5.2 Integrity of Reactor Coolant Pressure Boundary

The information in this section of the reference ABWR DCD, including all subsections, tables, and figures, is incorporated by reference with the following departures and supplements.

STD DEP T1 2.3-1 (Table 5.2-6, Table 5.2-7, Figure 5.2-8)

STD DEP T1 2.4-2 (Table 5.2-6, Table 5.2-7)

STD DEP T1 2.14-1 (Table 5.2-6, Figure 5.2-8)

STD DEP 1.8-1 (Table 5.2-1 and Table 5.2-1a)

STD DEP 4.5-1 (Table 5.2-1 and Table 5.2-4)

STD DEP 5.2-2

STD DEP 5A-1 (Table 5.2-9)

STD DEP 7.3-11 (Figure 5.2-8)

STD DEP 7.3-12

STD DEP Vendor

STD DEP Admin

5.2.2.10 Inspection and Testing

STD DEP Vendor

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

5.2.3.2.2.3 Source of Impurities

STD DEP Admin

Condenser tubes and tube sheet~~s~~ materials *are ~~required to be made of titanium alloys~~* specified in subsection 10.4.1.2.3.

5.2.3.2.3 Compatibility of Construction Materials with Reactor Coolant

STD DEP 4.5-1

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

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

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

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

STD DEP Vendor

The design to perform preservice inspection is based on the requirements of ASME Code Section XI. The development of the preservice and inservice inspection program plans is based on ASME Code Section XI, Edition and Addenda specified in accordance with 10CFR50, Section 50.55a. For design certification, ~~GE~~Toshiba *is responsible for designing the reactor pressure vessel for accessibility to perform preservice and inservice inspection. Responsibility for designing other components for preservice and inservice inspection is the responsibility of the COL applicant. The COL applicant will be responsible for specifying the Edition of ASME Code Section XI to be used, based on the procurement date of the component per 10CFR50, Section 50.55a. The ASME Code requirements discussed in this section for preservice and inservice inspection are based on the edition of ASME Code Section XI specified in Table 1.8-21.*

5.2.4.2.2 Piping, Pumps, Valves and Supports

STD DEP 5.2-2

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

5.2.4.3.1 Examination Categories

STD DEP 5.2-2

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

5.2.4.3.2.1 Ultrasonic Examination of the Reactor Vessel

STD DEP 5A-1

~~*The GE Reactor Vessel Inspection System (GERIS) meets the detection and sizing requirements of Regulatory Guide 1.150, as cited in Table 5.2-9. Inner radius examinations are performed from the outside of the nozzle using several compound angle transducer wedges to obtain complete coverage of the required examination volume. Electronic gating used in the GERIS records up to eight different reflectors simultaneously to assure that all relevant indications are recorded. Appendix 5A demonstrates compliance with Regulatory Guide 1.150.*~~ The ultrasonic system for examination of the reactor vessel meets the qualification requirements discussed in Subsection 5.2.4.3.4.

5.2.5.1.1 Detection of Leakage Within Drywell

STD DEP T1 2.4-2

STD DEP 7.3-11

The detection of small identified leakage within the drywell is accomplished by monitoring drywell equipment drain sump pump activity and sump level increases. The equipment drain sump level monitoring instruments will activate an alarm in the control room when the ~~identified~~ total *leak rate reaches ~~95~~* 114 *liters/min.*

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

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

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

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

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

(1) High drywell temperature

~~*(2) High temperature in the valve stem seal (packing) drain lines*~~ NOT USED

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

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

(5) High drywell pressure

(6) High fission product releases

(7) Reactor vessel low water level

(8) Reactor vessel head seal drain line high pressure

(9) SRV discharge piping high temperature

(10) Feedwater lines pressure difference

5.2.5.1.2 Detection of Leakage External to Drywell

STD DEP T1 2.3-1

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

(a) ~~*High radiation in main steamlines (steam tunnel)*~~ NOT USED

5.2.5.2.1 Leak Detection Instrumentation and Monitoring Inside the Drywell

STD DEP T1 2.4-2

STD DEP 7.3-11

STD DEP 7.3-12

(1) Drywell Floor Drain Sump Monitoring

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

(2) Drywell Equipment Drain Sump Monitoring

*The drywell equipment drain sump collects only identified leakage from identified leakage sources. This sump monitors leakage from ~~valve stem packings,~~*the *RPV head flange seal~~,~~ and other known leakage sources which are piped directly into the drywell equipment drain sump.*

(10) ~~*(1) Valve Stem Packing Leakage Monitoring*~~ NOT USED

~~*Large (two inch or larger) remote power-operated valves located in the drywell for the Nuclear Boiler, Reactor Water Cleanup, Reactor Core Isolation Cooling, and Residual Heat Removal Systems are fitted with drain lines from the valve stems, from between the two sets of valve steam packing. Leakage through the inner packing is carried to the drywell equipment drain sump. Leakage during hydro-testing may be observed in drain line sight glasses installed in each drain line. Also, each drainline is equipped with temperature sensors for detecting leakage. A remote-operated solenoid valve on each line may be closed to shut off the leakage flow through the first seal in order to take advantage of the second seal, and may be used during plant operation, in conjunction with the sump instrumentation, to identify the specific process valve which is leaking.*~~

(14) Feedwater Lines Pressure Difference

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

5.2.5.2.2 Leak Detection Instrumentation and Monitoring External to Drywell

STD DEP T1 2.3-1

~~*(6) Main Steamline Radiation Monitoring*~~ NOT USED.

~~*Main steamline radiation is monitored by gamma sensitive radiation monitors of the Process Radiation Monitoring System (PRMS). The PRMS provide four divisional channel trip signals to the LDS to close all MSIVs and the MSL drain valves upon detection of high radiation in the main steamline tunnel area. A reactor trip (scram) is also initiated by the same PRMS channel trip signals. The PRMS trip signals are also used to shutdown the main condenser mechanical vacuum pump and isolate its discharge line. The detectors are geometrically arranged to detect significant increases in*~~

~~*radiation level with any number of main steamlines in operation. Control room indications and alarms are provided by the PRM System.RCIC Steamline Pressure Monitors.*~~

5.2.5.4.1 Total Leakage Rate

STD DEP 7.3-12

The total reactor coolant leakage rate consists of all leakage (identified and unidentified) that flows to the drywell floor drain and equipment drain sumps. The total leakage rate limit is well within the makeup capability of the RCIC System (182 m^3/h). The total reactor coolant leakage rate limit is established at ~~95~~ 114 *liters/min. ~~The identified and unidentified leakage rate limits are established at 95 liters/min and 3.785 liters/min, respectively.~~*

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

If either the total or unidentified leak rate limit is exceeded, an orderly shutdown shall be initiated and the reactor shall be placed in a cold shutdown condition within ~~24~~ 36 *hours.*

5.2.5.4.2 Identified Leakage Inside Drywell

STD DEP 7.3-11

*The ~~valve stem packing of large power-operated valves, the~~ reactor vessel head flange seal and other seals in systems that are part of the reactor coolant pressure boundary, and from which normal design identified source leakage is expected, are provided with leakoff drains. The ~~nuclear system valves inside the drywell and the~~ reactor vessel head flange ~~are~~*is *equipped with double seals. The leakage from ~~the inner valve stem packings and from~~ the reactor vessel head flange inner seal, which discharge*s *to the drywell equipment drain sump, ~~are~~*is *measured during plant operation.*

5.2.5.5.1 Unidentified Leakage Rate

STD DEP 7.3-12

The unidentified leakage rate is the portion of the total leakage rate received in the drywell sumps that is not identified as previously described. ~~A threat of significant compromise to the nuclear system process barrier exists if the barrier contains a crack that is large enough to propagate rapidly (critical crack length).~~ The unidentified leakage rate limit must be low because of the possibility that most of the unidentified leakage rate might be emitted from a ~~single crack~~ break *in the nuclear system process barrier.*

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

The unidentified leakage rate limit is established at ~~3.785~~ 19 *liters/min to allow time for corrective action before the process barrier could be significantly compromised. ~~This unidentified leakage rate is a small fraction of the calculated flow from a critical crack in a primary system pipe (Appendix 3E).~~*

5.2.5.5.2 Margins of Safety

STD DEP 7.3-12

The margins of safety for a detectable flaw to reach critical size are presented in Subsection 5.2.5.5.3. ~~Figure 3E-22 shows general relationships between crack length, leak rate, stress, and line size using mathematical models.~~

5.2.5.9 Regulatory Guide 1.45: Compliance

STD DEP 7.3-12

The Limiting unidentified leakage to ~~3.785~~ 19 *liters/min and identified* total *leakage to ~~95~~* 114 *liters/min satisfies Position C.9.*

5.2.6 COL License Information

5.2.6.1 Conversion of Indications

The following site-specific supplement addresses COL License Information Item 5.1.

Surveillance procedures convert the drywell leakage indications into a common leakage equivalent for unidentified and identified leakage to ensure that leakage requirements in the Technical Specifications are met.

There are four drywell leakage detection indications:

(1) Drywell floor drain sump monitoring system – The surveillance procedure measures the levels in various leakage collection tanks over prescribed time frames and converts these levels into a leakage rate.

(2) Airborne particulate channel of the drywell fission products monitoring system – The surveillance procedure converts the instantaneous detected radiation level into a leakage rate equivalent.

(3) Gaseous radioactivity channel of the drywell fission products monitoring system – The surveillance procedure converts the instantaneous detected radiation level into a leakage rate equivalent.

(4) Drywell air cooler condensate flow monitoring system – The surveillance procedure measures the flow rate in the drain line and converts this value to a leakage rate.

The surveillance procedures use the measured leakage rates from each of these monitors to determine a total unidentified leakage rate. The conversion of the information from the four leakage detection systems to a total leakage rate is accomplished by computerized programs. The drywell floor drain sump monitor, airborne particulates monitor, and drywell air cooler condensate flow monitor are capable of detecting leakage rates as low as 3.785 liters/min. The procedures include direction to the operators on actions to be taken before the TS limit is reached.

5.2.6.2 Plant-Specific ISI/PSI

The following standard supplement address COL License Information Item 5.2.

The ISI/PSI program will be based on the 2004 ASME Boiler and Pressure Vessel Code Section XI, no addenda (as identified on Table 1.8-21). This code will be used for selecting components for examinations, identifying components subject to examination, a description of the components exempted from examination by applicable code, and isometric drawings used for examination. NRC requirements for performance demonstration of ultrasonic examination of reactor pressure vessels for preservice and inservice inspections, once addressed by Regulatory Guide 1.150 will be conducted in accordance with ASME Boiler and Pressure Vessel Code Section XI, Appendix VIII as required by 10 CFR 50.55a. Ultrasonic examination systems shall be qualified in accordance with ASME Boiler and Pressure Vessel Code Section XI, Appendix VIII and ultrasonic examination shall be conducted in accordance with ASME Boiler and Pressure Vessel Code Section XI, Appendix I. ASME Boiler and Pressure Vessel Code Section XI, Appendices I and VIII address near surface examination and surface resolution including the use of electronic gating as well as internal surface examination. Code cases are listed in Table 5.2-1. Any additional relief requests shall be submitted with a supporting technical justification if needed.

The PSI/ISI program for reactor coolant pressure boundary is described in Section 5.2.4 and Table 5.2-8.This COL License Information Item is addressed by the commitment to provide a comprehensive site-specific PSI and ISI program plan to the NRC at least 12 months prior to respective unit commercial power operation as discussed in Subsection 6.6.9.1. (COM 6.6-1)

5.2.6.3 Reactor Vessel Water Level Instrumentation

The following standard supplement addresses COL License Information Item 5.3.

The Reactor Vessel Water Level Instrumentation backfill water flow is supplied from the Control Rod Drive (CRD) system to the reactor water level instrumentation leg to prevent potential formation of gas pocket (large bubbles) in the reference leg. The impact of non-condensable gases on the accuracy of reactor vessel level measurements is considered in the system design. The CRD system provides a process flow of approximately 4 L/min and is based on the results of BWR Owners Group testing in response to NRC Bulletin 93-03. This flow value is confirmed during preoperational testing in accordance with FSAR Subsection 14.2.12.16(3)(d).

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

Number	Title	Applicable Equipment	Remarks
[N-60-5	***(33)***	***Core Support***	**Accepted per RG 1.84**
~~[N-71-15~~	*~~(1)~~*	*~~Component Support]*~~*	*~~1.85~~*
*[N-71-~~17~~*18	***(1)***	***Component Support]****	**Conditionally Accepted per RG 1.84**
*[N-122-***2**	*(2)*	*Piping]**	Accepted per RG 1.84
[N-247	*(3)*	*Component Support]**	Accepted per RG 1.84
~~[N-249-9~~ **[N-249-14**	*(4)*	*Component Support]**	Conditionally Accepted per RG *~~1.85~~* **1.84**
[N-309-1	*(5)*	*Component Support]**	Accepted per RG 1.84
[N-313	*(6)*	*Piping]**	Accepted per RG 1.84
[N-316	*(7)*	*Piping]**	Accepted per RG 1.84
~~[N-318-3~~ ***[N-318-5***	*(8)*	*Piping]**	*~~Conditionally~~* Accepted per RG 1.84
[N-319-3	**(9)**	**Piping]***	**Accepted per RG 1.84**
*[N-391***-2**	*(10)*	*Piping]**	Accepted per RG 1.84
*[N-392***-3**	*(11)*	*Piping]**	Accepted per RG 1.84
[N-393	*(12)*	*Piping]**	Accepted per RG 1.84
[N-411-1	*(13)*	*Piping]**	Conditionally Accepted per RG 1.84
[N-414	*(14)*	*Component Support]**	Accepted per RG 1.84
[N-430	*(15)*	*Component Support]**	Accepted per RG 1.84
N-236-1	(16)	Containment	Conditionally Accepted Per RG 1.147
~~N-307-1~~ **N-307-2**	(17)	RPV Studs	Accepted per RG 1.147
N-416**-3**	(20)	Piping	Accepted Per RG 1.147
N-432	(21)	Class 1 Components	Accepted Per RG 1.147
N-435-1	(22)	Class 2 Vessels	Accepted Per RG 1.147
N-457	(23)	Bolt and Studs	Accepted Per RG 1.147
N-463**-1**	(24)	Piping	Accepted Per RG 1.147
N-460	(25)	Class 1 & 2 Components and Piping	Accepted Per RG 1.147
N-472	(26)	Pumps	Accepted Per RG 1.147

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

Number	Title	Applicable Equipment	Remarks
[N-476	*(26a)*	*Component Support]**	**Accepted per RG 1.84**
N-479**-1**	(27)	Main Steam System	~~Not Listed in~~ **Accepted per** RG 1.147
N-491	(28)	Component Supports	~~Not Listed in~~ **Accepted per** RG 1.147
N-496	(29)	Bolts and Studs	~~Not Listed in~~ **Accepted per** RG 1.147
N-580-2	**(30)**	**RPV, Reactor Internals, etc**	**Approved by ASME Standards Committee (2008)**
N-608	**(31)**	**Use of Applicable Code Edition and Addenda, NCA-1140(a)(2)**	**Accepted per RG 1.84**
N-613-1	**(32)**	**Reactor Vessel**	**Accepted per RG 1.147**
N-632	**(34)**	**Containment**	**Accepted per RG 1.84**

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

(30)	Use of Alloy 600 (UNS N066000) with Columbium added, Section III, Div. 1 (SC III File #N96-44) (MC97-86)
(31)	Applicable Code Edition and Addenda, NCA-1140(a)(2), Section III, Division 1
(32)	UT Exam of Penetration Nozzles in Vessels, Category B-D, Item Nos. B3.10 and B3.90, Reactor Nozzle to Vessel Welds, Figs. IWB 2500-7(a), (b), (c), Section XI, Division 1
(33)	Material for Core Support Structures, Section III, Division 1
(34)	Use of ASTM A 572, Grades 50 and 65 for Structural Attachments to Class CC Containment Liners, Section III, Division 2.

Table 5.2-4 Reactor Coolant Pressure Boundary Materials

Component	Form	Material	Specification (ASTM/ASME)
Main Steam Isolation Valves			
Valve Body	Cast	Carbon steel	SA352 LCB
Cover	Forged	Carbon Steel	SA350LF2
Poppet	Forged	Carbon Steel	SA350LF2
Valve stem	Rod	~~*17-4 ph*~~ **Precipitation Hardened Stainless Steel**	SA 564 630 (H1100)
Body bolt	Bolting	**Low-**Alloy steel	SA 540 B23 CL4 or 5
Hex nuts	Bolting Nuts	**Low-**Alloy steel	SA 194 GR7
Main Steam Safety/Relief Valve			
Body	Forging or Casting	Carbon steel Carbon steel	ASME SA 350 LF2 ASME SA 352 LCB
Bonnet (yoke)	Forging or Casting	Carbon steel Carbon steel	ASME SA 350 LF2 ASME SA 352 LCB
Nozzle (seat)	Forging or Casting	Stainless steel or Carbon steel	ASME SA 182 Gr F316 or SA351 CF3 or CF 3M ASME SA 350 LF2 or SA 352 LCB
Body to bonnet stud	~~*Bar/rod*~~ **Bolting**	Low-Alloy steel	ASME SA 193 Gr B7
Body to bonnet nut	~~*Bar/rod*~~ **Bolting Nuts**	**Low-**Alloy steel	ASME SA 194 Gr 7
Disk	Forging or Casting	~~*Alloy steel NiCrfe*~~ **NiCrFe** Alloy Stainless steel	ASME ~~*SA*~~**SB** 637 Gr 718 ASME SA 351 CF 3A
Spring washer &	Forging	Carbon steel	ASME SA 105
Adjusting Screw or	**Bolting**	Alloy steel	ASME SA 193 Gr B6 (Quenched + tempered or normalized & tempered)
Setpoint adjustment assembly	Forgings	Carbon and alloy steel parts	Multiple specifications

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

Component	Form	Material	Specification (ASTM/ASME)
Spindle (stem)	Bar	Precipitation-hardened **stainless** steel	ASTM A564 Type 630 (H 1100)
Spring	Wire or Bellville washers	Steel Alloy Steel	ASTM A304 Gr 4161 N 45 Cr Mo V67
Main Steam Piping (between RPV and the turbine stop valve)			
Pipe	Seamless	Carbon steel	ASME SA 333 Gr. 6
Contour nozzle **250A** 10.36 MpaG	Forging	*Carbon* steel	ASME SA *350 LF 2*
Large groove flange	Forging	Carbon steel	ASME SA 350 LF 2
50A special nozzle	Forging	Carbon steel	ASME SA 350 LF2
Elbow	Seamless	Carbon steel	ASME SA 420
Head fitting/penetration piping	Forging	Carbon steel	ASME SA 350 LF2
Feedwater Piping (between RPV and the seismic interface restraint)			
Pipe	Seamless	Carbon steel	ASME SA 333 Gr. 6
Elbow	Seamless	Carbon steel	ASME SA 420
Head fitting/penetration piping	Forging	Carbon steel	ASME SA 350 LF2
Nozzle	Forging	Carbon steel	ASME SA 350 LF2
Recirculation Pump Motor Cover			
Bottom flange (cover)	Forging	**Low-**Alloy steel	ASME SA 533 Gr. B Class 1 or SA 508 Class 3
Stud	Bolting	**Low-**Alloy steel	ASME SA 540 CL.3 Gr.B24 or SA 193, B7
Nut	Bolting **Nuts**	**Low-**Alloy steel	ASME SA 194 Gr. 7
CRD			
Middle flange	Forging	Stainless steel	SA 182/**182M**, F304L***, F304*,F**316L* **or F316*, or SA 336/336M, F304* or F316***
Spool piece	Forging	Stainless steel	SA 182/**182M**, F304L* **, F304*F**316L***, or SA 336/336M, F304* or F316***

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

Component	Form	Material	Specification (ASTM/ASME)
Mounting bolts	~~Bar~~ **Bolting**	**Low-**Alloy steel	~~SA 194~~ **SA-193/193M Grade** B7
Seal housing	Forging	Stainless steel	SA 182**/182M**, F304L* **, F304*F**316L* **or F316***, **or SA 336/336M, F304* or F316***
Seal housing nut	Bar	Stainless steel	SA 564, ~~17-4PH~~ **630**(H1100)
Reactor Pressure Vessel			
Shells and Heads	Plate	**Low-Alloy steel** ~~Mn-1/2 Mo-1/2 Ni~~	SA-533, Type B, Class 1
	Forging	~~¾ Ni-1/2 Mo-Cr-V~~ Low alloy steel	SA-508, Class 3
Shell and Head Flange	Forging	~~¾ Ni-1/2 Mo-Cr-V~~ Low alloy steel	SA-508 Class 3
Flanged Nozzles	Forging	~~C-Si~~ Low alloy steel	SA-508 Class 3
Drain Nozzles	Forging	~~C-Si~~ Carbon steel **or Stainless steel**	SA-508 Class 1 **or SA 182, F316L* or F316* SA-336, F316***
Appurtenances/Instrumentation Nozzles	Forging	~~Cr-Ni-Mo~~ Stainless steel	SA-182, Grade F316L* or ~~F316†~~ **F316*** or SA-336, ~~Class F316L* or F316‡~~ **Class F316***
	Bar, Smls. Pipe	Ni-Cr-Fe (UNS N06600)	~~SB-166‡ or SB-167‡~~ **Code Case N-580-2**
Stub Tubes	Forging	Ni-Cr-Fe (UNS N06600)	~~SB-564‡~~ **Code Case N-580-2**
	Bar, Smls. Pipe	Ni-Cr-Fe (UNS N06600)	~~SB-166‡ or SB-167‡~~ **Code Case N-580-2**

* Carbon content is maximum 0.020%.
~~† Carbon content is maximum 0.020% and nitrogen from 0.060 to 0.120%.~~
~~‡ Added niobium content is 1 to 4%.~~

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

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

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

Table 5.2-7 Leakage Sources vs. Monitored Trip Alarms

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

Table 5.2-7 Leakage Sources vs. Monitored Trip Alarms

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

I = Inside Drywell Leakage
O = Outside Drywell Leakage
Ⓧ = Reactor coolant leakage in cooling water to RHR Hx, RIP Hx, CUW Non-regen Hx's or to FP cooling Hx.

Table 5.2-9 ~~Ultrasonic Examination of RPV: Reg. Guide 1.150 Compliance~~

NOT USED

The following figure is located in Chapter 21:

Figure 5.2-8, Leak Detection and Isolation System IED (Sh. 1-10)

This figure is revised due to departure STD DEP 2.3-1, STD DEP T1 2.14-1, and STD DEP 7.3-11