3.9 Mechanical Systems and Components

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 Admin	(Table 3.9-1)
STD DEP T1 2.4-3	(Table 3.9-8, MPL# E51)
STD DEP T1 2.14-1	(Table 3.9-8, MPL # T49 and P21)
STD DEP 3.9-1	
STD DEP T1 2.4-1	(Table 3.9-8, MPL# E11)
STP DEP 9.2-5	(Table 3.9-8, MPL# P41)
STD DEP 9.3-2	(Table 3.9-8, MPL# ~~P56~~ P81)

~~STD DEP T1 2.4-3~~

~~The Reactor Core Isolation Cooling System (RCIC) alternate design description was provided in ABWR Licensing Topical Report NEDE-33299, "Advanced Boiling Water Reactor (ABWR) with Alternate RCIC Turbine-Pump," dated December 2006. The markup information on pages C-11 through C-20 of the Licensing Topical Report is incorporated by reference.~~

~~STD DEP T1 2.14-1~~

~~The Hydrogen Recombiner Requirements Elimination description was provided in ABWR Licensing Topical Report NEDO-33330, "Advanced Boiling Water Reactor (ABWR) Hydrogen Recombiner Requirements Elimination," Revision 1, dated September 2007. The markup information on page C-17, C-18, C-19, and C-175 of the Licensing Topical Report is incorporated by reference.~~

3.9.2.2.2.7 RCIC Pump and Turbine Assembly

STD DEP T1 2.4-3

~~*The RCIC pump construction is a horizontal, multistage type and is supported on a pedestal. The RCIC pump assembly is qualified analytically by static analysis for seismic and other RBV loadings, as well as the design operating loads of pressure, temperature, and external piping loads. The results of this analysis confirm that the stresses are less than the allowable (Subsection 3.9.3.2.2).*~~

The RCIC turbine-pump *is qualified for seismic and other RBV loads via a combination of static analysis and dynamic testing (Subsection 3.9.3.2.1.5* and 3.9.3.2.2*). The turbine*-pump *assembly consists of rigid masses (wherein static analysis is utilized)* *interconnected with control levers and electronic control systems, necessitating final qualification via dynamic testing. Static loading analyses are employed to verify the structural integrity of the turbine-pump assembly and the adequacy of bolting under operating, seismic, and other RBV loading conditions. The complete turbine-pump assembly is qualified via dynamic testing in accordance with IEEE-344. The qualification test program includes a demonstration of startup capability, as well as operability during dynamic loading conditions.* ~~*Operability under normal load conditions is assured by comparison to the operability of similar turbines in operating plants.*~~

3.9.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The following standard supplement addresses Regulatory Guide (R.G.) 1.206, Rev. 0:

~~The analytical methods and procedures to predict vibration of ABWR pressure vessel internals (including the steam dryer and other main steam components) as discussed in Section 3.9.2.3 of R.G. 1.206 are addressed in ABWR Licensing Tropical Report NEDO-33316, "Advance Boiling Water Reactor (ABWR) Vibration Assessment Program in compliance with The United States Nuclear Regulatory Commission Regulatory Guide 1.20," dated April 2007.~~

The reactor internals design of the first ABWR plant, which has been in operation since 1995, is considered to be the 1350 MWe ABWR "Valid Prototype" defined in Regulatory Guide 1.20 because:

(1) The prototype ABWR plant reactor internals have successfully completed a comprehensive vibration assessment program during the pre-operational and initial startup testing. This vibration assessment program consisted of a vibration and fatigue analysis, a vibration measurement program, an inspection program, and a correlation of their results.

(2) The reactor internals of the prototype ABWR plant have experienced no adverse in-service vibration phenomena.

Also, Regulatory Guide 1.20 Section C-1.4 defines non-prototype, Category I as "a reactor internals configuration with substantially the same arrangement, design, size, and operating conditions as a specified Valid Prototype and for which nominal differences in arrangement, design, size, and operating conditions have been shown by test or analysis to have no significant effect on the vibratory response and excitation of those reactor internals important to safety." STP 3 and 4 reactor internals are substantially the same as those of the valid prototype. Also, the valid prototype has no significant effect on the vibratory response and excitation of those reactor internals important to safety. Therefore, STP 3 and 4 reactor internals are classified as "Non-Prototype, Category I" of the 1350 MWe ABWR.

From the guidance of Regulatory Guide 1.206 Section C.I.3.9.2.3 for non-prototype, a brief summary of the valid prototype test and analysis results are shown as follows.

Following the guidance of Regulatory Guide 1.20, Section C-2.1, the vibration analysis program was performed for those steady-state and anticipated transient conditions that correspond to preoperational and initial startup test and normal operating conditions. The dynamic analytical finite-element models were developed to predict the natural vibration frequency, modal displacement, and modal strain and stress for the following components.

(1) Control Rod Guide Tubes and Control Rod Drive Housings

(2) In-core Guide Tubes and Housings

(3) High Pressure Core Flooder Sparger and Coupling

(4) Core Shroud

(5) Steam Dryer Skirt, Drain Channel and Hood

From the analyses results, it was verified that the maximum vibration stress amplitudes were all below the allowable limit for all normal steady-state and transient operating conditions (including the combination of several pumps stopping).

Following the guidance of Regulatory Guide 1.20, Section C-2.2, a vibration measurement program was developed and implemented to verify the structural integrity of the reactor internals, to determine the margin of safety associated with steady-state and anticipated transient conditions for normal operation, and to confirm the results of the vibration analysis. Strain gages, accelerometers, and/or linear variable differential transformers were utilized to measure vibration related data on the following reactor internal components:

(1) Steam Dryer Skirt, Drain Channel, Support Ring, Hood and Vessel Dome Region Pressure

(2) High Pressure Core Flooder Sparger, Coupling and Thermal Ring

(3) Control Rod Guide Tube and Control Rod Drive Housing

(4) In-core Monitor Guide Tube and Housing

(5) Core Shroud

(6) Top Guide

For the selection of instrument components, the following criteria were considered:

(1) History of flow-induced vibration problems

(2) New design or new flow condition

(3) Difficulty to repair or replace

The measurements results showed that the maximum vibration stress amplitudes were all below the allowable limit for all normal steady-state and transient operating conditions (including the combination of several pumps stopping).

From the guidance of Regulatory Guide 1.20, Section C-2.3, the inspection program was implemented prior to and following operation at those steady-state and transient modes consistent with the test conditions for Regulatory Guide 1.20 Section C-2.2.2. The reactor internals were removed from the reactor vessel for these inspections. For components in which removal was not feasible, the inspections were performed by means of examination equipment appropriate for in situ inspection.

The proposed design for STP 3 and 4 is substantially the same as the valid prototype reactor internal components. In addition, changes to the reactor internal components are not contemplated at this time. If any changes are determined necessary in the future, they will be addressed at the time the change is proposed with proper evaluation/ justification.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

The following standard supplement addresses Regulatory Guide (R.G.) 1.206, Rev. 0:

~~Analysis of potential adverse flow effects (e.g., flow-induced vibrations and acoustic resonances) that can impact ABWR reactor pressure vessel internals (including the steam dryer and other Main Steam System components) as discussed in Sections 3.9.2.4 of R.G. 1.206 are addressed in ABWR Licensing Topical Report NEDO-33316, "Advance Boiling Water Reactor (ABWR) Vibration Assessment Program in compliance with The United States Nuclear Regulatory Commission Regulatory Guide 1.20," dated April 2007.~~

As discussed in Subsection 3.9.2.3, STP 3 and 4 reactor internals are classified as non-prototype, Category I. In accordance with the requirement of Regulatory Guide 1.206 Section C.I.3.9.2.4 for non-prototype, a brief summary of test and analysis results are shown in Section 3.9.2.3.

For STP 3 and 4 reactor internals components, an inspection program will be implemented in lieu of a vibration measurement program as discussed in paragraph C.3.1.3 of Regulatory Guide 1.20.

The inspection of the reactor internals shall be implemented prior to and following operation at those steady-state and transient modes including the unbalanced pump operating condition. The test operating duration shall be determined to have the reactor internal components accumulate at least 10^6 cycles of vibration prior to the final inspection. Also, the duration of testing shall be no less than that for the valid prototype reactor internals. This test operating duration is adequate, because the operating ABWRs have not experienced problems caused by flow-excited acoustic resonances and flow-induced vibrations, and the valid prototype measurement results show no significant responses. These flow test and inspections shall be implemented prior to fuel loading.

Testing shall be performed with the reactor internals important to safety and the fuel assemblies (or dummy assemblies that provide equivalent dynamic mass and flow characteristics) in position. The testing may be conducted without real or dummy fuel assemblies if it can be shown (by analytical or experimental means) that such conditions will yield conservative results. For the reactor internals for which removal is not feasible, the inspections shall be performed by means of examination equipment appropriate for in situ inspection.

These inspections, when completed prior to fuel load, will permit closure of that portion of Tier 1 Table 2.1.1d ITAAC #7 for as-built vessel internals. It shall be verified that the as-built vessel internals have no damage or loose parts affected by flow-induced vibration. Details of the inspection requirements shall be provided in a specification, which shall be submitted prior to pre-operational test.

Although the steam dryer design of STP 3 and 4 is identical to the valid prototype design, the configuration of the main steamline may have an influence on the steam dryer loads. Therefore, additional test and analysis requirements are voluntarily adopted in accordance with Regulatory Guide 1.20, Rev. 3. It is noted that Regulatory Guide 1.20, Rev. 2 is applicable to the ABWR, per Table 1.8-20.

Pursuant to the guidance of Regulatory Guide 1.20, Rev.3, analyses and scale model tests will be performed to address the effects of any differences in the main steam line configuration between STP 3 and 4 and the Prototype plant, specifically with respect to the steam dryer loads.

Also, as discussed in Regulatory Guide 1.20, Rev. 3, the main steam lines in STP 3 and 4 will be instrumented with strain gages to provide measurements of pressure fluctuations due to flow-induced vibrations. The measurements will be used by the Acoustic Circuit Methodology to analytically predict the steam dryer flow-induced vibration loads. The predicted loads will then be used with a finite-element model of the dryer to confirm the acceptability of the flow-induced vibration loads.

After the first operating cycle of STP 3 and 4, detailed inspections of the steam dryer will be performed to confirm the structural adequacy of the dryer for flow-induced vibration loads.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

The following standard supplement addresses Regulatory Guide (R.G.) 1.206, Rev. 0:

~~The details of the test program to correlate the test measurements with the analytically predicted flow-induced dynamic response of the ABWR reactor internals (including steam dryers and other Main Steam System components) as discussed in Section 3.9.2.6 of R.G. 1.206 are addressed in ABWR Licensing Topical Report NEDO-33316, "Advance Boiling Water Reactor (ABWR) Vibration Assessment Program in compliance with The United States Nuclear Regulatory Commission Regulatory Guide 1.20," dated April 2007.~~

As discussed in Section 3.9.2.3, correlation between reactor internals vibration tests and analytical results was performed for the valid prototype. The test results ~~are~~were compared to the analytical results, and both results showed that the maximum vibration stress amplitudes were all below the allowable limit for all normal steady-state and transient operating conditions (including the combination of several pumps stopping).

Analytical models used for these analyses were verified by comparing the calculated natural frequencies of each component with those values measured by the hammering test or alternate calculation.

3.9.3.1.8 RCIC Turbine-Pump

STD DEP T1 2.4-3

*~~Although not under the jurisdiction of the ASME Code, the RCIC turbine is designed and evaluated and fabricated following the basic guidelines of~~*The RCIC turbine-pump is constructed in accordance with the requirements of *ASME Code Section III for Class 2 components.*

3.9.3.1.9 ECCS Pumps

STD DEP T1 2.4-3

The RHR~~, RCIC,~~ and HPCF pumps are constructed in accordance with the requirements of an ASME Code Section III, Class 2 component.

3.9.3.2.1.5 RCIC Turbine-Pump

STD DEP T1 2.4-3

The RCIC turbine-pump *is qualified by a combination of static analysis and dynamic testing as described in Subsection 3.9.2.2.2.7. The turbine*-pump *assembly consists of rigid masses (wherein static analysis is utilized) interconnected with control levers and electronic control systems, necessitating final qualification by dynamic testing. Static loading analysis has been employed to verify the structural integrity of the turbine-*pump *assembly, and the adequacy of bolting under operating and dynamic conditions. The complete turbine*-pump *assembly is qualified via dynamic testing, in accordance with IEEE-344. The qualification test program includes demonstration of startup capability, as well as operability during dynamic loading conditions. Operability under normal load conditions is assured by comparison to operability of similar turbines in operating plants.*

3.9.3.2.2 SLC Pump and Motor Assembly and RCIC Turbine-Pump Assembly

3.9.3.4.2 Reactor Pressure Vessel Support Skirt

STD DEP Admin

Replace the following equation (3.9-1)

$$\left(\frac{P}{P_{crit}}\right) + \left(\frac{q}{q_{crit}}\right) + \left(\frac{\tau}{\tau_{crit}}\right) < \left(\frac{1}{S.F.}\right)$$

with

$$\left(\frac{P}{P_{crit}}\right) + \left(\frac{q}{q_{crit}}\right) + \left(\frac{\tau}{\tau_{crit}}\right)^2 < \left(\frac{1}{S.F.}\right)$$

3.9.3.4.4 Floor-Mounted Major Equipment (Pumps, Heat Exchangers, and RCIC Turbine-Pump)

STD DEP T1 2.4-3

Since the major active valves are supported by piping and not tied to building structures, valve "supports" do not exist (Subsection 3.9.3.4.1).

The HPCF, RHR, ~~RCIC,~~ SLC, FPCCU, SPCU, and CUW pumps; RCW, RHR, CUW, and FPCCU heat exchangers; and RCIC turbine-pump *are all analyzed to verify the adequacy of their support structure under various plant operating conditions. In all cases, the load stresses in the critical support areas are within ASME Code allowables.*

Seismic Category I active pump supports are qualified for dynamic (seismic and other RBV) loads by testing when the pump supports together with the pump meet the following test conditions:

(1) Simulate actual mounting conditions.

3.9.5.1.2.9 Incore Guide Tubes and Stabilizers

STD DEP 3.9-1

These are Safety Class 3 components. The guide tubes protect the incore instrumentation from flow of water in the bottom head plenum and provide a means of positioning fixed detectors in the core, as well as a path for insertion and withdrawl of the calibration monitors (ATIP, Automated Traversing Incore Probe Subsystem). The incore flux monitor guide tubes extend from the top of the incore flux monitor housing to the top of the core plate. (The power range detectors for the power range monitoring units and the dry tubes for the startup range neutron monitoring and average power range monitoring (SRNM) detectors are inserted through the guide tubes). The local power range monitor (PRNM) detector assemblies and the dry tubes for the startup range monitoring (SRNM) assemblies are inserted through the guide tube.

Two levels of ~~stainless steel~~stabilizer latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The stabilizers are connected to the shroud and shroud support. The bolts are tack-welded after assembly to prevent loosening during reactor operation.

3.9.6 Testing of Pumps and Valves

STD DEP Admin

The following change is made in the 3rd sentence of the 2nd paragraph of this subsection.

For example, the periodic leak testing of the reactor coolant pressure isolation valves (See Appendix 3M for design changes made to prevent intersystem LOCAs) in Table 3.9-9 will be performed in accordance with Chapter 16 Surveillance Requirement SR ~~3.6.1.5.10~~ 3.4.4.1.

3.9.7 COL License Information

3.9.7.1 Reactor Internals Vibration Analysis, Measurement and Inspection Program

The following standard supplement addresses COL License Information Item 3.27.

~~LTR NEDO-33316 addresses the ABWR Vibration Assessment Program in compliance with NRC Regulatory Guide 1.20. The plant specific information will provide assessment results in accordance with the applicable portion of position C.3 of Regulatory Guide 1.20 for non-prototype internals. The plant specific information will be available for review following preoperational and initial start up testing. (COM 3.9-1)~~

The results of the vibration assessment program for the valid prototype ABWR internals are shown in Section 3.9.2.3. In addition, a vibration assessment program summarized in Section 3.9.2.4 will be implemented for non-prototype, ~~c~~Category I ABWR.

3.9.7.2 ASME Class 2 or 3 or Quality Group D Components with 60-Year Design Life

The following standard supplement addresses COL License Information Item 3.28.

The ASME Class 2 or 3 or Quality Group D components that are subjected to cyclic loadings, including operating vibration loads and thermal transients effects, of a magnitude and/or duration so severe the 60-year design life cannot be assured by required Code calculations and, if similar designs have not already been evaluated, will be identified and an appropriate analysis will be available to demonstrate the required design life or designs to mitigate the magnitude or duration of the cyclic loads will be available for review prior to fuel load. (COM 3.9-2)

3.9.7.3 Pump and Valve Testing Program

The following standard supplement addresses COL License Information Item 3.29.

The plant specific environmental parameters for the equipment qualification program will be available for NRC review as part of the ITAAC for basic configuration of systems, as provided in the reference ABWR DCD Tier 1 Section 1.2.

The pump and valve inservice testing and inspection program will be provided to the NRC as specified in section 13.4S. This program will include the following:

(1) Include baseline pre-service testing to support the periodic inservice testing of the components required by technical specifications. Provisions are included to disassemble and inspect the pump, check valves, POVs, and MOVs within the Code and safety-related classification as necessary, depending on test results.

(2) Provide a study to determine the optimal frequency of the periodic verification of the continuing MOV capability for design basis conditions.

The design qualification test, inspection and analysis criteria in Subsections 3.9.6.1, 3.9.6.2.1, 3.9.6.2.2 and 3.9.6.2.3 of Tier 2 of the reference ABWR DCD will be included in the respective safety-related pump and valve design specifications prior to fuel load. (COM 3.9-3)

The design, qualification, and preoperational testing for MOVs as discussed will conform to the provisions in Subsection 3.9.6.2.2 of Tier 2 of the reference ABWR DCD. (COM 3.9-4)

SRV IST requirements are included in Table 3.9-8 (B21 Nuclear Boiler System Valves) and additional SRV testing including technical specification testing is described in Section 5.2.2.10.

As is described for ISI in COL License Information item 6.6.9.1, inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) of this section on the date 12 months before the date of issuance of the operating license (or the optional ASME Code cases listed in NRC Regulatory Guide 1.192 that is incorporated by reference in 10 CFR 50.55a(b) of this section), subject to the limitations and modifications listed in 10 CFR 50.55a(b) of this section.

As is described for ISI in COL License information item 6.6.9.1, inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during successive 120-month intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) of this section 12 months before the start of the 120- month interval (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 14, or 1.192 that are incorporated by reference in 10 CFR 50.55a(b) of this section), subject to the limitations and modifications listed in 10 CFR 50.55a(b) of this section.

3.9.7.4 Audit of Design Specification and Design Reports

The following site-specific supplement addresses COL License Information Item 3.30.

The design specification and design reports required by ASME Code for vessels, pumps, valves and piping systems for the purpose of audit will be made available for NRC review.

The piping system design is consistent with the construction practices, including inspection and examination methods, of the ASME Code 1989 edition with no addenda.

ASME Code editions and addenda other than those listed in Tables 1.8-21 and 3.2-3, will not be used to design ASME Code Class 1, 2 and 3 pressure retaining components and supports.

Table 3.9-1 Plant Events

B. Dynamic Loading Events[5]		
	ASME Code Service Limit[1]	No. of Cycles/Events[2]
13. Safe Shutdown Earthquake (SSE) at Rated Power Operating Conditions	D[8]	1 ~~Cycle~~ **Event**[4]

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves

MPL	System	Pump Page No.	Valve Page No.
~~P56~~ P81	Breathing Air System		3.9-132
~~T49~~	~~Flammability Control~~		~~3.9-137~~

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig. (g)
E11 Residual Heat Removal System Valves								
F014	~~2~~3	Fuel Pool Cooling supply line inboard MOV	2	B	A	P S	2yr, 3 mo	5.4-10 sh.**3**,5,7
F015	~~2~~**3**	Fuel Pool Cooling supply line outboard MOV	2	B	A	P S	2 yr,3 mo	5.4-10 sh. **3**,5,7
F016	~~2~~**3**	Gate valve-line from Fuel Pool Cooling (FPC)	2	B	A	S	3 mo	5.4-10 sh. 2
E51 Reactor Core Isolation Cooling System Valves								
~~F012~~	~~1~~	~~RCIC turbine accessories cooling water line MOV~~	~~2~~	~~B~~	~~A~~	~~P S~~	~~2 yr 3 mo~~	~~5.4-8 sh. 3~~
~~F013~~	~~1~~	~~RCIC turbine accessories cooling water line PCV~~	~~2~~	~~B~~	~~A~~		~~E1~~	~~5.4-8 sh. 3~~
~~F015~~	~~1~~	~~Barometric condenser condensate pump discharge line valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 3~~
~~F016~~	~~1~~	~~Barometric condenser condensate pump discharge line check valve~~	~~2~~	~~C~~	~~P~~	~~P S~~	~~2 yr 3 mo~~	~~5.4-8 sh. 3~~
~~F030~~	~~1~~	~~Turbine accessories cooling water line relief valve~~	~~2~~	~~C~~	~~A~~	~~R~~	~~5 yr~~	~~5.4-8 sh. 3~~
~~F031~~	~~1~~	~~Barometric condenser condensate discharge line AOV to HCW~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 3~~
~~F032~~	~~1~~	~~Barometric condenser condensate discharge line AOV to HCW~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 3~~
~~F034~~	~~1~~	~~Barometric condenser condensate pump discharge line test line valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 3~~
~~F044~~	~~1~~	~~Steam admission valve bypass line maintenance valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 2~~
~~F045~~	~~1~~	~~Steam admission valve bypass line MOV~~	~~2~~	~~B~~	~~A~~	~~P S~~	~~2 yr 3 mo~~	~~5.4-8 sh. 2~~
~~F046~~	~~1~~	~~Barometric condenser vacuum pump discharge line check valve (h3)~~	~~2~~	~~A, C~~	~~I, A~~	~~L,S~~	~~RO~~	~~5.4-8 sh. 1~~

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

~~F047~~	~~1~~	~~Barometric condenser vacuum pump discharge line MOV~~	~~2~~	~~A~~	~~I,A~~	~~L,P S~~	~~RO 3 mo~~	~~5.4-8 sh. 1~~
~~F051~~	~~1~~	~~Turbine exhaust line drain line valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 3~~
~~F052~~	~~1~~	~~Turbine exhaust line drain line valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 3~~
~~F059~~	~~1~~	~~Barometric condenser vacuum pump discharge line test line valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 1~~
~~F712~~	~~1~~	~~Turbine accessories cooling water line instrument root valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 3~~
~~F713~~	~~1~~	~~Turbine accessories cooling water line instrument root valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 3~~
~~F714~~	~~1~~	~~Turbine accessories cooling water line instrument root valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~5.4-8 sh. 3~~
P21 Reactor Building Cooling Water System Valves								
~~F037~~	~~2~~	~~Cooling water supply line to FCS room air conditioner~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~9.2-1 sh. 2,5~~
~~F038~~	~~2~~	~~Cooling water return line from FCS room air conditioner~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~9.2-1 sh. 2,5~~
~~**T49 Flammability Control System Valves**~~								
~~F001~~	~~2~~	~~Inlet line from drywell inboard isolation valve~~	~~2~~	~~A~~	~~I,A~~	~~L, P S~~	~~2 yr 3 mo~~	~~6.2-40~~
~~F002~~	~~2~~	~~Inlet line from drywell outboard isolation valve~~	~~2~~	~~A~~	~~I,A~~	~~L, P S~~	~~2 yr 3 mo~~	~~6.2-40~~
~~F003~~	~~2~~	~~Flow control valve for the FCS inlet line from drywell~~	~~3~~	~~B~~	~~A~~	~~P S~~	~~2 yr 3 mo~~	~~6.2-40~~
~~F004~~	~~2~~	~~Blower bypass line flow control valve~~	~~3~~	~~B~~	~~A~~	~~P S~~	~~2 yr 3 mo~~	~~6.2-40~~
~~F005~~	~~2~~	~~Blower discharge line to wetwell check valve (h9)~~	~~3~~	~~C~~	~~A~~	~~S~~	~~RO~~	~~6.2-40~~
~~F006~~	~~2~~	~~Discharge line to wetwell outboard isolation valve~~	~~2~~	~~A~~	~~I,A~~	~~L, P S~~	~~2 yr 3 mo~~	~~6.2-40~~
~~F007~~	~~2~~	~~Discharge line to wetwell inboard isolation valve~~	~~2~~	~~A~~	~~I,A~~	~~L, P S~~	~~2 yr 3 mo~~	~~6.2-40~~

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

~~F008~~	~~2~~	~~Cooling water supply line from the RHR System MOV~~	~~3~~	~~B~~	~~A~~	~~P S~~	~~2 yr 3 mo~~	~~6.2-40~~
~~F009~~	~~2~~	~~Cooling water supply line maintenance valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F010~~	~~2~~	~~Cooling water supply line admission MOV~~	~~3~~	~~B~~	~~A~~	~~P S~~	~~2 yr 3 mo~~	~~6.2-40~~
~~F013~~	~~2~~	~~Inlet line from drywell drain line valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F015~~	~~1~~	~~Blower discharge line to wetwell pressure relief valve~~	~~2~~	~~A,C~~	~~I,A~~	~~R L~~	~~5 yr RO~~	~~6.2-40~~
~~F016~~	~~2~~	~~Blower discharge line to wetwell pressure relief line check valve (h3)~~	~~2~~	~~A,C~~	~~I,A~~	~~L, S~~	~~RO~~	~~6.2-40~~
~~F501~~	~~2~~	~~Inlet line from drywell test line valve~~	~~2~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F502~~	~~2~~	~~Discharge line to wetwell test line valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F504~~	~~2~~	~~Blower suction line test line valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F505~~	~~2~~	~~Blower discharge line test line valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F506~~	~~2~~	~~Drain line to low conductivity waste (LCW) valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F507~~	~~2~~	~~Cooling water supply line test line valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F701~~	~~2~~	~~FE-T49-FE002 upstream instrument line root valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F702~~	~~2~~	~~FE-T49-FE002 downstream instrument line root valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F703~~	~~2~~	~~Blower suction line pressure instrument line root valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F704~~	~~2~~	~~FE-T49-FE004 upstream instrument line root valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
~~F705~~	~~2~~	~~FE-T49-FE004 downstream instrument line root valve~~	~~3~~	~~B~~	~~P~~		~~E1~~	~~6.2-40~~
P41 Reactor Service Water System Valves								
~~F601~~	~~6~~	~~RSW Cooling tower drain valves~~	~~3~~	~~C~~	~~P~~		~~E1~~	~~9.2-7 sh. 1,2,3~~
F110	6	RSW return to cooling tower ~~A & D MOV~~	3	~~C~~B	A	P S	2 yr 3mo	9.2-7 sh. 1,2,3

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves (Continued)

F109	3	RSW cold bypass to cooling tower basin MOV	3	~~C~~B	A	P S	2 yr 3mo	9.2-7 sh. 1,2,3
F115	1	Makeup water to UHS basin MOV	3	~~C~~B	A	P S	2 yr 3mo	9.2-7 sh. 1~~,2,3~~
F113/ F116	2	Makeup water to UHS basin Manual Isolation valves	3	~~C~~B	~~I,~~P	~~P~~	~~2 yr~~E1	9.2-7 sh. 1~~,2,3~~
F114/ F117	2	Makeup water to UHS basin Check valves	3	C	A	~~P~~ S	~~2 yr~~ 3mo	9.2-7 sh. 1~~,2,3~~
F101	3	RSW line to HVAC Air Conditioning Condenser Manual Isolation valves	3	~~C~~B	~~I,~~P		E1	9.2-7 sh. 1,2,3
F102	3	RSW blowdown line to Main Cooling Reservoir MOV	3	~~C~~B	A	P S	2 yr 3mo	9.2-7 sh. 1,2,3
P51 Service Air System Valves								
F132	*1*	*Inboard isolation ~~manual~~check valve (h1)*	*2*	*A, C*	*I, ~~P~~A*	*L, S*	*RO*	*9.3-7*
~~P56~~P81 Breathing Air System								
~~F002~~ F252	~~2~~1	Inboard Isolation ~~check~~Manual valve~~s (h1)~~	2	A~~,C~~	I,~~A~~P	L~~,S~~	RO	9.3-10
~~F001~~ F251	~~2~~1	Outboard Isolation Manual valve~~s~~	2	A	I,P	L	RO	9.3-10