temperature conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.3.2-1 under pre-operational test conditions.	System startup test results documenting that the motor-operated valves change position as indicated in Table 2.3.2-1 under pre-operational test conditions.
	iv) Exercise testing of the check valves with active safety functions identified in Table 2.3.2-1 will be performed under pre-operational test pressure, temperature and fluid flow conditions.	iv) Each check valve changes position as indicated in Table 2.3.2-1.	System startup test results documenting that the check valves change position as indicated in Table 2.3.2-1.

2.3.2 Chemical and Volume Control System

Design Description

The chemical and volume control system (CVS) provides reactor coolant system (RCS) purification, RCS inventory control and makeup, chemical shim and chemical control, and oxygen control, and provides for auxiliary pressurizer spray. The CVS performs these functions during normal modes of operation including power generation and shutdown.

The CVS is as shown in Figure 2.3.2-1 and the component locations of the CVS are as shown in Table 2.3.2-5.

1. The functional arrangement of the CVS is as described in the Design Description of this Section 2.3.2.

2. a) The components identified in Table 2.3.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.

   b) The piping identified in Table 2.3.2-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.

3. a) Pressure boundary welds in components identified in Table 2.3.2-1 as ASME Code Section III meet ASME Code Section III requirements.

   b) Pressure boundary welds in piping identified in Table 2.3.2-2 as ASME Code Section III meet ASME Code Section III requirements.

4. a) The components identified in Table 2.3.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.

   b) The piping identified in Table 2.3.2-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.

5. The seismic Category I equipment identified in Table 2.3.2-1 can withstand seismic design basis loads without loss of safety function.

6. a) The Class 1E equipment identified in Table 2.3.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.

   b) The Class 1E components identified in Table 2.3.2-1 are powered from their respective Class 1E division.

   c) Separation is provided between CVS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.

7. The CVS provides the following safety-related functions:

   a) The CVS preserves containment integrity by isolation of the CVS lines penetrating the containment.

   b) The CVS provides termination of an inadvertent RCS boron dilution by isolating demineralized water from the RCS.

   c) The CVS provides isolation of makeup to the RCS.

8. The CVS provides the following nonsafety-related functions:

   a) The CVS provides makeup water to the RCS.

   b) The CVS provides the pressurizer auxiliary spray.

9. Safety-related displays in Table 2.3.2-1 can be retrieved in the main control room (MCR).

10. a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.3.2-1 to perform active functions.

    b) The valves identified in Table 2.3.2-1 as having protection and safety monitoring system (PMS) control perform an active safety function after receiving a signal from the PMS.

11. a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.

    b) After a loss of motive power, the remotely operated valves identified in Table 2.3.2-1 assume the indicated loss of motive power position.

12. a) Controls exist in the MCR to cause the pumps identified in Table 2.3.2-3 to perform the listed function.

    b) The pumps identified in Table 2.3.2-3 start after receiving a signal from the PLS.

13. Displays of the parameters identified in Table 2.3.2-3 can be retrieved in the MCR.

14. The nonsafety-related piping located inside containment and designated as reactor coolant pressure boundary, as identified in Table 2.3.2-2 (pipe lines with "No" in the ASME Code column), has been designed to withstand a seismic design basis event and maintain structural integrity.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3.2-4 specifies the inspections, tests, analyses, and associated acceptance criteria for the CVS.

Table 2.3.2-4 Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The functional arrangement of the CVS is as described in the Design Description of this Section 2.3.2.	Inspection of the as-built system will be performed.	The as-built CVS conforms with the functional arrangement as described in the Design Description of this Section 2.3.2.
2.a) The components identified in Table 2.3.2-1 as ASME Code Section III are designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built components as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built components identified in Table 2.3.2-1 as ASME Code Section III.
2.b) The piping identified in Table 2.3.2-2 as ASME Code Section III is designed and constructed in accordance with ASME Code Section III requirements.	Inspection will be conducted of the as-built piping as documented in the ASME design reports.	The ASME Code Section III design reports exist for the as-built piping identified in Table 2.3.2-2 as ASME Code Section III.
3.a) Pressure boundary welds in components identified in Table 2.3.2-1 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
3.b) Pressure boundary welds in piping identified in Table 2.3.2-2 as ASME Code Section III meet ASME Code Section III requirements.	Inspection of the as-built pressure boundary welds will be performed in accordance with the ASME Code Section III.	A report exists and concludes that the ASME Code Section III requirements are met for non-destructive examination of pressure boundary welds.
4.a) The components identified in Table 2.3.2-1 as ASME Code Section III retain their pressure boundary integrity at their design pressure.	A hydrostatic test will be performed on the components required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the components identified in Table 2.3.2-1 as ASME Code Section III conform with the requirements of the ASME Code Section III.

Table 2.3.2-4 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
4.b) The piping identified in Table 2.3.2-2 as ASME Code Section III retains its pressure boundary integrity at its design pressure.	A hydrostatic test will be performed on the piping required by the ASME Code Section III to be hydrostatically tested.	A report exists and concludes that the results of the hydrostatic test of the piping identified in Table 2.3.2-2 as ASME Code Section III conform with the requirements of the ASME Code Section III.
5. The seismic Category I equipment identified in Table 2.3.2-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.3.2-1 is located on the Nuclear Island.	i) The seismic Category I equipment identified in Table 2.3.2-1 is located on the Nuclear Island.
	ii) Type tests, analyses, or a combination of type tests and analyses of seismic Category I equipment will be performed.	ii) A report exists and concludes that the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of safety function.
	iii) Inspection will be performed for the existence of a report verifying that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.	iii) A report exists and concludes that the as-installed equipment including anchorage is seismically bounded by the tested or analyzed conditions.
6.a) The Class 1E equipment identified in Table 2.3.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.	Type tests, analyses, or a combination of type tests and analyses will be performed on Class 1E equipment located in a harsh environment.	A report exists and concludes that the Class 1E equipment identified in Table 2.3.2-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function for the time required to perform the safety function.
6.b) The Class 1E components identified in Table 2.3.2-1 are powered from their respective Class 1E division.	Testing will be performed on the CVS by providing a simulated test signal in each Class 1E division.	A simulated test signal exists at the Class 1E equipment identified in Table 2.3.2-1 when the assigned Class 1E division is provided the test signal.

Table 2.3.2-4 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.c) Separation is provided between CVS Class 1E divisions, and between Class 1E divisions and non-Class 1E cable.	See Tier 1 Material, Section 3.3, Nuclear Island Buildings.	See Tier 1 Material, Section 3.3, Nuclear Island Buildings.
7.a) The CVS preserves containment integrity by isolation of the CVS lines penetrating the containment.	See Tier 1 Material, subsection 2.2.1, Containment System.	See Tier 1 Material, subsection 2.2.1, Containment System.
7.b) The CVS provides termination of an inadvertent RCS boron dilution by isolating demineralized water from the RCS.	See item 10b in this table.	See item 10b in this table.
7.c) The CVS provides isolation of makeup to the RCS.	See item 10b in this table.	See item 10b in this table.
8.a) The CVS provides makeup water to the RCS.	i) Testing will be performed by aligning a flow path from each CVS makeup pump, actuating makeup flow to the RCS at pressure greater than or equal to 2000 psia, and measuring the flow rate in the makeup pump discharge line with each pump suction aligned to the boric acid tank.	i) Each CVS makeup pump provides a flow rate of greater than or equal to 100 gpm.
	ii) Inspection of the boric acid tank volume will be performed.	ii) The volume in the boric acid tank is at least 70,000 gallons between the tank outlet connection and the tank overflow.
	iii) Testing will be performed to measure the delivery rate from the DWS to the RCS. Both CVS makeup pumps will be operating and the RCS pressure will be below 6 psig.	iii) The total CVS makeup flow to the RCS is less than or equal to 200 gpm.

Table 2.3.2-4 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
8.b) The CVS provides the pressurizer auxiliary spray.	Testing will be performed by aligning a flow path from each CVS makeup pump to the pressurizer auxiliary spray and measuring the flow rate in the makeup pump discharge line with each pump suction aligned to the boric acid tank and with RCS pressure greater than or equal to 2000 psia.	Each CVS makeup pump provides spray flow to the pressurizer.
9. Safety-related displays identified in Table 2.3.2-1 can be retrieved in the MCR.	Inspection will be performed for retrievability of the safety-related displays in the MCR.	Safety-related displays identified in Table 2.3.2-1 can be retrieved in the MCR.
10.a) Controls exist in the MCR to cause the remotely operated valves identified in Table 2.3.2-1 to perform active functions.	Stroke testing will be performed on the remotely operated valves identified in Table 2.3.2-1 using the controls in the MCR.	Controls in the MCR operate to cause the remotely operated valves identified in Table 2.3.2-1 to perform active functions.
10.b) The valves identified in Table 2.3.2-1 as having PMS control perform an active safety function after receiving a signal from the PMS.	i) Testing will be performed using real or simulated signals into the PMS.	i) The valves identified in Table 2.3.2-1 as having PMS control perform the active function identified in the table after receiving a signal from the PMS.
	ii) Testing will be performed to demonstrate that the remotely operated CVS isolation valves CVS-V090, V091, V136A/B close within the required response time.	ii) These valves close within the following times after receipt of an actuation signal: V090, V091 < 10 sec V136A/B < 20 sec
11.a) The motor-operated and check valves identified in Table 2.3.2-1 perform an active safety-related function to change position as indicated in the table.	i) Tests or type tests of motor-operated valves will be performed that demonstrate the capability of the valve to operate under its design conditions.	i) A test report exists and concludes that each motor-operated valve changes position as indicated in Table 2.3.2-1 under design conditions.
	ii) Inspection will be performed for the existence of a report verifying that the as-installed motor-operated valves are bounded by the tested conditions.	ii) A report exists and concludes that the as-installed motor-operated valves are bounded by the tests or type tests.

Table 2.3.2-4 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
	iii) Tests of the as-installed motor-operated valves will be performed under pre-operational flow, differential pressure, and temperature conditions. iv) Exercise testing of the check valves with active safety functions identified in Table 2.3.2-1 will be performed under pre-operational test pressure, temperature and fluid flow conditions.	iii) Each motor-operated valve changes position as indicated in Table 2.3.2-1 under pre-operational test conditions. iv) Each check valve changes position as indicated in Table 2.3.2-1.
11.b) After loss of motive power, the remotely operated valves identified in Table 2.3.2-1 assume the indicated loss of motive power position.	Testing of the installed valves will be performed under the conditions of loss of motive power.	Upon loss of motive power, each remotely operated valve identified in Table 2.3.2-1 assumes the indicated loss of motive power position.
12.a) Controls exist in the MCR to cause the pumps identified in Table 2.3.2-3 to perform the listed function.	Testing will be performed to actuate the pumps identified in Table 2.3.2-3 using controls in the MCR.	Controls in the MCR cause pumps identified in Table 2.3.2-3 to perform the listed function.
12.b) The pumps identified in Table 2.3.2-3 start after receiving a signal from the PLS.	Testing will be performed to confirm starting of the pumps identified in Table 2.3.2-3.	The pumps identified in Table 2.3.2-2 start after a signal is generated by the PLS.
13. Displays of the parameters identified in Table 2.3.2-3 can be retrieved in the MCR.	Inspection will be performed for retrievability of the displays identified in Table 2.3.2-3 in the MCR.	Displays identified in Table 2.3.2-3 can be retrieved in the MCR.
14. The nonsafety-related piping located inside containment and designated as reactor coolant pressure boundary, as identified in Table 2.3.2-2, has been designed to withstand a seismic design basis event and maintain structural integrity.	Inspection will be conducted of the as-built components as documented in the CVS Seismic Analysis Report.	The CVS Seismic Analysis Reports exist for the non-safety related piping located inside containment and designated as reactor coolant pressure boundary as identified in Table 2.3.2-2.

Table 2.3.2-1									
Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
RCS Purification Motor-operated Isolation Valve	CVS-PL-V001	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Purification Motor-operated Isolation Valve	CVS-PL-V002	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
RCS Purification Motor-operated Isolation Valve	CVS-PL-V003	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
CVS Resin Flush Line Containment Isolation Valve	CVS-PL-V040	Yes	Yes	No	- / -	-	-	-	-
CVS Resin Flush Line Containment Isolation Valve	CVS-PL-V041	Yes	Yes	No	- / -	-	-	-	-
CVS Demineralizer Resin Flush Line Containment Isolation Thermal Relief Valve	CVS-PL-V042	Yes	Yes	No	- / -	-	-	Transfer Open/ Transfer Closed	-
CVS Letdown Containment Isolation Valve	CVS-PL-V045	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed
CVS Letdown Containment Isolation Valve	CVS-PL-V047	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

Table 2.3.2-1 (cont.)									
Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
CVS Purification Return Line Pressure Boundary Check Valve	CVS-PL-V080	Yes	Yes	No	- / -	-	-	Transfer Closed	-
CVS Purification Return Line Pressure Boundary Isolation Check Valve	CVS-PL-V081	Yes	Yes	No	- / -	No	-	Transfer Closed	-
CVS Purification Return Line Pressure Boundary Check Valve	CVS-PL-V082	Yes	Yes	No	- / -	-	-	Transfer Closed	-
CVS Auxiliary Pressurizer Spray Line Pressure Boundary Valve	CVS-PL-V084	Yes	Yes	Yes	Yes/Yes	No	Yes	Transfer Closed	Closed
CVS Auxiliary Pressurizer Spray Line Pressure Boundary Check Valve	CVS-PL-V085	Yes	Yes	No	Yes/Yes	-	-	Transfer Closed	-
CVS Makeup Line Containment Isolation Motor-operated Valve	CVS-PL-V090	Yes	Yes	Yes	Yes/No	Yes (Valve Position)	Yes	Transfer Closed	As Is
CVS Makeup Line Containment Isolation Motor-operated Valve	CVS-PL-V091	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	As Is
CVS Hydrogen Addition Line Containment Isolation Valve	CVS-PL-V092	Yes	Yes	Yes	Yes/Yes	Yes (Valve Position)	Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

Table 2.3.2-1 (cont.)

Equipment Name	Tag No.	ASME Code Section III	Seismic Cat. I	Remotely Operated Valve	Class 1E/ Qual. for Harsh Envir.	Safety-Related Display	Control PMS	Active Function	Loss of Motive Power Position
CVS Hydrogen Addition Line Containment Isolation Check Valve	CVS-PL-V094	Yes	Yes	No	- / -	-	-	Transfer Closed	-
CVS Makeup Line Containment Isolation Thermal Relief Valve	CVS-PL-V100	Yes	Yes	No	- / -	-	-	Transfer Open/ Transfer Closed	-
CVS Demineralized Water Isolation Valve	CVS-PL-V136A	Yes	Yes	Yes	Yes/No	No	Yes	Transfer Closed	Closed
CVS Demineralized Water Isolation Valve	CVS-PL-V136B	Yes	Yes	Yes	Yes/No	No	Yes	Transfer Closed	Closed

Note: Dash (-) indicates not applicable.

The ITAAC for TGS2 – ABWR ITAAC 2.10.9.2			
Requirement	Inspection, Test or Analysis	Acceptance Criteria	ITAAC Determination Basis
2. Main control room displays provided for the TGS System are as defined in Section 2.10.9.	2. Inspections will be performed on the main control room displays for the TGS System.	2. Displays exist or can be retrieved in the main control room as defined in Section 2.10.9.	Pre-operational test results documenting that the TGS system displays are retrievable in the main control room.

2.10.9 Turbine Gland Seal System

Design Description

The Turbine Gland Seal (TGS) System prevents the escape of radioactive steam from the turbine shaft casing penetrations and valve stems and prevents air inleakage through subatmospheric turbine glands. Figure 2.10.9 shows the basic system configuration.

The TGS System consists of a sealing steam pressure regulator, steam seal header and a gland seal condenser (GSC) with two full capacity exhaust blowers and associated piping, valves and instrumentation.

The TGS System is bounded by the Main Turbine and the Turbine Bypass System. The TGS System receives steam from either the Turbine Main Steam System, the feedwater heater drain tank vent header or auxiliary steam sources. The exhaust blowers discharge to the Turbine Building compartment exhaust system.

The TGS System is classified as non-safety-related.

The TGS System is located in the Turbine Building.

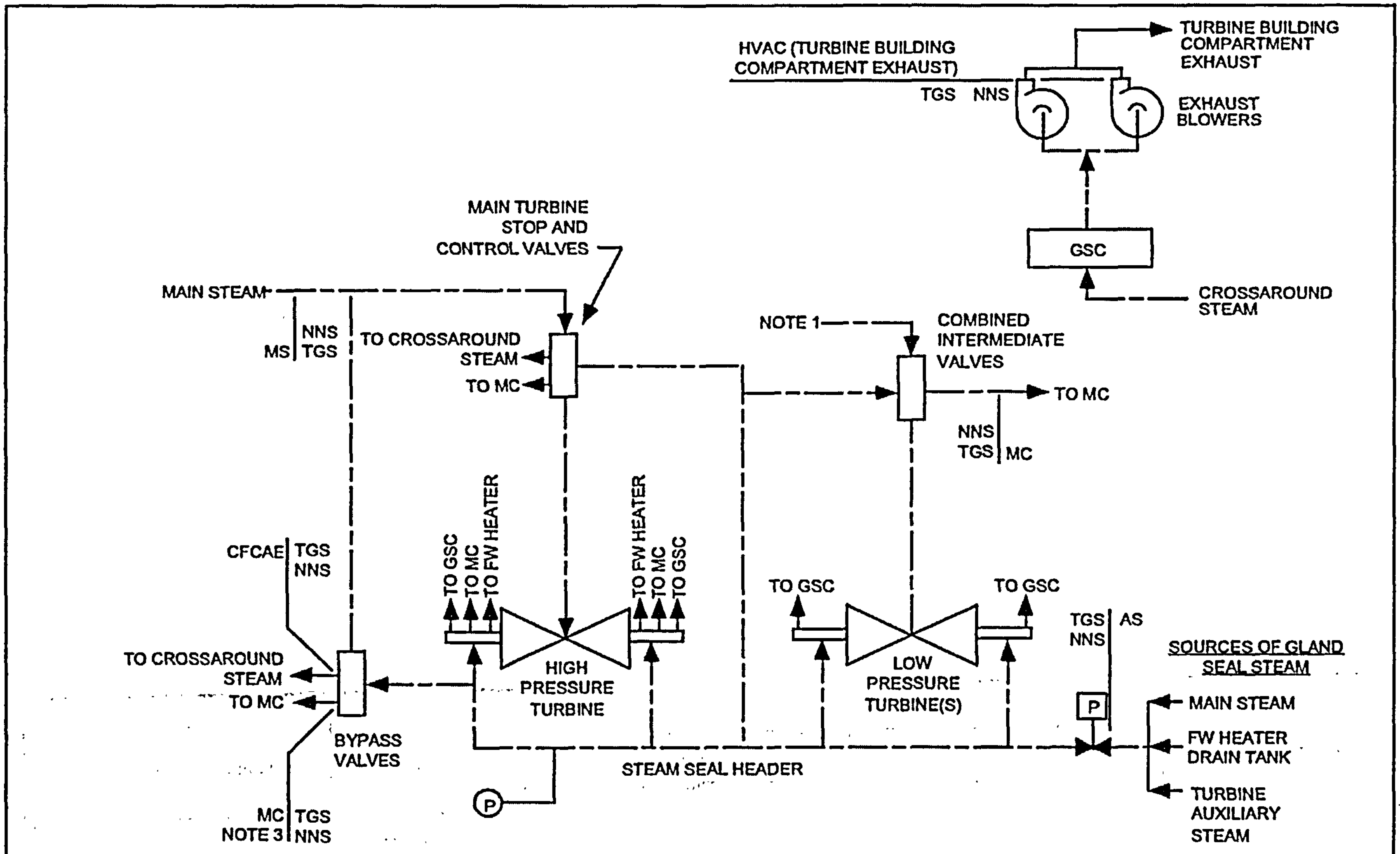
The TGS System has displays for gland seal condenser and steam seal header pressure in the main control room.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10.9 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the TGS System.

Table 2.10.9 Turbine Gland Seal System

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The basic configuration of the TGS System is as shown on Figure 2.10.9.	1. Inspections of the as-built system will be conducted.	1. The as-built TGS System conforms with the basic configuration shown on Figure 2.10.9.
2. Main control room displays provided for the TGS System are as defined in Section 2.10.9.	2. Inspections will be performed on the main control room displays for the TGS System.	2. Displays exist or can be retrieved in the main control room as defined in Section 2.10.9.



NOTES;
1. STEAM PATH BETWEEN HIGH AND LOW PRESSURE TURBINES.
2. DELETED
3. TYPICAL FOR INTERFACES WITH MC.

Figure 2.10.9 Turbine Gland Seal System

The ITAAC for STR3 – ABWR ITAAC 2.15.10.3			
Requirement	Inspection, Test or Analysis	Acceptance Criteria	ITAAC Determination Basis
3. Inter-divisional walls, floors, doors and penetrations, and penetrations in the external R/B walls to connecting tunnels, have a three-hour fire rating.	3. Inspections of the as-installed inter-divisional boundaries and external wall penetrations to connecting tunnels will be conducted.	3. The as-installed walls, floors, doors and penetrations that form the inter-divisional boundaries and external wall penetrations to connecting tunnels have a three-hour fire rating.	The construction work planning and inspection records will document that the R/B as-installed walls, floors, doors and penetrations that form the inter-divisional boundaries and external wall penetrations to connecting tunnels have a three-hour fire rating.

2.15.10 Reactor Building

Design Description

The Reactor Building (R/B) is a structure which houses and provides protection and support for the reactor primary systems, the primary containment and much of the plant safety-related equipment. Figures 2.15.10a through 2.15.10o show the basic configuration and scope of the R/B*.

The R/B is constructed of reinforced concrete and structural steel with a steel frame and reinforced concrete roof. The R/B encloses the primary containment. The R/B slabs and fuel pool girders are integrated with the reinforced concrete containment vessel (RCCV). The R/B slabs are supported by columns, shear walls and beams to carry vertical loads to the basemat and transfer horizontal loads through the RCCV and R/B shear walls to the basemat and R/B foundation. The R/B, together with the RCCV and the reactor pedestal, are supported by a common basemat. Inside the RCCV, the basemat is considered part of the Primary Containment System (PCS); outside the RCCV, the basemat is part of the R/B. The top of the R/B basemat is located 20.2m ± 0.3m below the finished grade elevation.

The R/B is divided into three separate divisional areas for mechanical and electrical equipment and four divisional areas for instrumentation racks. Inter-divisional boundaries have the following features:

(1) Inter-divisional walls, floors, doors and penetrations, and penetrations in the external R/B walls to connecting tunnels, which have three-hour fire rating.

(2) Watertight doors in the basement to prevent flooding in one division from propagating to other divisions.

(3) Divisional walls in the basement are 0.6 meters thick or greater.

Watertight doors on Emergency Core Cooling System rooms have open/close sensors with status indication and alarms in the main control room.

The R/B flooding that results from component failures in any of the R/B divisions does not prevent safe shutdown of the reactor. The basement floor is the collection location point for floods. The building configuration at this elevation is such that even for a flooding event involving release of either the suppression pool or the condensate storage tank (CST) water into the R/B, no more than one division of safety-related equipment is affected. Except for the basement area, safety-related electrical, instrumentation and control equipment is located at least 20 cm above the floor surface.

* The overall building dimensions provided in Figures 2.15.10a through 2.15.10o are provided for information only and are not intended to be part of the certified ABWR information.

The R/B is protected against external flood. The following design features are provided:

(1) External walls below flood level are equal to or greater than 0.6 meters thick to prevent ground water seepage.

(2) Penetrations in the external walls below flood level are provided with flood protection features.

(3) A tunnel connects the Radwaste Building, Turbine Building, Control Building and Reactor Building for the liquid radwaste system piping. The penetrations from the tunnel to the Reactor Building are watertight.

The R/B is protected against the pressurization effects associated with postulated rupture of pipes containing high-energy fluid that occur in subcompartments of the R/B.

There are three divisionally separated tunnels for routing Oil Storage and Transfer (OST) System piping and cable from the fuel oil storage tanks to the R/B. These tunnels are configured so that any fuel oil leakage does not accumulate at the R/B boundary. Tunnel flooding due to site flood conditions is precluded by protecting the entrances against water entry.

The R/B and oil transfer tunnels are classified as Seismic Category I. They are designed and constructed to accommodate the dynamic and static loading conditions associated with the various loads and load combinations which form the structural design basis. The loads are (as applicable) those associated with:

(1) Natural phenomena—wind, floods, tornados (including tornado missiles), earthquakes, rain and snow.

(2) Internal events—floods, pipe breaks and missiles.

(3) Normal plant operation—live loads, dead loads, temperature effects and building vibration loads.

Systems, structures, and components located in the R/B and classified as safety-related are protected against inter-divisional flooding that results from postulated failures in Seismic Category I or non-nuclear safety (NNS) components located in the R/B or from external flooding events. Each postulated flooding event is documented in a Flood Analysis Report which concludes the reactor can be shutdown safely and maintained in a safe, cold shutdown condition without offsite power.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.15.10 provides a definition of the inspections, tests, and/or analyses, together with associated acceptance criteria, which will be undertaken for the R/B.

Table 2.15.10 Reactor Building

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
1. The basic configuration of the R/B is shown on Figures 2.15.10a through 2.15.10o.	1. Inspections of the as-built structure will be conducted.	1. The as-built R/B conforms with the basic configuration shown in Figures 2.15.10a through 2.15.10o.
2. The top of the R/B basemat is located 20.2m ±0.3m below the finished grade elevation.	2. Inspections of the as-built structure will be conducted.	2. The top of the R/B basemat is located 20.2m ±0.3m below the finished grade elevation.
3. Inter-divisional walls, floors, doors and penetrations, and penetrations in the external R/B walls to connecting tunnels, have a three-hour fire rating.	3. Inspections of the as-installed inter-divisional boundaries and external wall penetrations to connecting tunnels will be conducted.	3. The as-installed walls, floors, doors and penetrations that form the inter-divisional boundaries and external wall penetrations to connecting tunnels have a three-hour fire rating.
4. The R/B has divisional areas with walls and watertight doors are as shown on Figures 2.15.10a through 2.15.10o.	4. Inspections of the as-built walls and watertight doors will be conducted.	4. The as-built R/B has walls and watertight doors as shown on Figures 2.15.10a through 2.15.10o.
5. Main control room displays and alarms provided for the R/B are as defined in Section 2.15.10.	5. Inspections will be performed on the main control room displays and alarms for the R/B.	5. Displays and alarms exist or can be retrieved in the main control room as defined in Section 2.15.10.
6. A flooding event involving release of either the suppression pool or the CST water does not affect more than one division of safety-related equipment.	6. Inspections will be conducted of the divisional boundaries shown on Figure 2.15.10c.	6. Penetrations (except for watertight doors) in the divisional walls are at least 2.5m above the floor level of -8200 mm.
7. Except for the basement area, safety-related electrical, instrumentation, and control equipment is located at least 20 cm above the floor surface.	7. Inspections will be conducted of the as-built equipment.	7. Except for the basement area, safety-related electrical, instrumentation, and control equipment is located at least 20 cm above the floor surface.

Table 2.15.10 Reactor Building (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
8. The R/B is protected against external floods by having: a. External walls below flood level that are equal to or greater than 0.6m thick to prevent ground water seepage. b. Penetrations in the external walls below flood level provided with flood protection features. c. Watertight penetrations to the Reactor Building from the tunnel that connects the Radwaste Building, Turbine Building and Reactor Building for the liquid radwaste system piping.	8. Inspections of the as-built structure will be conducted.	8. a. External walls below flood level are equal to or greater than 0.6m thick to prevent ground water seepage. b. Penetrations in the external walls below flood level are provided with flood protection features. c. Penetrations from the tunnel to the Reactor Building are watertight.
9. There are three divisionally separated tunnels for routing OST system piping from the fuel storage tanks to the R/B. These tunnels are configured so that any fuel oil leakage does not accumulate at the R/B boundary. Tunnel flooding due to site flood conditions is precluded by protecting the entrances against water entry.	9. Inspections of the as-bulit tunnels will be conducted.	9. There are three divisionally separated tunnels for routing OST System piping from the fuel storage tanks to the R/B. These tunnels are configured so that any fuel oil leakage does not accumulate at the R/B boundary. Tunnel flooding due to site flood conditions is precluded by protecting the entrances against water entry.
10. The R/B and oil transfer tunnels are able to withstand the structural design basis loads as defined in Section 2.15.10.	10. A structural analysis will be performed which reconciles the as-built data with structural design basis as defined in Section 2.15.10.	10. A structural analysis report exists which concludes that the as-built R/B and oil tranfer tunnels are able to withstand the structural design basis loads as defined in Section 2.15.10.

Table 2.15.10 Reactor Building (Continued)

Inspections, Tests, Analyses and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
11. Systems, structures and components located in the R/B and classified as safety-related are protected against inter-divisional flooding that results from postulated failures in Seismic Category I or NNS related components located in the R/B or from external flooding events. Each postulated flooding event is documented in a Flood Analysis Report which concludes the reactor can be shutdown safely and maintained in a safe, cold shutdown condition without offsite power.	11. Inspections of the Flood Analysis Report and the as-built flood protection features will be conducted.	11. A Flood Analysis Report exists for the as-built R/B and concludes that for each postulated flooding event, the reactor can be shutdown safely and maintained in a safe, cold shutdown condition without offsite power. The Flood Analysis Report includes the results of inspections of the as-built flood protection features.

The ITAAC for STR4 – AP1000 ITAAC 3.3.6.5.a), b) & c)			
Requirement	Inspection, Test or Analysis	Acceptance Criteria	ITAAC Determination Basis
5.a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.	An inspection of the as-built exterior walls and the basemat of the nuclear island up to floor elevation 100'-0", for application of water barrier will be performed during construction before the walls are poured.	A report exists that confirms that a water barrier exists on the nuclear island exterior walls up to site grade.	The construction work planning and inspection records for the exterior walls and basemat up to elevation 100' will document that the barrier was installed before the concrete pours were made.
5.b) The boundaries between rooms identified in Table 3.3-2 of the auxiliary building are designed to prevent flooding of rooms that contain safety-related equipment.	An inspection of the auxiliary building rooms will be performed.	A report exists that confirms floors and walls as identified on Table 3.3-2 have provisions to prevent flooding between rooms up to the maximum flood levels for each room defined in Table 3.3-2.	The construction work planning and inspection records will document that the floors and walls as identified on Table 3.3-2 have provisions to prevent flooding between rooms up to the maximum flood levels for each room defined in Table 3.3-2.
5.c) The boundaries between the following rooms, which contain safety-related equipment – PXS valve/accumulator room A (11205), PXS valve/accumulator room B (11207), and CVS room (11209) – are designed to prevent flooding between these rooms.	An inspection of the boundaries between the following rooms which contain safety-related equipment – PXS Valve/ Accumulator Room A (11205), PXS Valve/Accumulator Room B (11207), and CVS Room (11209) – will be performed.	A report exists that confirms that flooding of the PXS Valve/ Accumulator Room A (11205), and the PXS/Accumulator Room B (11207) is prevented to a maximum flood level of 110 feet, and of the CVS room (11209) to a maximum flood level of 109'-10".	A flooding calculation along with construction work planning and inspection records will document that flooding of the PXS Valve/ Accumulator Room A (11205), and the PXS/Accumulator Room B (11207) is prevented to a maximum flood level of 110 feet, and of the CVS room (11209) to a maximum flood level of 109'-10".

3.3 Buildings

Design Description

The nuclear island structures include the containment (the steel containment vessel and the containment internal structure) and the shield and auxiliary buildings. The containment, shield and auxiliary buildings are structurally integrated on a common basemat which is embedded below the finished plant grade level. The containment vessel is a cylindrical welded steel vessel with elliptical upper and lower heads, supported by embedding a lower segment between the containment internal structures concrete and the basemat concrete. The containment internal structure is reinforced concrete with structural modules used for some walls and floors. The shield building is reinforced concrete and, in conjunction with the internal structures of the containment building, provides shielding for the reactor coolant system and the other radioactive systems and components housed in the containment. The shield building roof is a reinforced concrete structure containing an integral, steel lined passive containment cooling water storage tank. The auxiliary building is reinforced concrete and houses the safety-related mechanical and electrical equipment located outside the containment and shield buildings.

The portion of the annex building adjacent to the nuclear island is a structural steel and reinforced concrete seismic Category II structure and houses the technical support center, non-1E electrical equipment, and hot machine shop.

The radwaste building is a steel framed structure and houses the low level waste processing and storage.

The turbine building is a non-safety related structure that houses the main turbine generator and the power conversion cycle equipment and auxiliaries. There is no safety-related equipment in the turbine building. The turbine building is located on a separate foundation. The turbine building structure is adjacent to the nuclear island structures.

The diesel generator building is a non-safety related structure that houses the two standby diesel engine powered generators and the power conversion cycle equipment and auxiliaries. There is no safety-related equipment in the diesel generator building. The diesel generator building is located on a separate foundation at a distance from the nuclear island structures.

The plant gas system (PGS) provides hydrogen, carbon dioxide, and nitrogen gases to the plant systems as required. The component locations of the PGS are located either in the turbine building or the yard areas.

1. The physical arrangement of the nuclear island structures and the annex building is as described in the Design Description of this Section 3.3, and as shown on Figures 3.3-1 through 3.3-14. The physical arrangement of the radwaste building, the turbine building, and the diesel generator building is as described in the Design Description of this Section 3.3.

2. a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads, as specified in the Design Description, without loss of structural integrity and the safety-related functions. The design bases loads are those loads associated with:

    - Normal plant operation (including dead loads, live loads, lateral earth pressure loads, and equipment loads, including hydrodynamic loads, temperature and equipment vibration);

- External events (including rain, snow, flood, tornado, tornado generated missiles and earthquake); and
- Internal events (including flood, pipe rupture, equipment failure, and equipment failure generated missiles).

b) Site grade level is located relative to floor elevation 100'-0" per Table 3.3-5. Floor elevation 100'-0" is defined as the elevation of the floor at design plant grade.

c) The containment and its penetrations are designed and constructed to ASME Code Section III, Class MC.[1]

d) The containment and its penetrations retain their pressure boundary integrity associated with the design pressure.

e) The containment and its penetrations maintain the containment leakage rate less than the maximum allowable leakage rate associated with the peak containment pressure for the design basis accident.

f) The key dimensions of the nuclear island structures are as defined on Table 3.3-5.

g) The containment vessel greater than 7 feet above the operating deck provides a heat transfer surface. A free volume exists inside the containment shell above the operating deck.

3. Walls and floors of the nuclear island structures as defined on Table 3.3-1, except for designed openings and penetrations, provide shielding during normal operations.

4. Walls and floors of the annex building as defined on Table 3.3-1, except for designed openings and penetrations, provide shielding during normal operations.

5. a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.

   b) The boundaries between mechanical equipment rooms and the electrical and instrumentation and control (I&C) equipment rooms of the auxiliary building as identified in Table 3.3-2 are designed to prevent flooding of rooms that contain safety-related equipment up to the maximum flood level for each room defined in Table 3.3-2.

   c) The boundaries between the following rooms, which contain safety-related equipment – passive core cooling system (PXS) valve/accumulator room A (11205), PXS valve/accumulator room B (11207), and chemical and volume system (CVS) room (11209) – are designed to prevent flooding between these rooms.

6. a) The radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" contains adequate volume to contain the liquid volume of faulted liquid radwaste system (WLS) storage tanks. The available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceeds the volume of the liquid

1. Containment isolation devices are addressed in subsection 2.2.1, Containment System.

radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).

b) The radwaste building packaged waste storage room has a volume greater than or equal to 1293 cubic feet.

7. a) Class 1E electrical cables, fiber optic cables associated with only one division, and raceways are identified according to applicable color-coded Class 1E divisions.

   b) Class 1E divisional electrical cables and communication cables associated with only one division are routed in their respective divisional raceways.

   c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.

   d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.

   e) Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.

8. Equipment labeled as essential targets in Table 3.3-4 and located in rooms identified in Table 3.3-4 are protected from the dynamic effects of postulated pipe breaks.

9. The reactor cavity sump has a minimum concrete thickness as shown on Table 3.3-5 between the bottom of the sump and the steel containment.

10. The shield building roof and the passive containment cooling system (PCS) storage tank support and retain the PCS water. The passive containment cooling system tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided over the tank boundary liner welds.

11. Deleted

12. The extended turbine generator axis intersects the shield building.

13. Separation is provided between the structural elements of the turbine, annex, and radwaste buildings and the nuclear island structure. This separation permits horizontal motion of the buildings in a safe shutdown earthquake without impact between structural elements of the buildings.

14. Protected Area/Vital Area walls that are accessible and unmonitored are security hardened.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.3-6 specifies the inspections, tests, analyses, and associated acceptance criteria for the buildings.

Table 3.3-6
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The physical arrangement of the nuclear island structures and the annex building is as described in the Design Description of this Section 3.3 and Figures 3.3-1 through 3.3-14. The physical arrangement of the radwaste building, the turbine building, and the diesel generator building is as described in the Design Description of this Section 3.3.	An inspection of the nuclear island structures, the annex building, the radwaste building, the turbine building, and the diesel generator building will be performed.	The as-built nuclear island structures, the annex building, the radwaste building, the turbine building, and the diesel generator building conform with the physical arrangement as described in the Design Description of this Section 3.3 and Figures 3.3-1 through 3.3-14.
2.a) The nuclear island structures, including the critical sections listed in Table 3.3-7, are seismic Category I and are designed and constructed to withstand design basis loads as specified in the Design Description, without loss of structural integrity and the safety-related functions.	i) An inspection of the nuclear island structures will be performed. Deviations from the design due to as-built conditions will be analyzed for the design basis loads.	i) A report exists which reconciles deviations during construction and concludes that the as-built nuclear island structures, including the critical sections, conform to the approved design and will withstand the design basis loads specified in the Design Description without loss of structural integrity or the safety-related functions.
	ii) An inspection of the as-built concrete thickness will be performed.	ii) A report exists that concludes that the as-built concrete thicknesses conform with the building sections defined on Table 3.3-1.
2.b) Site grade level is located relative to floor elevation 100'-0" per Table 3.3-5.	Inspection of the as-built site grade will be conducted.	Site grade is consistent with design plant grade within the dimension defined on Table 3.3-5.
2.c) The containment and its penetrations are designed and constructed to ASME Code Section III, Class MC.[1]	See Tier 1 Material, Subsection 2.2.1, Containment System.	See Tier 1 Material, Subsection 2.2.1, Containment System.

1. Containment isolation devices are addressed in subsection 2.2.1, Containment System.

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
2.d) The containment and its penetrations retain their pressure boundary integrity associated with the design pressure.	See Tier 1 Material, Subsection 2.2.1, Containment System.	See Tier 1 Material, Subsection 2.2.1, Containment System.
2.e) The containment and its penetrations maintain the containment leakage rate less than the maximum allowable leakage rate associated with the peak containment pressure for the design basis accident.	See Tier 1 Material, Subsection 2.2.1, Containment System.	See Tier 1 Material, Subsection 2.2.1, Containment System.
2.f) The key dimensions of nuclear island structures are defined on Table 3.3-5.	An inspection will be performed of the as-built configuration of the nuclear island structures.	A report exists and concludes that the key dimensions of the as-built nuclear island structures are consistent with the dimensions defined on Table 3.3-5.
2.g) The containment vessel greater than 7 feet above the operating deck provides a heat transfer surface. A free volume exists inside the containment shell above the operating deck.	The maximum containment vessel inside height from the operating deck is measured and the inner radius below the spring line is measured at two orthogonal radial directions at one elevation.	The containment vessel maximum inside height from the operating deck is 146'-7" (with tolerance of +12", -6"), and the inside diameter is 130 feet nominal (with tolerance of +12", -6").
3. Walls and floors of the nuclear island structures as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built nuclear island structures wall and floor thicknesses will be performed.	A report exists and concludes that the shield walls and floors of the nuclear island structures as defined on Table 3.3-1 except for designed openings or penetrations are consistent with the concrete wall thicknesses provided in Table 3.3-1.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4. Walls and floors of the annex building as defined on Table 3.3-1 except for designed openings or penetrations provide shielding during normal operations.	Inspection of the as-built annex building wall and floor thicknesses will be performed.	A report exists and concludes that the shield walls and floors of the annex building as defined on Table 3.3-1 except for designed openings or penetrations are consistent with the minimum concrete wall thicknesses provided in Table 3.3-1.
5.a) Exterior walls and the basemat of the nuclear island have a water barrier up to site grade.	An inspection of the as-built exterior walls and the basemat of the nuclear island up to floor elevation 100'-0", for application of water barrier will be performed during construction before the walls are poured.	A report exists that confirms that a water barrier exists on the nuclear island exterior walls up to site grade.
5.b) The boundaries between rooms identified in Table 3.3-2 of the auxiliary building are designed to prevent flooding of rooms that contain safety-related equipment.	An inspection of the auxiliary building rooms will be performed.	A report exists that confirms floors and walls as identified on Table 3.3-2 have provisions to prevent flooding between rooms up to the maximum flood levels for each room defined in Table 3.3-2.
5.c) The boundaries between the following rooms, which contain safety-related equipment – PXS valve/accumulator room A (11205), PXS valve/accumulator room B (11207), and CVS room (11209) – are designed to prevent flooding between these rooms.	An inspection of the boundaries between the following rooms which contain safety-related equipment – PXS Valve/Accumulator Room A (11205), PXS Valve/Accumulator Room B (11207), and CVS Room (11209) – will be performed.	A report exists that confirms that flooding of the PXS Valve/Accumulator Room A (11205), and the PXS/Accumulator Room B (11207) is prevented to a maximum flood level of 110 feet, and of the CVS room (11209) to a maximum flood level of 109'-10".
6.a) The available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceed the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).	An inspection will be performed of the as-built radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" to define volume.	A report exists and concludes that the as-built available room volumes of the radiologically controlled area of the auxiliary building between floor elevations 66'-6" and 82'-6" exceed the volume of the liquid radwaste storage tanks (WLS-MT-05A, MT-05B, MT-06A, MT-06B, MT-07A, MT-07B, MT-07C, MT-11).

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
6.b) The radwaste building package waste storage room has a volume greater than or equal to 1293 cubic feet.	An inspection of the radwaste building packaged waste storage room (50352) is performed.	The volume of the radwaste building packaged waste storage room (50352) is greater than or equal to 1293 cubic feet.
7.a) Class 1E electrical cables, communication cables associated with only one division, and raceways are identified according to applicable color-coded Class 1E divisions.	Inspections of the as-built Class 1E cables and raceways will be conducted.	Class 1E electrical cables, communication cables associated with only one division, and raceways are identified by the appropriate color code.
7.b) Class 1E divisional electrical cables and communication cables associated with only one division are routed in their respective divisional raceways.	Inspections of the as-built Class 1E divisional cables and raceways will be conducted.	Class 1E electrical cables and communication cables associated with only one division are routed in raceways assigned to the same division. There are no other safety division electrical cables in a raceway assigned to a different division.
7.c) Separation is maintained between Class 1E divisions in accordance with the fire areas as identified in Table 3.3-3.	i) Inspections of the as-built Class 1E division electrical cables, communication cables associated with only one division, and raceways located in the fire areas identified in Table 3.3-3 will be conducted.	i) Results of the inspection will confirm that the separation between Class 1E divisions is consistent with Table 3.3-3.
	ii) Inspections of the as-built fire barriers between the fire areas identified in Table 3.3-3 will be conducted.	ii) Results of the inspection will confirm that fire barriers exist between Class 1E divisions consistent with the fire areas identified in Table 3.3-3.

Table 3.3-6 (cont.) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Tests, Analyses**	**Acceptance Criteria**
7.d) Physical separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables.	Inspections of the as-built Class 1E raceways will be performed to confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the following: – Within the main control room and remote shutdown room, the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch. – Within other plant areas (limited hazard areas), the minimum separation is defined by one of the following: 1) The minimum vertical separation is 5 feet and the minimum horizontal separation is 3 feet. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG. 3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.	Results of the inspection will confirm that the separation between Class 1E raceways of different divisions and between Class 1E raceways and non-Class 1E raceways is consistent with the followings: – Within the main control room and remote shutdown room, the vertical separation is 3 inches or more and the horizontal separation is 1 inch or more. – Within other plant areas (limited hazard areas), the separation meets one of the following: 1) The vertical separation is 5 feet or more and the horizontal separation is 3 feet or more except. 2) The minimum vertical separation is 12 inches and the minimum horizontal separation is 6 inches for raceways containing only instrumentation and control and low-voltage power cables <2/0 AWG. 3) For configurations that involve exclusively limited energy content cables (instrumentation and control), the minimum vertical separation is 3 inches and the minimum horizontal separation is 1 inch.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	4) For configurations involving an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the open raceway. 5) For configuration involving enclosed raceways, the minimum separation is 1 inch in both horizontal and vertical directions. – Where minimum separation distances are not maintained, the circuits are run in enclosed raceways or barriers are provided. – Separation distances less than those specified above and not run in enclosed raceways or provided with barriers are based on analysis – Non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is considered as associated circuits and subject to Class 1E requirements.	4) For configurations that involve an enclosed raceway and an open raceway, the minimum vertical separation is 1 inch if the enclosed raceway is below the raceway. 5) For configurations that involve enclosed raceways, the minimum vertical and horizontal separation is 1 inch. – Where minimum separation distances are not met, the circuits are run in enclosed raceways or barriers are provided. – A report exists and concludes that separation distances less than those specified above and not provided with enclosed raceways or barriers have been analyzed. – Non-Class 1E wiring that is not separated from Class 1E or associated wiring by the minimum separation distance or by a barrier or analyzed is treated as Class 1E wiring.
7.e) Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.	Inspections of the as-built Class 1E communication cables will be conducted.	Class 1E communication cables which interconnect two divisions are routed and separated such that the Protection and Safety Monitoring System voting logic is not defeated by the loss of any single raceway or fire area.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8. Equipment labeled as essential targets in Table 3.3-4 and located in rooms identified in Table 3.3-4 are protected from the dynamic effects of postulated pipe breaks.	An inspection will be performed of the as-built high energy pipe break pipe whip restraints features for systems located in rooms identified in Table 3.3-4.	An as-built Pipe Rupture Hazard Analysis Report exists and concludes that equipment labeled as essential targets in Table 3.3-4 and located in rooms identified in Table 3.3-4 can withstand the effects of postulated pipe rupture without loss of required safety function.
9. The reactor cavity sump has a minimum concrete thickness as shown in Table 3.3-5 between the bottom of the sump and the steel containment.	An inspection of the as-built containment building internal structures will be performed.	A report exists and concludes that the reactor cavity sump has a minimum concrete thickness as shown on Table 3.3-5 between the bottom of the sump and the steel containment.
10. The shield building roof and PCS storage tank support and retain the PCS water sources. The PCS storage tank has a stainless steel liner which provides a barrier on the inside surfaces of the tank. Leak chase channels are provided on the tank boundary liner welds.	i) A test will be performed to measure the leakage from the PCS storage tank based on measuring the water flow out of the leak chase collection system. ii) An inspection of the PCS storage tank exterior tank boundary and shield building tension ring will be performed before and after filling of the PCS storage tank to the overflow level. The vertical elevation of the shield building roof will be measured at a location at the outer radius of the roof (tension ring) and at a location on the same azimuth at the outer radius of the PCS water storage tank before and after filling the PCS storage tank.	i) A report exists and concludes that total water flow from the leak chase collection system does not exceed 10 gal/hr. ii) A report exists and concludes that there is no visible water leakage from the PCS storage tank and that inspection and measurement of the structure before and after filling of the tank shows structural behavior under normal loads to be acceptable.
11. Deleted		
12. The extended turbine generator axis intersects the shield building.	An inspection of the as-built turbine generator will be performed.	The extended axis of the turbine generator intersects the shield building.

Table 3.3-6 (cont.)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
13. Separation is provided between the structural elements of the turbine, annex and radwaste buildings and the nuclear island structure. This separation permits horizontal motion of the buildings in the safe shutdown earthquake without impact between structural elements of the buildings.	An inspection of the separation of the nuclear island from the annex, radwaste and turbine building structures will be performed. The inspection will verify the specified horizontal clearance between structural elements of the adjacent buildings, consisting of the reinforced concrete walls and slabs, structural steel columns and floor beams.	The minimum horizontal clearance above floor elevation 100'-0" between the structural elements of the annex and radwaste buildings and the nuclear island is 4 inches. The minimum horizontal clearance above floor elevation 100'-0" between the structural elements of the turbine building and the nuclear island is 12 inches.
14. Protected Area/Vital Area walls that are accessible and unmonitored are security hardened.	An inspection of the as-built Protected Area/Vital Area walls that are accessible and unmonitored will be performed.	The as-built inspection report exists and concludes that the Protected Area/Vital Area walls that are accessible and unmonitored meet the requirements of being security hardened.

Table 3.3-2 Nuclear Island Building Room Boundaries Required to Have Flood Barrier Floors and Walls		
	Between Room Number to Room Number	
Boundary/ Maximum Flood Level (inches)	Room with Postulated Flooding Source	Adjacent Room
Floor/36	12306	12211
Floor/3	12303	12203/12207
Floor/3	12313	12203/12207
Floor/1	12300	12201/12202/12207 12203/12204/12205
Floor/3	12312	12212
Wall/36	12306	12305
Floor/1	12401	12301/12302/12303 12312/12313
Wall/1	12401	12411/12412
Floor/36	12404	12304
Floor/4	12405	12305
Floor/36	12406	12306
Wall/36	12404	12401
Wall/1	12421	12452
Floor/3	12501	12401/12411/12412
Floor/3	12555	12421/12423/12422
Wall/36	12156/12158	12111/12112

The ITAAC for STR1 – ABWR ITAAC 2.15.10.1			
Requirement	Inspection, Test or Analysis	Acceptance Criteria	ITAAC Determination Basis
1. The basic configuration of the R/B is shown on Figures 2.15.10a through 2.1.5.10o.	1. Inspections of the as-built structure will be conducted.	1. The as-built R/B conforms with the basic configuration shown in Figures 2.15.10a through 2.1.5.10o.	

This ITAAC is not consistent with ITAAC written in and after AP600. I.e. We will discuss with NRC the term "basic configuration."

1.2 General Provisions

The following general provisions are applicable to the Design Descriptions and associated ITAAC:

Verifications for Basic Configuration for Systems

Verifications for Basic Configuration of systems include and are limited to inspection of the system functional arrangement and the following inspections, tests, and analyses:

(1) Inspections, including non-destructive examination (NDE), of the as-built, pressure boundary welds for ASME Code Class 1, 2, or 3 components identified in the Design Description to demonstrate that the requirements of ASME Code Section III for the quality of pressure boundary welds are met.

(2) Type tests, analyses, or a combination of type tests and analyses of the Seismic Category I mechanical and electrical equipment (including connected instrumentation and controls) identified in the Design Description to demonstrate that the as-built equipment, including associated anchorage, is qualified to withstand design basis dynamic loads without loss of its safety function.

(3) Type tests, or type tests and analyses, of the Class 1E electrical equipment identified in the Design Description (or on accompanying figures) to demonstrate that it is qualified to withstand the environmental conditions that would exist during and following a design basis accident without loss of its safety function for the time needed to be functional. These environmental conditions, as applicable to the bounding design basis accident(s), are as follows: expected time-dependent temperature and pressure profiles, humidity, chemical effects, radiation, aging, submergence, and their synergistic effects which have a significant effect on equipment performance. As used in this paragraph, the term "Class 1E electrical equipment" constitutes the equipment itself, connected instrumentation and controls, connected electrical components (such as cabling, wiring, and terminations), and the lubricants necessary to support performance of the safety functions of the Class 1E electrical components identified in the Design Description, to the extent such equipment is not located in a mild environment during or following a design basis accident.

Electrical equipment environmental qualification shall be demonstrated through analysis of the environmental conditions that would exist in the location of the equipment during and following a design basis accident and through a determination that the equipment is qualified to withstand those

conditions for the time needed to be functional. This determination may be demonstrated by:

(a) Type testing of an identical item of equipment under identical or similar conditions with a supporting analysis to show that the equipment is qualified; or

(b) type testing of a similar item of equipment under identical or similar conditions with a supporting analysis to show that the equipment is qualified; or

(c) experience with identical or similar equipment under identical or similar conditions with supporting analysis to show that the equipment is qualified; or

(d) analysis in combination with partial type test data that supports the analytical assumptions and conclusions to show that the equipment is qualified.

(4) Tests or type tests of active safety-related motor-operated valves (MOVs) identified in the Design Description to demonstrate that the MOVs are qualified to perform their safety functions under design basis differential pressure, system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and/or maximum stroke times.

Treatment of Individual Items

The absence of any discussion or depiction of an item in the Design Description or accompanying figures shall not be construed as prohibiting a licensee from utilizing such an item, unless it would prevent an item from performing its safety functions as discussed or depicted in the Design Description or accompanying figures.

When the term "operate", "operates", or "operation" is used with respect to an item discussed in the Acceptance Criteria, it refers to the actuation and running of the item. When the term "exist", "exists", or "existence" is used with respect to an item discussed in the Acceptance Criteria, it means that the item is present and meets the Design Description.

Implementation of ITAAC

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) are provided in tables with the following three-column format:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria

Each Design Commitment in the left-hand column of the ITAAC tables has an associated Inspections, Tests, or Analyses (ITA) requirement specified in the middle

column of the tables. The identification of a separate ITA entry for each Design Commitment shall not be construed to require that separate inspections, tests, or analyses must be performed for each Design Commitment. Instead, the activities associated with more than one ITA entry may be combined, and a single inspection, test, or analysis may be sufficient to implement more than one ITA entry.

An ITA may be performed by the licensee of the plant, or by its authorized vendors, contractors, or consultants. Furthermore, an ITA may be performed by more than a single individual or group, may be implemented through discrete activities separated by time, and may be performed at any time prior to fuel load (including before issuance of the Combined Operating License for those ITAAC that do not necessarily pertain to as-installed equipment). Additionally, ITA may be performed as part of the activities that are required to be performed under 10CFR Part 50 (including, for example, the Quality Assurance (QA) program required under Appendix B to Part 50); therefore, an ITA need not be performed as a separate or discrete activity.

Discussion of Matters Related to Operations

In some cases, the Design Descriptions in this document refer to matters that relate to operation, such as normal valve or breaker alignment during normal operation modes. Such discussions are provided solely to place the Design Description provisions in context (e.g., to explain automatic features for opening or closing valves or breakers upon off-normal conditions). Such discussions shall not be construed as requiring operators during operation to take any particular action (e.g., to maintain valves or breakers in a particular position during normal operation).

Interpretation of Figures

In many but not all cases, the Design Descriptions in Section 2 include one or more figures, which may represent a functional diagram, general structural representation, or other general illustration. For I&C systems, the figures also represent aspects of the relevant logic of the system or part of the system. Unless specified explicitly, these Figures are not indicative of the scale, location, dimensions, shape, or spatial relationships of as-built structures, systems, or components. In particular, the as-built attributes of structures, systems, and components may vary from the attributes depicted on these figures, provided that those safety functions discussed in the Design Description pertaining to the figure are not adversely affected.

Rated Reactor Core Thermal Power

The rated reactor core thermal power for the ABWR is 3926 Mw_t.