3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

The plant events affecting the mechanical systems, components and equipment are summarized in Table 3.9-1 in two groups: (1) plant operating events during which thermal-hydraulic transients occur, and (2) dynamic loading events due to accidents, earthquakes and certain operating conditions. The number of cycles associated with each event for the design of the reactor pressure vessel (RPV), for example, are listed in Table 3.9-1. The plant operating conditions are identified as normal, upset, emergency, faulted, or testing as defined in Subsection 3.9.3.1.1. Appropriate Service Levels (A, B, C, D or testing), as defined in ASME Code Section III, are designated for design limits. The design and analysis of safety-related piping and equipment using specific applicable thermal-hydraulic transients which are derived from the system behavior during the events listed in Table 3.9-1 are documented in the design specification and/or stress report of the respective equipment. Table 3.9-2 shows the loading combinations and the standard acceptance criteria.

3.9.1.2 Computer Programs Used in Analyses

The computer programs used in the analyses of the major safety-related components are described in Appendix 3D.

The computer programs used in the analyses of Seismic Category I components are maintained and controlled using appropriate software maintenance procedures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems.

The updates to Appendix 3D will be provided to indicate any additional programs used or the later version of the described programs, and the method of their verification.

[*Computer codes used for piping system dynamic analysis shall be benchmarked against the NRC Benchmark Problems for ABWR, defined in Reference 3.9-11. The results of piping dynamic analysis shall be compared with the results of the Benchmark Problems provided in Reference 3.9-11. The piping results to be compared and evaluated and the acceptance criteria or range of acceptable values are specified in Reference 3.9-11. Any deviations from these values as well as justification for such deviations shall be documented and submitted to the NRC staff for review and approval before initiating the final certified piping analysis.*]*

* See Subsection 3.9.1.7.

3.9.1.3 Experimental Stress Analysis

The following subsections list those NSSS components for which experimental stress analyses are performed in conjunction with analytical evaluations. The experimental stress analysis methods used are in compliance with the provisions of Appendix II of ASME Code Section III.

3.9.1.3.1 Piping Snubbers and Restraints

The following components have been tested to verify their design adequacy:

(1) Piping seismic snubbers

(2) Pipe whip restraints

Descriptions of the snubber and whip restraint tests are contained in Subsection 3.9.3.4 and Section 3.6, respectively.

3.9.1.3.2 Fine Motion Control Rod Drive (FMCRD)

Experimental data was used in developing the hydraulic analysis computer program called FMCRD01. The output of FMCRD01 is used in the dynamic analysis of both ASME Code and non-Code parts. Pressures used in the analysis of these parts are also determined during actual testing of prototype control rod drives.

3.9.1.4 Considerations for the Evaluation of Faulted Condition

All Seismic Category I equipment are evaluated for the faulted (Service Level D) loading conditions identified in Tables 3.9-1 and 3.9-2. In all cases, the calculated actual stresses are within the allowable Service Level D limits. The following subsections address the evaluation methods and stress limits used for the equipment and identify the major components evaluated for faulted conditions. Additional discussions of faulted analysis can be found in Subsections 3.9.2.5, 3.9.3, and 3.9.5.

Deformations under faulted conditions are evaluated in critical areas and the necessary design deformation limits (e.g., as clearance limits) are satisfied.

3.9.1.4.1 Control Rod Drive System Components

3.9.1.4.1.1 Fine Motion Control Rod Drive

The fine motion control rod drive (FMCRD) major components that are part of the reactor coolant pressure boundary (RCPB) are analyzed and evaluated for the faulted conditions in accordance with ASME Code Section III, Appendix F.

3.9.1.4.1.2 Hydraulic Control Unit

The hydraulic control unit (HCU) is analyzed and tested for withstanding the faulted condition loads. Dynamic tests establish the “g” loads in horizontal and vertical directions as the HCU

capability for the frequency range that is likely to be experienced in the plant. These tests also insure that the scram function of the HCU can be performed under these loads. Dynamic analyses of the HCU with the mounting beams are performed to assure that the maximum faulted condition loads remain below the HCU capability.

3.9.1.4.2 Reactor Pressure Vessel Assembly

The reactor pressure vessel (RPV) assembly includes: (1) the RPV boundary out to and including the nozzles and housings for FMCRD, internal pump and incore instrumentation; (2) support skirt; and (3) the shroud support, including legs, cylinder, and plate. The design and analyses of these three parts comply with Subsections NB, NF, and NG, respectively, of ASME Code Section III. For faulted conditions, the reactor vessel is evaluated using elastic analysis. For the support skirt and shroud support, an elastic analysis is performed, and buckling is evaluated for compressive load cases for certain locations in the assembly.

3.9.1.4.3 Core Support Structures and Other Safety Reactor Internal Components

The core support structures and other safety class reactor internal components are evaluated for faulted conditions. The bases for determining the faulted loads for seismic events and other dynamic events are given in Section 3.7 and Subsection 3.9.5, respectively. The allowable Service Level D limits for evaluation of these structures are provided in Subsection 3.9.5.

3.9.1.4.4 RPV Stabilizer and FMCRD—and Incore Housing Restraints (Supports)

The calculated maximum stresses meet the allowable stress limits stated in Table 3.9-1 and 3.9-2 under faulted conditions for the RPV stabilizer and support for the FMCRD housing and incore housing for faulted conditions. These supports restrain the components during earthquake, pipe rupture or other Reactor Building vibration events.

3.9.1.4.5 Main Steam Isolation Valve, Safety/Relief Valve and Other ASME Class 1 Valves

Elastic analysis methods and standard design rules, as defined in ASME Code Section III, are utilized in the analysis of the pressure boundary, Seismic Category I, ASME Class 1 valves. The Code-allowable stresses are applied to assure integrity under applicable loading conditions, including faulted condition. Subsection 3.9.3.2.4 discusses the operability qualification of the major active valves, including MSIV and the main steam SRV for seismic and other dynamic conditions. The allowable stresses for various operating conditions, including faulted, for active ASME Class 1 valves are provided in Footnote 12 of Table 3.9-2.

3.9.1.4.6 ECCS and SLC Pumps, RRS and RHR Heat Exchangers, RCIC Turbine, and RRS Motor

The ECCS (RHR, RCIC and HPCF) pumps, SLC pumps, RHR heat exchangers, and RCIC turbine are analyzed for the faulted loading conditions. The ECCS and SLC pumps are active ASME Class 2 components. The allowable stresses for active pumps are provided in a footnote to Table 3.9-2.

The RCPB components of the Reactor Recirculation System (RRS) pump motor cover and Recirculation Motor Cooling (RMC) Subsystem heat exchanger are ASME Class 1 and Class 2, respectively, and are analyzed for the faulted loading conditions. All equipment stresses are within the elastic limits.

3.9.1.4.7 Fuel Storage and Refueling Equipment

Storage, refueling, and servicing equipment which is important to safety is classified as essential components per the requirements of 10CFR50 Appendix A. This equipment and other equipment which, in case of a failure, would degrade an essential component is defined in Subsection 9.1 and is classified as Seismic Category I. These components are subjected to an elastic dynamic finite-element analysis to generate loadings. This analysis utilizes appropriate floor response spectra and combines loads at frequencies up to 33 Hz for seismic loads and up to 60 Hz for other dynamic loads in three directions. Imposed stresses are generated and combined for normal, upset, and faulted conditions. Stresses are compared, depending on the specific safety class of the equipment, to Industrial Codes, ASME, ANSI or Industrial Standards, AISC allowables.

3.9.1.4.8 Fuel Assembly (Including Channel)

BWR fuel assembly (including channel) design bases, and analytical and evaluation methods, including those applicable to the faulted conditions, are the same as those contained in References 3.9-1 and 3.9-2.

3.9.1.4.9 ASME Class 2 and 3 Vessels

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 vessels. The equivalent allowable stresses using elastic techniques are obtained from NC/ND-3300 and NC-3200 of ASME Code Section III. These allowables are above elastic limits.

3.9.1.4.10 ASME Class 2 and 3 Pumps

Elastic analysis methods are used for evaluating faulted loading conditions for Class 2 and 3 pumps. The equivalent allowable stresses for nonactive pumps using elastic techniques are obtained from NC/ND-3400 of ASME Code Section III. These allowables are above elastic limits. The allowables for active pumps are provided in footnote 12 to Table 3.9-2.

3.9.1.4.11 ASME Class 2 and 3 Valves

Elastic analysis methods and standard design rules are used for evaluating faulted loading conditions for Class 2 and 3 valves. The equivalent allowable stresses for nonactive valves using elastic techniques are obtained from NC/ND-3500 of ASME Code Section III. These allowables are above elastic limits. The allowables for active valves are provided in footnote 12 of Table 3.9-2.

3.9.1.4.12 ASME Class 1, 2 and 3 Piping

Elastic analysis methods are used for evaluating faulted loading conditions for Class 1, 2, and 3 piping. The equivalent allowable stresses using elastic techniques are obtained from NB/NC/ND-3600 of the ASME Code Section III. The allowables for functional capability are provided in footnote 9 of Table 3.9-2.

3.9.1.5 Inelastic Analysis Methods

Inelastic analysis is only applied to ABWR components to demonstrate the acceptability of three types of postulated events. Each event is an extremely low-probability occurrence and the equipment affected by these events would not be reused. These three events are:

(1) Postulated gross piping failure

(2) Postulated blowout of a reactor internal recirculation (RIP) motor casing due to a weld failure

(3) Postulated blowout of a control rod drive (CRD) housing due to a weld failure

The loading combinations and design criteria for pipe whip restraints utilized to mitigate the effects of postulated piping failures are provided in Subsection 3.6.2.3.3.

In the case of the RIP motor casing failure event, there are specific restraints applied to mitigate the effects of the failure. The mitigation arrangement consists of lugs on the RPV bottom head to which are attached two long rods for each RIP. The lower end of each rod engages two lugs on the RIP motor/cover. The use of inelastic analysis methods is limited to the middle slender body of the rod itself. The attachment lugs, bolts and clevises are shown to be adequate by elastic analysis. The selection of stainless steel for the rod is based on its high ductility assumed for energy absorption during inelastic deformation.

The mitigation for a failure in the CRD housing welded attachment (see Subsection 4.6.1.2.2.9) is by somewhat different means than are those of the RIP, in that the components with regular functions also function to mitigate the weld failure effect. The components are specifically:

(1) Core plate

(2) Control rod guide tube (CRGT)

(3) CRD housing

(4) CRD outer tube and middle flange, and

(5) Bayonet connector of CRD (internal CRD blowout support)

Only the cylindrical body of the CRGT deforms inelastically with other components deforming elastically in energy absorption. All other components are evaluated elastically, except the CRGT base, which is evaluated by the limit load method but is assumed conservatively in load prediction to absorb less energy by deforming elastically.

In elastic analyses for the latter two events, together with the criteria used for evaluation, are consistent with the procedures described in Subsection 3.6.2.3.3 for a pipe whip restraint. Figure 3.9-6 shows the stress-strain curve used for the inelastic restraints. The component evaluations are consistent with the elastic or limit load method of Appendix F of the ASME Code Section III.

3.9.1.6 Welding Methods and Acceptance Criteria for ASME Code Welding and Welding of Non-ASME Pressure Retaining Piping.

3.9.1.6.1 ASME Code Welding

Welding activities for pressure boundary and core support structure shall be performed in accordance with the requirements of Section III or Section VIII as applicable, of the ASME Code. The required nondestructive examination and acceptance criteria are stated in Table 3.9-10. Component supports shall be fabricated and examined in accordance with the requirements of Subsection NF of Section III of the ASME Code and NCIG-01 (Reference 3.9-12).

3.9.1.6.2 Welding of Non-ASME Pressure Retaining Piping

Welding activities involving non-ASME pressure retaining piping shall be accomplished in accordance with written procedures and shall meet the requirements of the ANSI B31.1 Code. The weld acceptance criteria shall be as defined for the applicable nondestructive examination method described in ANSI B31.1 Code.

3.9.1.7 Piping Design Acceptance Criteria

[*Table 7 of DCD/Introduction identifies the piping design acceptance criteria, which, if changed, requires NRC Staff review and approval prior to implementation. The applicable portions of the Tier 2 sections, tables, and figures identified on Table 7 of DCD/Introduction for this restriction, are italicized on the sections, tables, and figures themselves.*]*

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

The overall test program is divided into two phases: the preoperational test phase and the initial startup test phase. Piping vibration, thermal expansion and dynamic effects testing will be performed during both of these phases, as described in Chapter 14. Subsections 14.2.12.1.51, 14.2.12.2.10 and 14.2.12.2.11 relate the specific role of this testing to the overall test program.

* See section 3.5 of DCD/Introduction.

Discussed below are the general requirements for this testing. It should be noted that, because one goal of the dynamic effects testing is to verify the adequacy of the piping support system, such components are addressed in the subsections that follow. However, the more specific requirements of the design and testing of the piping support system are described in Subsection 3.9.3.4.1.

3.9.2.1.1 Vibration and Dynamic Effects Testing

The purpose of these tests is to confirm that the piping, components, restraints and supports of specified high- and moderate-energy systems have been designed to withstand the dynamic effects of steady-state flow-induced vibration (FIV) and anticipated operational transient conditions. The general requirements for vibration and dynamic effects testing of piping systems are specified in Regulatory Guide 1.68, “Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors”. More specific vibration testing requirements are defined in ANSI/ASME OM3, “Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems”. Preparation of detailed test specifications will be in full accordance with this standard and will address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points and acceptance criteria. The development and specification of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary are discussed in more detail below.

3.9.2.1.1.1 Measurement Techniques

There are essentially three methods available for determining the acceptability of steady state and transient vibration for the affected systems. These three measurement techniques are (1) visual observation, (2) local measurements, or (3) remotely monitored/recorded measurements. The technique used in each case will depend on such factors as (1) the safety significance of the particular system, (2) the expected mode and/or magnitude of the vibration, (3) the assessability of the system during designated testing conditions, or (4) the need for a time history recording of the vibratory behavior. Typically, the systems where vibration has the greatest safety implication will be subject to more rigorous testing and precise instrumentation requirements and, therefore, will require remote monitoring techniques. Local measurement techniques, such as the use of a hand-held vibrometer, are more appropriate in cases where it is expected that the vibration will be less complex and of lessor magnitude. Many systems that are assessable during the preoperational test phase and that do not show significant intersystem interactions will fall into this category. Visual observations are utilized where vibration is expected to be minimal and the need for a time history record of transient behavior is not anticipated. However, unexpected visual observations or local indications may require that a more sophisticated technique be used. Also, the issue of assessability should be considered. Application of these measurement techniques is detailed in the appropriate testing specification consistent with the guidelines contained in ANSI/ASME OM3.

3.9.2.1.1.2 Monitoring Requirements

As described in Subsections 14.2.12.1.51, 14.2.12.2.10 and 14.2.12.2.11, all safety-related piping systems will be subjected to steady-state and transient vibration measurements. The scope of such testing shall include safety-related instrumentation piping and attached small-bore piping (branch piping). Special attention should be given to piping attached to pumps, compressors, and other rotating or reciprocating equipment. Monitoring location selection considerations should include the proximity of isolation valves, pressure or flow control valves, flow orifices, distribution headers, pumps and other elements where shock or high turbulence may be of concern. Location and orientation of instrumentation and/or measurements will be detailed in the appropriate test specification. Monitored data should include actual deflections and frequencies as well as related system operating conditions. The time duration of data recording should be sufficient to indicate whether the vibration is continuous or transient. Steady-state monitoring should be performed at critical conditions such as minimum or maximum flow, or abnormal combinations or configurations of system pumps or valves. Transient monitoring should include anticipated system and total plant operational transients where critical piping or components are expected to show significant response. Steady-state conditions and transient events to be monitored will be detailed in the appropriate testing specifications consistent with OM3 guidelines.

3.9.2.1.1.3 Test Evaluation and Acceptance Criteria

The piping response to test conditions shall be considered acceptable if the review of the test results indicates that the piping responds in a manner consistent with predictions of the stress report and/or that piping stresses are within ASME Code Section III (NB-3600) limits. Acceptable limits are determined after the completion of piping systems stress analysis and are provided in the piping test specifications.

To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. For steady-state and transient vibration, the pertinent acceptance criteria are usually expressed in terms of maximum allowable displacement/deflection. Visual observation should only be used to confirm the absence of significant levels of vibration and not to determine acceptability of any potentially excessive vibration. Therefore, in some cases, other measurement techniques will be required with appropriate quantitative acceptance criteria.

There are typically two levels of acceptance criteria for allowable vibration displacements/deflections. Level I criteria are bounding type criteria associated with safety limits, while Level 2 criteria are stricter criteria associated with system or component expectations. For steady-state vibration, the Level 1 criteria are based on the endurance limit (68.6 MPaG) to assure no failure from fatigue over the life of the plant. The corresponding Level 2 criteria are based on one half the endurance limit (34.3 MPaG). For transient vibration, the Level 1 criteria are based on either the ASME Code Section III upset primary stress limit or

the applicable snubber load capacity. Level 2 criteria are based on a given tolerance about the expected deflection value.

3.9.2.1.1.4 Reconciliation and Corrective Actions

During the course of the tests, the remote measurements will be regularly checked to verify compliance with acceptance criteria. If trends indicate that criteria may be violated, the measurements should be monitored at more frequent intervals. The test will be held or terminated as soon as criteria are violated. As soon as possible after the test hold or termination, appropriate investigative and corrective actions will be taken. If practicable, a walkdown of the piping and suspension system should be made in an attempt to identify potential obstructions or improperly operating suspension components. Hangers and snubbers should be positioned such that they can accommodate the expected deflections without bottoming out or extending fully. All signs of damage to piping supports or anchors shall be investigated.

Instrumentation indicating criteria failure shall be checked for proper operation and calibration, including comparison with other instrumentation located in the proximity of the excessive vibration. The assumptions used in the calculations that generated the applicable limits should be verified against actual conditions and discrepancies noted should be accounted for in the criteria limits. This may require a reanalysis at actual system conditions.

Should the investigation of instrumentation and calculations fail to reconcile the criteria violations, physical corrective actions may be required, including (1) identification and reduction or elimination of offending forcing functions, (2) detuning of resonant piping spans by appropriate modifications, (3) addition of bracing, or (4) changes in operating procedures to avoid troublesome conditions. Any such modifications will require retest to verify vibrations have been sufficiently reduced.

3.9.2.1.2 Thermal Expansion Testing

A thermal expansion preoperational and startup testing program performed through the use of visual observation and remote sensors has been established to verify that normal unrestrained thermal movement occurs in specified safety-related high- and moderate-energy piping systems. The purpose of this program is to ensure the following:

(1) The piping system during system heatup and cooldown is free to expand and move without unplanned obstruction or restraint in the x, y, and z directions.

(2) The piping system does shakedown after a few thermal expansion cycles.

(3) The piping system is working in a manner consistent with the assumption of the stress analysis.

(4) There is adequate agreement between calculated values and measured values of displacements.

(5) There is consistency and repeatability in thermal displacements during heatup and cooldown of the systems.

The general requirements for thermal expansion testing of piping systems are specified in Regulatory Guide 1.68, “Preoperational and Initial Startup Testing Programs for Water-Cooled Power Reactors.” More specific requirements are defined in ANSI/ASME OM7, “Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems.” Detailed test specifications will be prepared in full accordance with this standard and will address such issues as prerequisites, test conditions, precautions, measurement techniques, monitoring requirements, test hold points, and acceptance criteria. The development and specification of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary are discussed in more detail below.

3.9.2.1.2.1 Measurement Techniques

Verification of acceptable thermal expansion of specified piping systems can be accomplished by several methods. One method is to physically walkdown the piping system and verify by visual observation that free thermal movement is unrestrained. This might include verification that piping supports such as snubbers and spring hangers are not fully extended or bottomed out and that the piping (including branch lines and instrument lines) and its insulation is not in hard contact with other piping or support structures. Another method would involve local measurements, using a hand-held scale or ruler, against a fixed reference or by recording the position of a snubber or spring can. A more precise method would be using permanent or temporary instrumentation that directly measures displacement, such as a lanyard potentiometer, that can be monitored via a remote indicator or recording device. The technique to be used will depend on such factors as the amount of movement predicted and the assessability of the piping.

Measurement of piping temperature is also of importance when evaluating thermal expansion. This may be accomplished either indirectly via the temperature of the process fluid or by direct measurement of the piping wall temperature, and such measurements may be obtained either locally or remotely. The choice of technique used shall depend on such considerations as the accuracy required and the assessability of the piping.

3.9.2.1.2.2 Monitoring Requirements

As described in Subsections 14.2.12.1.51 and 14.2.12.2.10, all safety-related piping shall be included in the thermal expansion testing program. Thermal expansion of specified piping systems should be measured at both the cold and hot extremes of their expected operating conditions. Physical walkdowns and recording of hanger and snubber positions should also be conducted where possible considering assessability and local environmental and radiological conditions in the hot and cold states. Displacements and appropriate piping/process temperatures shall be recorded for those systems and conditions specified. Sufficient time shall

have passed before taking such measurements to ensure the piping system is at a steady-state condition. In selecting locations for monitoring piping response, consideration shall be given to the maximum responses predicted by the piping analysis. Specific consideration should also be given to the first run of pipe attached to component nozzles and pipe adjacent to structures requiring a controlled gap.

3.9.2.1.2.3 Test Evaluation and Acceptance Criteria

To ensure test data integrity and test safety, criteria have been established to facilitate assessment of the test while it is in progress. Limits of thermal expansion displacements are established prior to start of piping testing to which the actual measured displacements are compared to determine acceptability of the actual motion. If the measured displacement does not vary from the acceptance limit values by more than the specified tolerance, the piping system is responding in a manner consistent with the predictions and is therefore acceptable. The piping response to test conditions shall be considered acceptable if the review of the test results indicates that the piping responds in a manner consistent with the predictions of the stress report and/or that piping stresses are within ASME Code Section III (NB-3600) limits. Acceptable thermal expansion limits are determined after the completion of piping systems stress analysis and are provided in the piping test specifications. Level 1 criteria are bounding criteria based on ASME Code Section III stress limits. Level 2 criteria are stricter criteria based on the predicted movements using the calculated deflections plus a selected tolerance.

3.9.2.1.2.4 Reconciliation and Corrective Actions

During the course of the tests, the remote measurements will be regularly checked to verify compliance with acceptance criteria. If trends indicate that criteria may be violated, the measurements should be monitored at more frequent intervals. The test will be held or terminated as soon as criteria are violated. As soon as possible after the test hold termination, appropriate investigative and corrective actions will be taken. If practicable, a walkdown of the affected piping and suspension system should be made in an attempt to identify potential obstruction to free piping movement. Hangers and snubbers should be positioned within their expected cold and hot settings. All signs of damage to piping or supports shall be investigated.

Instrumentation indicating criteria failure shall be checked for proper operation and calibration, including comparison with other instrumentation located in the proximity of the out-of-bounds movement. Assumptions, such as piping temperature, used in the calculations that generated the applicable limits should be compared with actual test conditions. Discrepancies noted should be accounted for in the criteria limits, including possible reanalysis.

Should the investigation of instrumentation and calculations fail to reconcile the criteria violations or should the visual inspection reveal an unintended restraint, physical corrective actions may be required. This might include (1) complete or partial removal of an interfering structure, (2) replacing, readjusting or repositioning piping system supports, (3) modifying the

pipe routing, or (4) modifying system operating procedures to avoid the temperature conditions that resulted in the unacceptable thermal expansion.

3.9.2.1.3 Thermal Stratification in Feedwater Piping

This special test is part of the startup program to monitor the conditions and effects of temperature stratification that may exist during certain operating conditions in:

(1) The feedwater piping header inside and outside the containment.

(2) The short horizontal runs of the riser piping inside the containment where feedwater enters the vessel nozzles.

Stratification in the feedwater piping can occur during plant startup when hot CUW is added to the cold feedwater line. The hot CUW flows on top of the colder water in the feedwater line and does not mix with the cold water until mixing of the two streams occurs at the outer swing check isolation valve. Stratification for this condition can thus affect only the feedwater piping outside containment.

A second condition of plant operation which can cause stratification in the feedwater piping is when the plant is in hot standby condition following a scram. After a scram, the temperature of the entire feedwater line is hot when cold water is introduced to make up for decay heat boiloff in the RPV. The colder water flows along the bottom of the large diameter horizontal feedwater pipe at low flow rate, creating stratification. The temperature difference between the top and bottom of the pipe will decrease along the pipe in the direction of flow, but stratification could still exist in the feedwater piping inside the containment, since the swing check valves are not effective in mixing the cold water flowing along the bottom of the pipe.

The test program will consist of measurement of:

(1) Temperature around the circumference of the feedwater pipe at various locations inside and outside the containment.

(2) Strains at points of highest stress inside the containment.

(3) Measurements of pipe displacements and movements inside and outside the containment due to pipe bowing because of stratification.

This test will be performed in accordance with the general requirements of Regulatory Guide 1.68 and the more specific requirements in ANSI/ASME OM7. Detailed test procedures will be prepared in accordance with these documents. The development and specifications of the types of measurements required, the systems and locations to be monitored, the test acceptance criteria, and the corrective actions that may be necessary are discussed in Subsection 3.9.2.1.2.

The feedwater thermal stratification test is not required if the applicant can show that a test performed at a previous plant meets the requirements of this paragraph and is applicable to this plant.

3.9.2.2 Seismic Qualification of Safety-Related Mechanical Equipment (Including Other RBV Induced Loads)

This subsection describes the criteria for dynamic qualification of safety-related mechanical equipment and associated supports, and also describes the qualification testing and/or analyses applicable to the major components on a component-by-component basis. Seismic and other events that may induce Reactor Building Vibration (RBV) are considered (Appendix 3B). In some cases, a module or assembly consisting of mechanical and electrical equipment is qualified as a unit (e.g., ECCS pumps). These modules are generally discussed in this subsection and Subsection 3.9.3.2, rather than providing discussion of the separate electrical parts in Section 3.10. Electrical supporting equipment such as control consoles, cabinets, and panels are discussed in Section 3.10.

3.9.2.2.1 Tests and Analysis Criteria and Methods

The ability of equipment to perform its safety function during and after the application of a dynamic load is demonstrated by tests and/or analysis. The analysis is performed in accordance with Section 3.7. Selection of testing, analysis, or a combination of the two, is determined by the type, size, shape, and complexity of the equipment being considered. When practical, the equipment operability is demonstrated by testing. Otherwise, operability is demonstrated by mathematical analysis.

Equipment which is large, simple, and/or consumes large amounts of power is usually qualified by analysis or static bend test to show that the loads, stresses and deflections are less than the allowable maximum. Analysis and/or static bend testing is also used to show that there are no natural frequencies below 33 Hz for seismic loads and 60 Hz for other RBV loads[*]. If a natural frequency lower than 33 Hz in the case of seismic loads and 60 Hz in the case of other RBV-induced loads is discovered, dynamic tests and/or mathematical analyses may be used to verify operability and structural integrity at the required dynamic input conditions.

[*When the equipment is qualified by dynamic test, the response spectrum or time history of the attachment point is used in determining input motion.*][†]

Natural frequency may be determined by running a continuous sweep frequency search using a sinusoidal steady-state input of low magnitude. Dynamic load conditions are simulated by testing using random vibration input or single frequency input (within equipment capability)

* The 60 Hz frequency cutoff for dynamic analysis of suppression pool dynamic loads is the minimum requirement based on a generic Reference 3.9-8, using the missing strain energy method, performed for representative BWR equipment under high-frequency input loadings.

† See Section 3.10.

over the frequency range of interest. [*Whichever method is used, the input amplitude during testing envelopes the actual input amplitude expected during the dynamic loading condition.*]†

[*The equipment being dynamically tested is mounted on a fixture which simulates the intended service mounting and causes no dynamic coupling to the equipment.*]†

Equipment having an extended structure, such as a valve operator, may be analyzed by applying static equivalent dynamic loads at the center of gravity of the extended structure. In cases where the equipment structural complexity makes mathematical analysis impractical, a combination of testing and analysis in accordance with IEEE-344 is used to determine operational capability at maximum equivalent dynamic load conditions.

3.9.2.2.1.1 Random Vibration Input

When random vibration input is used, the actual input motion envelopes the appropriate floor input motion at the individual modes. However, single frequency input such as sine beats can be used, provided one of the following conditions are met:

(1) The characteristics of the required input motion are dominated by one frequency.

(2) The anticipated response of the equipment is adequately represented by one mode.

(3) The input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra will envelop the corresponding response spectra of the individual modes.

3.9.2.2.1.2 Application of Input Modes

When dynamic tests are performed, the input motion is applied to one vertical and one horizontal axis simultaneously. However, if the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction and vice versa, then the input motion is applied to one direction at a time. In the case of single frequency input, the time phasing of the inputs in the vertical and horizontal directions are such that a purely rectilinear resultant input is avoided.

3.9.2.2.1.3 Fixture Design

The fixture design simulates the actual service mounting and causes no dynamic coupling to the equipment.

3.9.2.2.1.4 Prototype Testing

Equipment testing is conducted on prototypes of the equipment to be installed in the plant.

3.9.2.2.2 Qualification of Safety-Related Mechanical Equipment

The following subsections discuss the testing or analytical qualification of the safety-related major mechanical equipment and other ASME Code Section III equipment, including equipment supports.

3.9.2.2.2.1 CRD and CRD Housing

The qualification of the CRD housing (with enclosed CRD) is done analytically, and the stress results of their analysis establish the structural integrity of these components. Preliminary dynamic tests are conducted to verify the operability of the control rod drive during a dynamic event. A simulated test, imposing dynamic deflection in the fuel channels up to values greater than the expected seismic response, is performed with the CRD demonstrated functioning satisfactorily.

The test was conducted in two phases due to facility limitations. The seismic test facility cannot be pressurized while shaking; therefore, the charging pressure of the hydraulic control unit is reduced to simulate the backpressure that is applied in the reactor. The appropriate adjustment was determined by first running scram tests with the full reactor pressure and with peak transient pressure. Then with the test vessel at atmospheric pressure, the scram tests were repeated with reduced charging pressures until the scram performance matched that of the pressurized tests. This was repeated for the peak pressure also. The seismic tests were then performed with the appropriate pressure adjustments for the conditions being tested. The tests were run for various vibration levels with fuel channel deflections being the independent variable. The test facility was driven to vibration levels that produced various channel deflections up to 41 mm and the scram curves recorded. The 41 mm channel deflection is several times the channel deflection calculated for the actual seismic condition. The correlation of the test with analysis is via the channel deflection not the housing structural analysis, since scramability is controlled by channel deflection, not housing deflection.

3.9.2.2.2.2 Core Support (Fuel Support and CR Guide Tube)

A detailed analysis imposing dynamic effects due to seismic and other RBV events is performed to show that the maximum stresses developed during these events are much lower than the maximum allowed for the component material.

3.9.2.2.2.3 Hydraulic Control Unit (HCU)

The HCU is analyzed for the seismic and other RBV loads faulted condition, and the maximum stress on the HCU frame is calculated to be below the maximum allowable for the faulted condition. As discussed in Subsection 3.9.1.4.1.2, the faulted condition loads are calculated to be below the HCU maximum capability.

3.9.2.2.2.4 Fuel Assembly (Including Channel)

BWR fuel channel design bases, analytical methods, and seismic considerations are similar to those contained in References 3.9-1 and 3.9-2. The resulting combined acceleration profiles, including fuel lift for all normal/upset and faulted events are to be shown less than the respective design basis acceleration profiles.

3.9.2.2.2.5 Reactor Internal Pump and Motor Assembly

The reactor internal pump (RIP) and motor assembly, including its appurtenances and support, is classified as Seismic Category I, but not active, and is designed to withstand the seismic forces, including other RBV loads. The qualification of the assembly is done analytically and with a dynamic test.

3.9.2.2.2.6 ECCS Pump and Motor Assembly

A prototype ECCS (RHR and HPCF) pump motor assembly is qualified for seismic and other RBV loads via a combination of dynamic analysis and dynamic testing. The complete motor 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 (see Subsection 3.9.3.2.1.4 for details).

The pump and motor assemblies, as units operating under seismic and other RBV load conditions, are qualified by dynamic analysis, and results of the analysis indicate that the pump and motor are capable of sustaining the above loadings without exceeding the allowable stresses (see Subsections 3.9.3.2.1.1 and 3.9.3.2.1.2 for details).

3.9.2.2.2.7 RCIC Pump and Turbine Assembly

The RCIC turbine is qualified for seismic and other RBV loads via a combination of static analysis and dynamic testing (Subsection 3.9.3.2.1.5). The turbine 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 assembly and the adequacy of bolting under operating, seismic, and other RBV loading conditions. The complete turbine 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.

3.9.2.2.2.8 Standby Liquid Control Pump and Motor Assembly

The SLCS positive displacement pump and motor assembly, which is mounted on a common base plate, is qualified analytically by static analysis of 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 allowables (Subsection 3.9.3.2.2).

3.9.2.2.2.9 RMC and RHR Heat Exchangers

A three-dimensional finite-element model is developed for each of the Recirculation Motor Cooling (RMC) and Residual Heat Removal (RHR) System heat exchangers and supports. The model is used to dynamically analyze the heat exchanger and its supports using the response spectrum analysis method, and to verify that the heat exchanger and supports can withstand seismic and other RBV loads. The same model is used to statically analyze and evaluate the nozzles due to the effect of the external piping loads and dead weight in order to ensure that nozzle load criteria and limits are met. Critical location stresses are evaluated and compared with the allowable stress criteria. The results of the analysis demonstrate that the stresses at all investigated locations are less than their corresponding allowable values.

3.9.2.2.2.10 Standby Liquid Control Storage Tank

The Standby Liquid Control Storage Tank (SLCST) is a cylindrical tank, with approximate dimensions of 3.05m diameter and 4.9m height, bolted to the concrete floor. The SLCST is qualified for seismic and other RBV loads by analysis for:

(1) Stresses in the tank bearing tank plate

(2) Bolt stresses

(3) Sloshing loads imposed at the sloshing natural frequency

(4) Minimum wall thickness

(5) Buckling

The results of this analysis confirm that the calculated stresses at all investigated locations are less than their corresponding allowable values.

3.9.2.2.2.11 Main Steam Isolation Valves

The main steam isolation valves (MSIVs) are qualified for seismic and other RBV loads. The fundamental requirement of the MSIV following an SSE or other faulted RBV loadings is to close and remain closed after the event. This capability is demonstrated by the test and analysis as outlined in Subsection 3.9.3.2.4.1.

3.9.2.2.2.12 Standby Liquid Control Valve (Injection Valve)

The motor-operated standby liquid control valve is qualified by type test to IEEE-344 for seismic and other RBV loads. The qualification test as discussed in Subsection 3.9.3.2.4.3 demonstrates the ability to remain operable after the application of horizontal and vertical dynamic loading in excess of the required response spectra. The valve and motor assemblies are qualified by dynamic analysis, and the results of the analysis indicate that the valve is capable of sustaining the dynamic loads without overstressing the pressure retaining components.

3.9.2.2.2.13 Main Steam Safety/Relief Valves

Due to the complexity of the structure and the performance requirements of the valve, the total assembly of the SRV (including electrical and pressure devices) is tested at dynamic accelerations equal to or greater than the combined SSE and other RBV loadings determined for the plant. Tests and analyses demonstrate the satisfactory operation of the valves during and after the test (Subsection 3.9.3.2.4.2).

3.9.2.2.2.14 Fuel Pool Cooling and Cleanup System Pump and Motor Assembly

A static analysis is performed on the pump and motor assembly of the Fuel Pool Cooling and Cleanup (FPC) System. This analysis shows that the pump and motor will continue to operate if subjected to a combination of SSE, other RBV, and normal operating loads. Analysis also ensures that pump running clearances, which include deflection of the pump shaft and pump pedestal, are met during seismic and other RBV loadings.

3.9.2.2.2.15 Other ASME III Equipment

Other equipment, including associated supports, is qualified for seismic and other RBV loads to ensure its functional integrity during and after the dynamic event. The equipment is tested, if necessary, to ensure its ability to perform its specified function before, during, and following a test.

Dynamic load qualification is done by a combination of test and/or analysis (Subsection 3.9.2.2.1). Natural frequency, when determined by an exploratory test, is in the form of a single-axis continuous-sweep frequency search using a sinusoidal steady-state input at the lowest possible amplitude which is capable of determining resonance. The search is conducted on each principal axis with a minimum of two continuous sweeps over the frequency range of interest at a rate no greater than one octave per minute. If no resonances are located, then the equipment is considered as rigid and single frequency tests at every one third octave frequency interval are acceptable. Also, if all natural frequencies of the equipment are greater than 33 Hz for seismic loads and 60 Hz for other RBV loads, the equipment may be considered rigid and analyzed statically as such. In this static analysis, the dynamic forces on each component are obtained by concentrating the mass at the center of gravity and multiplying the mass by the appropriate floor acceleration. The dynamic stresses are then added to the operating stresses and a determination made of the adequacy of the strength of the equipment. The search for the natural frequency is done analytically if the equipment shape can be defined mathematically and/or by prototype testing.

If the equipment is a rigid body while its support is flexible, the overall system can be modeled as a single-degree-of-freedom system consisting of a mass and a spring. The natural frequency of the system is computed; then the acceleration is determined from the floor response spectrum curve using the appropriate damping value. A static analysis is then performed using this acceleration value. In lieu of calculating the natural frequency, the peak acceleration from the

spectrum curve is used. The critical damping values for welded steel structures from Table 3.7-1 are employed.

In case the equipment cannot be considered as a rigid body, it can be modeled as a multi-degree-of-freedom system. It is divided into a sufficient number of mass points to ensure adequate representation. The mathematical model can be analyzed using the modal analysis technique or direct integration of the equations of motion. Specified structural damping is used in the analysis unless justification for other values can be provided. A stress analysis is performed using the appropriate inertial forces or equivalent static loads obtained from the dynamic analysis of each mode.

For a multiple-degree-of-freedom modal analysis, the modal response accelerations can be taken directly from the applicable floor response spectrum. The maximum spectral values within ±10% band of the calculated frequencies of the equipment are used for computation of modal dynamic response inertial loading. The total dynamic stress is obtained by combining the modal stresses. The dynamic stresses are added to the operating stresses using the loading combinations stipulated in the specific equipment specification and then compared with the allowable stress levels.

If the equipment being analyzed has no definite orientation, the worst possible orientation is considered. Furthermore, equipment is considered to be in its operational configuration (i.e., filled with the appropriate fluid and/or solid). The investigation ensures that the point of maximum stress is considered. Lastly, a check is made to ensure that partially filled or empty equipment do not result in higher response than the operating condition. The analysis includes an evaluation of the effects of the calculated stresses on mechanical strength, alignment, electrical performance (microphonics, contact bounce, etc.), and noninterruption of function. Maximum displacements are computed and interference effects determined and justified.

Individual devices are tested separately, when necessary, in their operating condition. Then the component to which the device is assembled is tested with a similar but inoperative device installed upon it.

The equipment, component, or device to be tested is mounted on the vibration generator in a manner that simulates the final service mounting. If the equipment is too large, other means of simulating the service mounting are used. Support structures such as air conditioning units, consoles, racks, etc., could be vibration tested without the equipment and/or devices being in operation, provided they are performance tested after the vibration test. However, the components are in their operational configuration during the vibration test. The goal is to determine that, at the specified vibratory accelerations, the support structure does not amplify the forces beyond that level to which the devices have been qualified.

Equipment could alternatively be qualified by presenting historical performance data which demonstrates that the equipment satisfactorily sustains dynamic loads which are equal to or

greater than those specified for the equipment, and that the equipment performs a function equal to or better than that specified for it.

Equipment for which continued function is not required after a seismic and other RBV loads event, but whose postulated failure could produce an unacceptable influence on the performance of systems having a primary safety function, is evaluated. Such equipment is qualified to the extent required to ensure that an SSE, including other RBV loads, in combination with normal operating conditions, would not cause unacceptable failure. Qualification requirements are satisfied by ensuring that the equipment in its functional configuration, complete with attached appurtenances, remains structurally intact and affixed to the interface. The structural integrity of internal components is not required; however, the enclosure of such components is required to be adequate to ensure their confinement. Where applicable, fluid or pressure boundary integrity is demonstrated. With a few exceptions, simplified analytical techniques are adequate.

Historically, it has been shown that the main cause for equipment damage during a dynamic excitation has been the failure of its anchorage. Stationary equipment is designed with anchor bolts or other suitable fastening strong enough to prevent overturning or sliding. The effect of friction on the ability to resist sliding is neglected. The effect of upward dynamic loads on overturning forces and moments is considered. Unless specifically specified otherwise, anchorage devices are designed in accordance with the requirements of ASME Code Section III, Division 1, Subsection NF, or the AISC Manual of Steel Construction and ACI 318.

Dynamic design data are provided in the form of acceleration response spectra for each floor area of the equipment. Dynamic data for the ground or building floor to which the equipment is attached is used. For the case of equipment having supports with different dynamic motions, the most severe floor response spectrum is applied to all of the supports.

Refer to Subsections 3.9.3.2.3.1.4 and 3.9.3.2.5.1.2 for additional information on the dynamic qualification of active pumps and valves, respectively.

3.9.2.2.2.16 Supports

Subsections 3.9.3.4 and 3.9.3.5 address analyses or tests that are performed for component supports to assure their structural capability to withstand seismic and other dynamic excitations.

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

The major reactor internal components within the vessel are subjected to extensive testing coupled with dynamic system analyses to properly evaluate the resulting flow-induced vibration (FIV) phenomena during normal reactor operation and from anticipated operational transients. The response of the reactor internals to operational flow transients and steady-state conditions are evaluated for STP 3, which will be the ABWR prototype reactor, according to

the STP 3 ABWR Prototype Reactors Internals Flow-Induced Vibration Assessment Program (Reference 3.9-13).

In general, the vibration forcing functions for operational flow transients and steady-state conditions are not predetermined by detailed analysis. Special analyses of the response signals measured for reactor internals of many similar designs are performed to obtain the parameters which determine the amplitude and model contributions in the vibration responses. These studies provide useful predictive information for extrapolating the results from tests of components with similar designs to components of different designs. This vibration prediction method is appropriate where standard hydrodynamic theory cannot be applied due to complexity of the structure and flow conditions. Elements of the vibration prediction method are outlined as follows:

(1) Dynamic analyses of major components and subassemblies are performed to identify vibration modes and frequencies. The analysis models used for Seismic Category I structures are similar to those outlined in Subsection 3.7.2.

(2) Data from previous plant vibration measurements are assembled and examined to identify predominant vibration response modes of major components. In general, response modes are similar but response amplitudes vary among BWRs of differing size and design.

(3) Parameters are identified which are expected to influence vibration response amplitudes among the several reference plants. These include hydraulic parameters such as velocity and steam flow rates and structural parameters such as natural frequency and significant dimensions.

(4) Correlation functions of the variable parameters are developed which, multiplied by response amplitudes, tend to minimize the statistical variability between plants. A correlation function is obtained for each major component and response mode.

(5) Predicted vibration amplitudes for components of the prototype plant are obtained from these correlation functions based on applicable values of the parameters for the prototype plant. The predicted amplitude for each dominant response mode is stated in terms of a range taking into account the degree of statistical variability in each of the correlations. The predicted mode and frequency are obtained from the dynamic analyses of item (1).

The dynamic modal analysis forms the basis for interpretation of the preoperational and initial startup test results (Subsection 3.9.2.4). Modal stresses are calculated and relationships are obtained between sensor response amplitudes and peak component stresses for each of the lower normal modes. The allowable amplitude in each mode is that which produces a peak stress amplitude of ±68.6 MPa.

Vibratory loads are continuously applied during normal operation and the stresses are limited to ±68.6 MPa to prevent fatigue failure. Prediction of vibration amplitudes, mode shapes, and frequencies of normal reactor operations are based on statistical extrapolation of actual measured results on the same or similar components in reactors now in operation.

The dynamic loads due to flow-induced vibration from the feedwater jet impingement have no significant effect on the steam separator assembly. Analyses are performed to show that the impingement feedwater jet velocity is below the critical velocity. Also, it can be shown that the excitation frequency of the steam separator skirt is very different from the natural frequency of the skirt.

The calculated stresses due to hydrodynamic forces during core flooding operation are small and considered negligible when compared to the design-allowable stresses. Locations for which calculations were made include the weld joints, elbows, and rings.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Reactor internals vibration measurement and inspection programs are conducted during preoperational and initial startup testing in accordance with guidelines of Regulatory Guide 1.20 for prototype reactor internals. A flanged nozzle is provided in the top head of the reactor pressure vessel for bolting of the flange associated with the prototype test instrumentation. These programs are conducted in the three phases described as follows:

(1) **Preoperational Tests Prior to Fuel Loading:** Steady-state test conditions include balanced recirculation system operation and unbalanced operation over the full range of flow rates up to rated flow. Transient flow conditions include single and multiple pump trips from rated flow. This subjects major components to a minimum of 10^6 cycles of vibration at the anticipated dominant response frequency and at the maximum response amplitudes. Vibration measurements are obtained during this test and a close visual inspection of internals is conducted before and after the test.

(2) **Precritical Testing with Fuel:** This vibration measurement series is conducted with the reactor assembly complete but prior to reactor criticality. Flow conditions include balanced, unbalanced, and transient conditions as for the first test series. The purpose of this series is to verify the anticipated effects of the fuel on the vibration response of internals. Previous vibration measurements in BWRs (Reference 3.9-3) have shown that the fuel adds damping and reduces vibration amplitudes of major internal structures; thus, the first test series (without fuel) is a conservative evaluation of the vibration levels of these structures.

(3) **Initial Startup Testing:** Vibration measurements are made during reactor startup at conditions up to 100% rated flow and power. Balance, unbalanced, and transient conditions of recirculation system operation will be evaluated. The primary purpose of this test series is to verify the anticipated effect of two-phase flow on the vibration

response of internals. Previous vibration measurements in BWRs (Reference 3.9-3) have shown that the effect of the two-phase flow is to broaden the frequency response spectrum and diminish the maximum response amplitude of the shroud and core support structures.

Vibration sensor types may include strain gauges, displacement sensors (linear variable transformers), and accelerometers.

Accelerometers are provided with double integration signal conditioning to give a displacement output. Sensor locations include the following:

(1) Top of shroud head, lateral acceleration(displacement)

(2) Top of shroud, lateral displacement

(3) Control rod drive housings, bending strain

(4) Incore housings, bending strain

(5) Core flooder internal piping, bending strain

In addition to these components, vibration of the core spray sparger is measured during preoperational testing of that system at the designated prototype.

In all prototype plant vibration measurements, only the dynamic component of strain or displacement is recorded. Data are recorded on magnetic tape and provision is made for selective online analysis to verify the overall quality and level of the data. Interpretation of the data requires identification of the dominant vibration modes of each component by the test engineer using frequency, phase, and amplitude information for the component dynamic analyses. Comparison of measured vibration amplitudes to predicted and allowable amplitudes is then to be made on the basis of the analytically obtained normal mode which best approximates the observed mode.

The visual inspections conducted prior to and following preoperational testing are for vibration, wear, or loose parts. At the completion of preoperational testing, the reactor vessel head and the shroud head are removed, the vessel is drained, and major components are inspected on a selected basis. The inspections cover the shroud, shroud head, core support structures, recirculation internal pumps, the peripheral control rod drive, and incore guide tubes. Access is provided to the reactor lower plenum for these inspections.

The analysis, design and/or equipment that are to be utilized in a facility will comply with Regulatory Guide 1.20 as explained below.

Regulatory Guide 1.20 describes a comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing. The vibration assessment program

meets the requirements of Criterion 1, Quality Standards and Records 10CFR50 Appendix A and Section 50.34 (Contents of Applications; Technical Information) of 10CFR50. This Regulatory Guide is applicable to the core support structures and other reactor internals.

STP 3 is classified as the prototype ABWR and will be instrumented and subjected to preoperation and startup flow testing to demonstrate that flow-induced vibrations similar to those expected during operation will not cause damage. In accordance with the requirement of Regulatory Guide 1.206 Section C.I.3.9.2.4 for prototype, the STP 3 assessment program addresses the flow modes, vibration monitoring, sensor types and locations, procedures and methods to be used to process and interpret the measured data, planned visual inspections, and planned comparisons of test results with analytical predictions. Also, as discussed in Regulatory Guide 1.20, the main steam lines in STP 3 will be instrumented with strain gages to provide measurements of pressure fluctuations due to flow-induced vibrations. Subsequent plants which have internals similar to those of the prototypes are also tested in compliance with the requirements of Regulatory Guide 1.20. Satisfactory vibration performance of internals in these plants is confirmed through preoperational flow testing followed by inspection for evidence of excessive vibration. Extensive vibration measurements in prototype plants, together with satisfactory operating experience in all BWR plants, have established the adequacy of reactor internal designs.

See Subsection 3.9.7.1 for COL license information pertaining to the reactor internals vibration testing program.

3.9.2.5 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

The faulted events that are evaluated are defined in Subsection 3.9.5.2.1. The loads that occur as a result of these events and the analysis performed to determine the response of the reactor internals are as follows:

(1) **Reactor Internal Pressures—**The reactor internal pressure differentials (Figure 3.9-1) due to assumed break of main steam or feedwater line are determined by analysis as described in Subsection 3.9.5.2.2. In order to assure that no significant dynamic amplification of load occurs as a result of the oscillatory nature of the blowdown forces during an accident, a comparison is made of the periods of the applied forces and the natural periods of the core support structures being acted upon by the applied forces. These periods are determined from a comprehensive vertical dynamic model of the RPV and internals with 12 degrees of freedom. Besides the real masses of the RPV and core support structures, account is made for the water inside the RPV.

(2) **External Pressure and Forces on the Reactor Vessel—**An assumed break of the main steamline, the feedwater line or the RHR line at the reactor vessel nozzle results in jet reaction and impingement forces on the vessel and asymmetrical pressurization of the annulus between the reactor vessel and the shield wall. These time-varying pressures are applied to the dynamic model of the reactor vessel system. Except for

the nature and locations of the forcing functions, the dynamic model and the dynamic analysis method are identical to those for seismic analysis as described below. The resulting loads on the reactor internals, defined as LOCA loads, are shown in Table 3.9-3.

(3) **Safety/Relief Valve Loads (SRV Loads)—**The discharge of the SRVs results in reactor building vibration (RBV) due to suppression pool dynamics (Appendix 3B). The response of the reactor internals to the RBV is also determined with the dynamic model and dynamic analysis method described below for seismic analysis.

(4) **LOCA Loads—**The assumed LOCA also results in RBV due to suppression pool dynamics (Appendix 3B), and the responses of the reactor internals are again determined with the dynamic model and dynamic analysis method used for seismic analysis. Various types of LOCA loads are identified on Table 3.9-2.

(5) **Seismic Loads—**The theory, methods, and computer codes used for dynamic analysis of the reactor vessel, internals, attached piping and adjoining structures are described in Section 3.7 and Subsection 3.9.1.2. Dynamic analysis is performed by coupling the lumped-mass model of the reactor vessel and internals with the building model to determine the system natural frequencies and mode shapes. The relative displacement, acceleration, and load response is then determined by either the time-history method or the response-spectrum method. The load on the reactor internals due to faulted event SSE are obtained from this analysis.

The above loads are considered in combination as defined in Table 3.9-2. The SRV, LOCA (SBL, IBL or LBL) and SSE loads defined in Table 3.9-2 are all assumed to act in the same direction. The peak colinear responses of the reactor internals to each of these loads are added by the square-root-of-the-sum-of-the-squares (SRSS) method. The resultant stresses in the reactor internal structures are directly added with stress resulting from the static and steady-state loads in the faulted load combination, including the stress due to peak reactor internal pressure differential during the LOCA. The reactor internals satisfy the stress deformation and fatigue limits as defined in Subsection 3.9.5.3.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Prior to initiation of the instrumented vibration measurement program for the prototype plant, extensive dynamic analyses of the reactor and internals are performed. The results of these analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test will be analyzed in detail.

The results of the data analyses, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. The additional knowledge gained from previous vibration tests has been utilized in the generation of the dynamic models for seismic and LOCA analyses for this plant.

The models used for this plant are similar to those used for the vibration analysis of earlier prototype BWR plants.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

This section delineates the criteria for selection and definition of design limits and loading combinations associated with normal operation, postulated accidents, and specified seismic and other Reactor Building vibration (RBV) events for the design of safety-related ASME Code components (except containment components, which are discussed in Section 3.8).

This section discusses the ASME Class 1, 2, and 3 equipment and associated pressure-retaining parts and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. A discussion of major equipment is included on a component-by-component basis to provide examples. Design transients and dynamic loading for ASME Class 1, 2, and 3 equipment are covered in Subsection 3.9.1.1. Seismic-related loads and dynamic analyses are discussed in Section 3.7. The suppression pool-related RBV loads are described in Appendix 3B. Table 3.9-2 presents the combinations of dynamic events to be considered for the design and analysis of all ABWR ASME Code Class 1, 2, and 3 components, component supports, core support structures and equipment. Specific loading combinations considered for evaluation of each specific equipment are derived from Table 3.9-2 and are contained in the design specifications and/or design reports of the respective equipment.

Piping loads due to the thermal expansion of the piping and thermal anchor movements at supports are included in the piping load combinations. All operating modes are evaluated and the maximum moment ranges are included in the fatigue evaluation. [*Piping systems with maximum operating temperatures of less than or equal to 65°C are not required to be analyzed for thermal expansion loading.*]*

[*Low-pressure piping systems that interface with the reactor coolant pressure boundary will be designed with a pipe wall thickness calculated for a pressure equal to 0.4 times the reactor coolant system pressure but no less than that of a schedule 40 pipe.*]* See Appendix 3M for additional information on intersystem LOCA.

Thermal stratification of fluids in a piping system is one of the specific operating conditions included in the loads and load combinations contained in the piping design specifications and design reports. It is known that stratification can occur in the feedwater piping during plant startup and when the plant is in hot standby conditions following scram (Subsection 3.9.2.1.3). [*If, during design or startup, evidence of thermal stratification is detected in any other piping system, then stratification will be evaluated. If it cannot be shown that the stresses in the pipe are low and that movement due to bowing is acceptable, then stratification will be treated as a*

* See Subsection 3.9.1.7.

design load. In general, if temperature differences between the top and bottom of the pipe are less than 27°C, it may be assumed that design specification and stress reports need not be revised to include stratification. The piping design reports shall be inspected to confirm that the piping systems have been designed for thermal stratification in accordance with the requirements of this paragraph.]*

Under thermally stratified flow conditions, it has been observed that a relatively thin dynamic interface region exists, which oscillates in a wave pattern. This results in undulation in the hot-to-cold interface region which produces thermal striping on the inside of the pipe wall. Thermal striping stresses are the result of differences between the pipe inside surface temperatures which vary with time due to the interface oscillation and the average through-wall temperatures. The results of the feedwater piping thermal striping stress analysis confirm that the feedwater thermal striping fatigue usage is minimal per the ASME Code, Section III, fatigue evaluation requirement; therefore, thermal striping fatigue effects are negligible.

[*Supplement 3 to Bulletin 88-08 involves the development of potential cyclic stratified flow and associated thermal striping that may occur because of possible leakage past the valve disk and out the valve stem packing gland. This flow stratification and striping may occur when the pressure on the upstream side of the valve is less than the RPV system pressure during normal operation. Sections of the RHR and HPCF Systems could be susceptible to unacceptable thermal stresses, owing to this phenomenon. To address the potential problem described in Supplement 3 to Bulletin 88-08, for the affected piping sections, it is required that either (1) the gate valve in each of the unisolable piping sections be located at a distance equal or greater than 25 pipe diameters from the RPV nozzle or (2) stress analysis be performed to show that stresses and fatigue from potential stratification and thermal striping are acceptable per ASME Code. Their requirements are incorporated in the design by note 32 of the RHR P&ID Figure 5.4-10 and note 29 of the HPCF P&ID Figure 6.3-7.*]*

The design life for the ABWR Standard Plant is 60 years. A 60-year design life is a requirement for all major plant components with reasonable expectation of meeting this design life. However, all plant operational components and equipment, except the reactor vessel, are designed to be replaceable, design life not withstanding. The design life requirement allows for refurbishment and repair, as appropriate, to assure that the design life of the overall plant is achieved. [*In effect, essentially all piping systems, components and equipment are designed for a 60-year design life. Many of these components are classified as ASME Class 2 or 3 or Quality Group D. In the event that any non-Class 1 components are subjected to cyclic loadings, including operating vibration loads and thermal transient effects, of a magnitude and/or duration so severe that the 60-year design life can not be assured by required Code calculations. COL applicants will identify these components and either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads. Components excluded from this requirement are:*

(1) Tees where mixing of hot and cold fluids occurs and thermal sleeves have been provided in accordance with the P&IDs.

(2) *Feedwater piping outside containment that is designed so cyclic loadings and stresses are no more severe than experienced by Class 1 piping inside the containment.*]*

Severe thermal transients that will be evaluated for possible effect on plant life are temperature rate changes faster than 830°C/h, when the total fluid temperature change is greater than 55°C.

[*The safety relief valve (SRV) discharge piping in the wetwell and the SRV Quenchers are subjected to severe thermal transients during SRV blowdown events. Therefore, the COL applicant will perform ASME Class 1 fatigue analyses of the ASME Class 3 SRV discharge piping in the wetwell and the SRV Quenchers.*]* The purpose of these fatigue evaluations is to confirm that the fatigue stresses are less than their allowables. The fatigue evaluations will include the SRV blowdown thermal transient loads, thermohydraulic loads, Safe Shutdown earthquake loads and the reactor building vibration loads due to SRV blowdown. Environmental effects will be considered in the fatigue analysis in accordance with the requirements for ASME Section III Class 1 carbon steel piping specified in Subsection 3.9.3.1.1.7.

The SRV discharge piping in the wetwell will be analyzed for SRV blowdown thermal stresses due to a step change in temperature inside the pipe from 32°C to 166°C. In order to minimize piping thermal stresses, no shear lugs will be welded to the SRV discharge piping.

The fatigue analysis of the Quencher will be performed in accordance with ASME Section III, Subsection NB-3200. The quencher will be analyzed for heat transfer transient during SRV blowdown where there is a step change in temperature inside the quencher from 20°C to 166°C, and the outside of the quencher remains at 20°C. The fatigue evaluation will also include the SRV discharge pipe applied thermal loads, thermohydraulic transient loads, safe shutdown earthquake (SSE) loads and the SRV blowdown reactor building vibration loads.

See Subsection 3.9.7.2 for COL license information requirements.

3.9.3.1.1 Plant Conditions

All events that the plant will or might credibly experience during a reactor year are evaluated to establish design basis for plant equipment. These events are divided into four plant conditions. The plant conditions described in the following paragraphs are based on event probability (i.e., frequency of occurrence as discussed in Subsection 3.9.3.1.1.5) and correlated to service levels for design limits defined in ASME B&PV Code Section III as shown in Tables 3.9-1 and 3.9-2.

3.9.3.1.1.1 Normal Condition

Normal conditions are any conditions in the course of system startup, operation in the design power range, normal hot standby (with condenser available), and system shutdown other than upset, emergency, faulted, or testing.

3.9.3.1.1.2 Upset Condition

An upset condition is any deviation from normal conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The upset conditions include system operational transients (SOT) which result from (1) any single operator error or control malfunction, (2) fault in a system component

* See Subsection 3.9.1.7.

requiring its isolation from the system, (3) a loss of load or power. Hot standby with the main condenser isolated is an upset condition.

3.9.3.1.1.3 Emergency Condition

An emergency condition includes deviations from normal conditions which require shutdown for correction of the condition(s) or repair of damage in the reactor coolant pressure boundary (RCPB). Such conditions have a low probability of occurrence but are included to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system. Emergency condition events include, but are not limited to, infrequent operational transients (IOT) caused by one of the following: (1) a multiple valve blowdown of the reactor vessel; (2) LOCA from a small break or crack (SBL) which does not depressurize the reactor systems, does not actuate automatically the ECCS operation, nor results in leakage beyond normal makeup system capacity, but which requires the safety functions of isolation of containment and shutdown and may involve inadvertent actuation of automatic depressurization system (ADS); (3) improper assembly of the core during refueling; or (4) improper or sudden start of one recirculation pump. Anticipated transient without scram (ATWS) or reactor overpressure with delayed scram (Tables 3.9-1 and 3.9-2) is an IOT classified as an emergency condition.

3.9.3.1.1.4 Faulted Condition

A faulted condition is any of those combinations of conditions associated with extremely low-probability postulated events whose consequences are such that the integrity and operability of the system may be impaired to the extent that considerations of public health and safety are involved. Faulted conditions encompass events, such as LOCAs, that are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These events are the most drastic that must be considered in the design and thus represent limiting design bases. Faulted condition events include, but are not limited to, one of the following: (1) a control rod drop accident; (2) a fuel-handling accident; (3) a main steamline or feedwater line break; (4) the combination of any small/intermediate break LOCA (SBL or IBL) with the safe shutdown earthquake, and a loss of offsite power; or (5) the safe shutdown earthquake plus large break LOCA (LBL) plus a loss of offsite power.

The IBL classification covers those breaks for which the ECCS operation will occur during the blowdown, and which results in reactor depressurization. The LBL classification covers the sudden, double-ended severance of a main steamline inside or outside the containment that results in transient reactor depressurization, or any pipe rupture of equivalent flow cross-sectional area with similar effects.

3.9.3.1.1.5 Correlation of Plant Condition with Event Probability

The probability of an event occurring per reactor year associated with the plant conditions is listed below. This correlation identifies the appropriate plant conditions and assigns the appropriate ASME Section III service levels for any hypothesized event or sequence of events.

Plant Condition	ASME Code Service Level	Event Encounter Probability per Reactor Year
Normal (planned)	A	1.0
Upset (moderate probability)	B	$1.0 > P \geq 10^{-2}$
Emergency (low probability)	C	$10^{-2} > P \geq 10^{-4}$
Faulted (extremely low probability)	D	$10^{-4} > P > 10^{-6}$

3.9.3.1.1.6 Safety Class Functional Criteria

For any normal or upset design condition event, Safety Class 1, 2, and 3 equipment and piping (Subsection 3.2.3) shall be capable of accomplishing its safety functions as required by the event and shall incur no permanent changes that could deteriorate its ability to accomplish its safety functions as required by any subsequent design condition event.

For any emergency or faulted design condition event, Safety Class 1, 2, and 3 equipment and piping shall be capable of accomplishing its safety functions as required by the event, but repairs could be required to ensure its ability to accomplish its safety functions as required by any subsequent design condition event.

For active Class 2 and 3 pumps and active Class 1, 2 and 3 valves, specific stress criteria to meet the functional requirements are identified in a footnote to Table 3.9-2. For piping the specific stress criteria for functional requirements are identified in footnote 9 of Table 3.9-2.

3.9.3.1.1.7 Environmental Effects on Fatigue Evaluation of Carbon Steel Piping

[*Environmental effects on the fatigue design of ASME Section III Class 1 carbon steel piping will be evaluated in accordance with GE document, 408HA414 (Reference 3.9-9). Additional fatigue evaluations for environmental effects are not required for any of the following conditions: (a) water temperature is below 245°C, (b) fittings, such as elbows and tees, that are conservatively designed and analyzed using the ASME Section III stress indices and (c) for transients having total cycle times of 10 seconds or less and no tensile hold time, provided that the oxygen content of the water does not exceed 0.3 ppm.*

Environmental effects are considered by increasing the local peak stress through four factors used as multipliers to the stress indices. The four factors are: (1) the notch factor, (2) the mean stress factor, (3) the environmental correction factor, and (4) the butt weld strength reduction factor.]*

3.9.3.1.2 Reactor Pressure Vessel Assembly

The reactor pressure vessel assembly consists of the RPV, vessel support skirt, and shroud support.

The reactor pressure vessel, vessel support skirt, and shroud support are constructed in accordance with ASME Code Section III. The shroud support consists of the shroud support plate and the shroud support cylinder and its legs. The RPV assembly components are classified as ASME Class 1. Complete stress reports on these components are prepared in accordance with ASME Code requirements. NUREG-0619 (Reference 3.9-5) is also considered for feedwater nozzle and other such RPV inlet nozzle design.

The stress analysis is performed on the reactor pressure vessel (RPV), vessel support skirt, and shroud support for various plant operating conditions (including faulted conditions) by using the elastic methods except as noted in Subsection 3.9.1.4.2. Loading conditions, design stress limits, and methods of stress analysis for the core support structures and other reactor internals are discussed in Subsection 3.9.5.

3.9.3.1.3 Main Steam (MS) System Piping

The piping systems extending from the reactor pressure vessel to and including the outboard MSIV are constructed in accordance with ASME Code Section III, Class 1 criteria. Stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 of ASME Code Section III.

The MS System piping extending from the outboard MSIV to the turbine stop valve is constructed in accordance with ASME Code Section III, Class 2 criteria.

Turbine stop valve (TSV) closure in the main steam (MS) piping system results in a transient that produces momentary unbalanced forces acting on the MS piping system. Upon closure of the TSV, a pressure wave is created and it travels at sonic velocity toward the reactor vessel through each MS line. Flow of steam into each MS line from the reactor vessel continues until the steam compression wave reaches the reactor vessel. Repeated reflections of the pressure wave at the reactor vessel and the TSV produce time varying pressures and velocities, throughout the MS lines.

The analysis of the MS piping TSV closure transient consists of a stepwise time-history solution of the steam flow equation to generate a time-history of the steam properties at numerous

* See Subsection 3.9.1.7.

locations along the pipe. Reaction loads on the pipe are determined at each elbow. These loads are composed of pressure-times-area, momentum change and fluid-friction terms.

The time-history direct integration method of analysis is used to determine the response of the MS piping system to TSV closure. The forces are applied at locations on the piping system where steam flow changes direction, thus causing momentary reactions. The resulting loads on the MS piping are combined with loads due to other effects specified in Subsection 3.9.3.1.

3.9.3.1.4 Recirculation Motor Cooling (RMC) Subsystem

The RMC System piping loop between the recirculation motor casing and the heat exchanger is constructed in accordance with ASME Code Section III, Subsection NB-3600. Stresses are calculated on an elastic basis and evaluated in accordance with NB-3600.

3.9.3.1.5 Recirculation Pump Motor Pressure Boundary

The motor casing of the recirculation internal pump is a part of and welded into an RPV nozzle and is constructed in accordance with requirements of an ASME Code Section III, Class 1 component. The motor cover is a part of the pump/motor assembly and is constructed as an ASME Class 1 component. These pumps are not required to operate during the SSE or after an accident.

3.9.3.1.6 Standby Liquid Control (SLC) Tank

The standby liquid control tank is constructed in accordance with the requirements of a ASME Code Section III, Class 2 component.

3.9.3.1.7 RRS and RHR Heat Exchangers

The primary and secondary sides of the RRS are constructed in accordance with the requirements of an ASME Code Section III, Class 2 component. The primary and secondary side of the RHR System heat exchanger is constructed as ASME Class 2 and Class 3 component, respectively.

3.9.3.1.8 RCIC Turbine

The RCIC turbine-pump is constructed in accordance with the requirementsof ASME Code Section III for Class 2 components.

3.9.3.1.9 ECCS Pumps

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

3.9.3.1.10 Standby Liquid Control (SLC) Pump

The SLC System pump is constructed in accordance with the requirements for ASME Code Section III, Class 2 components.

3.9.3.1.11 Standby Liquid Control (SLC) Valve (Injection Valve)

The SLC System injection valve is constructed in accordance with the requirements for ASME Code Section III, Class 1 components.

3.9.3.1.12 Main Steam Isolation and Safety/Relief Valves

The MSIVs and SRVs are constructed in accordance with ASME Code Section III, Subsection NB-3500, requirements for Class 1 components.

3.9.3.1.13 Safety/Relief Valve Piping and Quencher

The SRV discharge piping in the drywell extending from the relief valve discharge flange to the diaphragm floor penetration and the SRV discharge piping in the wetwell, extending from the diaphragm floor penetration to and including the quencher, is constructed in accordance with ASME Code Section III requirements for Class 3 components. In addition, all welds in the SRVDL piping in the wetwell above the surface of the suppression pool shall be non-destructively examined to the requirements of ASME Code Section III, Class 2.

3.9.3.1.14 Reactor Water Cleanup (CUW) System Pump and Heat Exchangers

The CUW pump and heat exchangers (regenerative and nonregenerative) are not part of a safety system and are non-Seismic Category I equipment. ASME Code Section III for Class 3 components is used as a guide in constructing the CUW System pump and heat exchanger components.

3.9.3.1.15 Fuel Pool Cooling and Cleanup System Pumps and Heat Exchangers

The pumps and heat exchangers are constructed in accordance with the requirements for ASME Code Section III Class 3 components.

3.9.3.1.16 ASME Class 2 and 3 Vessels

The Class 2 and 3 vessels (all vessels not previously discussed) are constructed in accordance with ASME Code Section III. The stress analysis of these vessels is performed using elastic methods.

3.9.3.1.17 ASME Class 2 and 3 Pumps

The Class 2 and 3 pumps (all pumps not previously discussed) are designed and evaluated in accordance with ASME Code Section III. The stress analysis of these pumps is performed using elastic methods. See Subsection 3.9.3.2 for additional information on pump operability.

3.9.3.1.18 ASME Class 1, 2 and 3 Valves

The Class 1, 2, and 3 valves (all valves not previously discussed) are constructed in accordance with ASME Code Section III.

All valves and their extended structures are designed to withstand the accelerations due to seismic and other RBV loads. The attached piping is supported so that these accelerations are not exceeded. The stress analysis of these valves is performed using elastic methods. See Subsection 3.9.3.2 for additional information on valve operability.

3.9.3.1.19 ASME Class 1, 2 and 3 Piping

The Class 1, 2 and 3 piping (all piping not previously discussed) is constructed in accordance with ASME Code Section III. For Class 1 piping, stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 of ASME Code Section III. For Class 2 and 3 piping, stresses are calculated on an elastic basis and evaluated in accordance with NC/ND-3600 of the Code.

[*The effects of displacement-limited, seismic anchor motions (SAM) due to an SSE shall be evaluated for ASME Class 1, 2 and 3 piping design in accordance with footnote 6 of Table 3.9-1.*]*

3.9.3.1.20 As-Built Stress Reports for ASME Class 1, 2 and 3 Piping Systems

[*For ASME Class 1, 2 and 3 piping systems, the as-built piping system shall be reconciled with the as-designed piping system. An as-built inspection of the pipe routing, location and orientation, the location, size, clearances and orientation of piping supports, and the location and weight of pipe mounted equipment shall be performed. This inspection will be performed by reviewing the as-built drawings containing verification stamps, and by performing a visual inspection of the installed piping system. The piping configuration and component location, size, and orientation shall be within the tolerances specified in the certified as-built piping stress report. The tolerances to be used for reconciliation of the as-built piping system with the as-designed piping system are provided in Reference 3.9-10. A reconciliation analysis using the as-built and as-designed information shall be performed. The certified as-built Stress Report shall document the results of the as-built reconciliation analysis.*]*

3.9.3.1.21 Pipe-Mounted Equipment Allowable Loads

[*The piping design reports shall document that the pipe applied loads on attached equipment; such as valves, pumps, tanks and heat exchangers, are less than the equipment vendor's specified allowable loads.*]*

3.9.3.1.22 ASME Class 1, 2 and 3 Piping System Clearance Requirements

[*ASME Class 1, 2 and 3 piping systems shall be designed to provide clearance from structures, systems, and components where necessary for the accomplishment of the structure, system, or component's safety function as specified in the representative structure or system design description. The maximum calculated piping system deflections under service conditions shall be verified that they do not exceed the minimum clearance between the piping system and*

* See Subsection 3.9.1.7.

nearby structures, systems, or components. The certified design stress report shall document that the clearance requirements have been met.]*

3.9.3.2 Pump and Valve Operability Assurance

Active mechanical (with or without electrical operation) equipment are Seismic Category I and each is designed to perform a mechanical motion for its safety-related function during the life of the plant under postulated plant conditions. Equipment with faulted condition functional requirements include active pumps and valves in fluid systems such as the RHR System, ECCS, and MS system.

This subsection discusses operability assurance of active ASME Code Section III pumps and valves, including motor, turbine or operator that is a part of the pump or valve (Subsection 3.9.2.2). The COL applicant must ensure that specific environmental parameters are properly defined and enveloped in the methodology for its specific plant and implemented in its equipment qualification program.

Safety-related valves and pumps are qualified by testing and analysis and by satisfying the stress and deformation criteria at the critical locations within the pumps and valves. Operability is assured by meeting the requirements of the programs defined in Subsection 3.9.2.2, design and qualification requirements Subsection 3.9.6, Sections 3.10 and 3.11, and the following subsections.

Section 4.4 of GE's Environmental Qualification Program (Reference 3.9-6) applies to this subsection, and the seismic qualification methodology presented therein is applicable to mechanical as well as electrical equipment.

3.9.3.2.1 ECCS Pumps, Motors and Turbine

Dynamic qualification of the ECCS (RHR, RCIC and HPCF) pumps with motor or turbine assembly is also described in Subsections 3.9.2.2.2.6 and 3.9.2.2.2.7.

3.9.3.2.1.1 Consideration of Loading, Stress, and Acceleration Conditions in the Analysis

In order to avoid damage to the ECCS pumps during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE, other RBV loads, and dynamic system loads are limited to the material elastic limit. A three-dimensional finite-element model of the pump and associated motor (see Subsections 3.9.3.2.2 and 3.9.3.2.1.5 for RCIC pump and turbine, respectively) and its support is developed and analyzed using the response spectrum and the dynamic analysis method. [*The same is analyzed due to static nozzle loads, pump thrust loads, and dead weight. Critical location stresses are compared with the allowable stresses and the critical location deflections with the allowables, and accelerations are checked to evaluate operability. The average membrane stress* σm *for the faulted condition loads is limited to 1.2S or approximately 0.75* σ_y (σ_y *= yield stress), and the maximum stress in local fibers* (σm *+ bending stress* σb) *is limited to 1.8S or approximately 1.1* σ_y. *The maximum faulted event nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur.*]*

Performing these analyses with the conservative loads stated and with the restrictive stress limits as allowables assures that critical parts of the pump and associated motor or turbine will not be damaged during the faulted condition and that the operability of the pump for post-faulted condition operation will not be impaired.

3.9.3.2.1.2 Pump/Motor Operation During and Following Dynamic Loading

Active ECCS pump/motor rotor combinations are designed to rotate at a constant speed under all conditions. Motors are designed to withstand short periods of severe overload. The high rotary inertia in the operating pump rotor and the nature of the random short duration loading characteristics of the dynamic event prevent the rotor from becoming seized. The seismic and other RBV loadings can be predicted to require only a slight increase, if any, in the torque (i.e., motor current) necessary to drive the pump at the constant design speed; therefore, the pump is expected to operate at the design speed during the faulted event loads.

The functional ability of the active pumps after a faulted condition is assured, since only normal operating loads and steady-state nozzle loads exist. For the active pumps, the faulted condition loads are greater than the normal condition loads only due to the SSE and other RBV transitory loads. These faulted events are infrequent and of relatively short duration compared to the design life of the equipment. Since it is demonstrated that the pumps would not be damaged during the faulted condition, the post-faulted condition operating loads will be no worse than the normal plant operating limits. This is assured by requiring that the imposed nozzle loads (steady-state loads) for normal conditions and post-faulted conditions be limited to the magnitudes of the normal condition nozzle loads. The post-faulted condition ability of the pumps to function under these applied loads is proven during the normal operating plant conditions for active pumps.

3.9.3.2.1.3 ECCS Pumps

All active ECCS (RHR, RCIC and HPCF) pumps are qualified for operability by first being subjected to rigid tests both prior to installation in the plant and after installation in the plant. The in-shop tests include: (1) hydrostatic tests of pressure-retaining parts of 125% of the design pressure; (2) seal leakage tests; and (3) performance tests while the pump is operated with flow to determine total developed head, minimum and maximum head and net positive suction head (NPSH) requirements. Also monitored during these operating tests are bearing temperatures (except water cooled bearings) and vibration levels. Both are shown to be below specified limits. After the pump is installed in the plant, it undergoes the cold hydro tests, functional tests, and the required periodic inservice inspection and operation. These tests demonstrate reliability of the pump for the design life of the plant.

In addition to these tests, these pumps are analyzed for operability during a faulted condition by assuring that (1) the pump will not be damaged during the dynamic (SSE and LOCA) event, and (2) the pump will continue operating despite the dynamic loads.

* See Section 3.10.

3.9.3.2.1.4 ECCS Motors

Qualification of the Class 1E motors used for the ECCS motors complies with IEEE-323. The qualification of all motor sizes is based on completion of a type test, followed up with review and comparison of design and material details, and seismic and other RBV loads analyses of production units, ranging from 447 to 2610 kW, with the motor used in the type test. All manufacturing, inspection, and routine tests by motor manufacturer on production units are performed on the test motor.

The type test is performed on a 932 kw vertical motor in accordance with IEEE-323, first simulating a normal operation during the design life, then subjecting the motor to a number of vibratory tests, and then to the abnormal environmental condition possible during and after a LOCA. The test plans for the type test are as follows:

(1) Thermal aging of the motor electrical insulation system (which is a part of the stator only) is based on extrapolation in accordance with the temperature life characteristic curve from IEEE-275 for the insulation type used on the ECCS motors. The amount of aging equals the total estimated operation days at maximum insulation surface temperature.

(2) Radiation aging of the motor electrical insulation equals the maximum estimated integrated dose of gamma during normal and abnormal conditions.

(3) The normal operational induced current vibration effect on the insulation system is simulated by 1.5g horizontal vibration acceleration at current frequency for one hour duration.

(4) The dynamic load deflection analysis on the rotor shaft is performed to ensure adequate rotation clearance, and is verified by static loading and deflection of the rotor for the type test motor.

(5) Dynamic load aging and testing is performed on a biaxial test table in accordance with IEEE-344. During this test, the shake table is activated to simulate the maximum design limit for the SSE and other RBV loads with as many motor starts and operation combinations consistent with the plant events of Table 3.9-1 and the ECCS inadvertent injections and tests planned over the life of the plant.

(6) An environmental test simulating a LOCA condition with a duration of 100 days is performed with the test motor fully loaded, simulating pump operation. The test consists of startup and six hours operation at 100^{o}C ambient temperature and 100% steam environment. Another startup and operation of the test motor after one hour standstill in the same environment is followed by sufficient operation at high humidity and temperature based on extrapolation in accordance with the temperature life characteristic curve from IEEE-275 for the insulation type used on the ECCS motors.

3.9.3.2.1.5 RCIC Turbine

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

These equipment assemblies are small, compact, rigid assemblies with natural frequencies well above 33 Hz. With this fact verified, each equipment assembly is qualified by the static analysis for seismic and other RBV loads. This qualification assures structural loading stresses within Code limitations, and verifies operability under seismic and other RBV loads (Subsections 3.9.2.2.2.8 and 3.9.2.2.2.7).

3.9.3.2.3 Other Active Pumps

The active pumps not previously discussed are ASME Class 2 or 3 and Seismic Category I. They are designed to perform their function including all required mechanical motions during and after a dynamic (seismic and other RBV) loads event and to remain operative during the life of the plant.

The program for the qualification of Seismic Category I components conservatively demonstrates that no loss of function results either before, during, or after the occurrence of the combination of events for which operability must be assured. No loss of function implies that the pressure boundary integrity will be maintained, that the component will not be caused to operate improperly, and that components required to respond actively will respond properly as appropriate to the specific equipment. In general, operability assurance is established during and after the dynamic loads event for active components.

3.9.3.2.3.1 Procedures

Procedures have been established for qualifying the mechanical portions of Seismic Category I pumps such as the body which forms a fluid pressure boundary, including the suction and discharge nozzles, shaft and seal retainers, impeller assembly (including blading, shaft, and bearings for active pumps), and integral supports.

All active pumps are qualified for operability by first being subjected to rigid tests both prior to installation and after installation in the plant. Electric motors for active pumps and

instrumentation, including electrical devices which must function to cause the pump to accomplish its intended function, are discussed separately in Subsection 3.9.3.2.5.1.3.

3.9.3.2.3.1.1 Hydrostatic Test

All seismic-active pumps shall meet the hydrostatic test requirements of ASME Code Section III according to the class rating of the given pump.

3.9.3.2.3.1.2 Leakage Test

The fluid pressure boundary is examined for leaks at all joints, connections, and regions of high stress such as around openings or thickness transition sections while the pump is undergoing a hydrostatic test or during performance testing. Leakage rates that exceed the rates permitted in the design specification are eliminated and the component retested to establish an observed leakage rate. The actual observed leakage rate, if less than permitted, is documented and made a part of the acceptable documentation package for the component.

3.9.3.2.3.1.3 Performance Test

The pump is demonstrated capable of meeting all hydraulic requirements while operating with flow at the total developed head, minimum and maximum head, NPSH, and other parameters as specified in the equipment specification.

Bearing temperature (except water cooled bearings) and vibration levels are also monitored during these operating tests. Both are shown to be below specified levels.

3.9.3.2.3.1.4 Dynamic Qualification

The safety-related active pumps are analyzed for operability during dynamic loading event by assuring that the pump is not damaged during the seismic event and the pump continues operating despite the dynamic loads.

A test or dynamic analysis is performed for a pump to determine the dynamic seismic and other RBV load from the applicable floor response spectra.

Response spectra for the horizontal vibration are used in two orthogonal horizontal directions simultaneously with the response spectra for the vertical vibration. The effects from the three simultaneous accelerations are combined by the square-root-of-the-sum-of-the-squares method. [*The pump is demonstrated by test or analysis that the faulted condition nozzle loads do not impair the operability of the pumps during or following the faulted condition.*]* Components of the pump are considered essentially rigid when having a natural frequency above 33 Hz. A static shaft deflection analysis of the motor rotor is performed with the conservative SSE accelerations acting in horizontal and vertical direction simultaneously.

* See Section 3.10.

The deflections determined from the static shaft analysis are compared to the allowable rotor clearances. The allowable rotor clearances are limited by the deflection which would cause the rotor to just make contact with the stator. In order to avoid damage during the faulted plant condition, the stresses caused by the combination of normal operating loads, SSE and dynamic system loads are limited to the material elastic limit.

[*The average membrane stress (σ_m) for the faulted conditions loads is limited to 1.2S or approximately 0.75 σ_y (σ_y = yield stress), and the maximum stress in local fibers (σm + bending stress σb) is limited to 1.8S or approximately 1.1 σ_y. The maximum dynamic nozzle loads are also considered in an analysis of the pump supports to assure that a system misalignment cannot occur.*]*

If the natural frequency is found to be below 33 Hz, an analysis is performed to determine the amplified input accelerations necessary to perform the static analysis.

In completing the seismic qualification procedures, the pump motor and all components vital to the operation of the pump are independently qualified for operation during the maximum seismic event by IEEE-344.

[*If the testing option is chosen, sine-beat testing for electrical equipment is performed by satisfying one or more of the following requirements to demonstrate that multi-frequency response is negligible or that the sine-beat input is of sufficient magnitude to conservatively account for this effect.*

(1) *The equipment response is basically due to one mode.*

(2) *The sine-beat response spectra envelope the floor response spectra in the region of significant response.*

(3) *The floor response spectra consist of one dominant mode and has a peak at this frequency.*]*

The degrees of cross coupling in the equipment shall determine if a single or multi-axis test is required. Multi-axis testing is required if there is considerable cross coupling. If coupling is very light, then single axis testing is justified. Or, if the degree of coupling can be determined, then single-axis testing can be used with input sufficiently increased to include the effect of coupling on the response of the equipment.

The combined stresses of the support structures are designed to be within the limits of ASME Code Section III, Subsection NF, component Support Structures and/or other comparable limits of industry standards such as the AISC Specification for Buildings, plus Addenda for building support structures.

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* See Section 3.10.

An analysis or test is accomplished which conservatively demonstrates structural integrity and/or functionality of the equipment supports.

The impeller, shaft, and bearings for active pumps are analyzed to determine adequacy while operating with the seismic and other RBV loading effects applied in addition to the applicable operating loads including nozzle loads. Functional requirements are partially demonstrated by a suitable analysis which conservatively shows the following:

(1) The stresses in the shaft do not exceed the minimum yield strength of the material used for its construction.

(2) The deflections of the shaft and/or impeller blades do not cause the impeller assembly to seize.

(3) The bearing temperature does not attain limits which may allow stresses in the bearing or bearing support to exceed minimum yield strength levels or jeopardize lubrication.

3.9.3.2.3.2 Documentation

All of the preceding requirements (Subsection 3.9.3.2.3.1) are satisfied to demonstrate that functionality is assured for active pumps. The documentation is prepared in a format that clearly shows that each consideration has been properly evaluated and tests have been validated by a designated quality assurance representative. The analysis is included as a part of the certified stress report for the assembly.

3.9.3.2.4 Major Active Valves

Some of the major safety-related active valves (Table 6.2-2, 6.2-3 and 3.2-1) discussed in this subsection for illustration are the MSIVs and SRVs, and SLC valves and HPCF valves (motor-operated). These valves are designed to meet ASME Code Section III requirements and perform their mechanical motion in conjunction with a dynamic (SSE and other RBV) load event. These valves are supported entirely by the piping
(i. e., the valve operators are not used as attachment points for piping supports)
(Subsection 3.9.3.4.1). The dynamic qualification for operability is unique for each valve type; therefore, each method of qualification is detailed individually below.

3.9.3.2.4.1 Main Steam Isolation Valve

The typical Y-pattern MSIVs described in Subsection 5.4.5.2 are evaluated by analysis and test for capability to operate under the design loads that envelop the predicted loads during a DBA and SSE.

The valve body is designed, analyzed and tested in accordance with ASME Code Section III Class 1 requirements. The MSIVs are modeled mathematically in the MS System analysis. The

loads, amplified accelerations and resonance frequencies of the valves are determined from the overall steamline analysis. The piping supports (snubbers, rigid restraints, etc.) are located and designed to limit amplified accelerations of and piping loads in the valves to the design limits.

As described in Subsection 5.4.5.3, the MSIV and associated electrical equipment (wiring, solenoid valves, and position switches) are dynamically qualified to operate during an accident condition.

3.9.3.2.4.2 Main Steam Safety/Relief Valve

The typical SRV design described in Subsection 5.2.2.4.1 is qualified by type test to IEEE-344 for operability during a dynamic event. Structural integrity of the configuration during a dynamic event is demonstrated by both Code (ASME Class 1) analysis and test.

(1) Valve is designed for maximum moments on inlet and outlet which may be imposed when installed in service. These moments are resultants due to dead weight plus dynamic loading of both valve and connecting pipe, thermal expansion of the connecting pipe, and reaction forces from valve discharge.

(2) A production SRV is demonstrated for operability during a dynamic qualification (shake table) type test with moment and “g” loads applied greater than the required equipment’s design limit loads and conditions.

A mathematical model of this valve is included in the MS System analysis, as with the MSIVs. This analysis assures that the equipment design limits are not exceeded.

3.9.3.2.4.3 Standby Liquid Control Valve (Injection Valve)

The typical SLC injection valve design is qualified by type test to IEEE-344. The valve body is designed, analyzed and tested per ASME Code Section III Class 1. The qualification test demonstrates the ability to remain operable after the application of the horizontal and vertical dynamic loading exceeding the predicted dynamic loading.

3.9.3.2.4.4 High Pressure Core Flooder Valve (Motor-Operated)

The typical HPCF valve body design, analysis and testing is in accordance with the requirements of the ASME Code Section III Class 1 or 2 components. The Class 1E electrical motor actuator is qualified by type test in accordance with IEEE-382, as discussed in Subsection 3.11.2. A mathematical model of this valve is included in the HPCF piping system analysis. The analysis results are assured not to exceed the horizontal and vertical dynamic acceleration limits acting simultaneously for a dynamic (SSE and other RBV) event, which is treated as an emergency condition. Subsection 3.9.3.2.5 discusses the operability qualification of the HPCF valve for seismic and other dynamic loads.

3.9.3.2.5 Other Active Valves

Other safety-related active valves are ASME Class 1, 2 or 3 and are designed to perform their mechanical motion during dynamic loading conditions. The operability assurance program ensures that these valves will operate during a dynamic seismic and other RBV event.

3.9.3.2.5.1 Procedures

Qualification tests accompanied by analyses are conducted for all active valves. Procedures for qualifying electrical and instrumentation components which are depended upon to cause the valve to accomplish its intended function are described in Subsection 3.9.3.2.5.1.3.

3.9.3.2.5.1.1 Tests

Prior to installation of the safety-related valves, the following tests are performed: (1) shell hydrostatic test to ASME Code Section III requirements; (2) back seat and main seat leakage tests; (3) disc hydrostatic test; (4) functional tests to verify that the valve will open and close within the specified time limits when subject to the design differential pressure; and (5) operability qualification of valve actuators for the environmental conditions over the installed life. Environmental qualification procedures for operation follow those specified in Section 3.11. The results of all required tests are properly documented and included as a part of the operability acceptance documentation package.

3.9.3.2.5.1.2 Dynamic Load Qualification

The functionality of an active valve during and after a seismic and other RBV event is demonstrated by test or by a combination of analysis and test. The qualification of electrical and instrumentation components controlling valve actuation is discussed in Subsection 3.9.3.2.5.1.3. The valves are designed using either stress analyses or the pressure temperature rating requirements based upon design conditions. A test or analysis of the extended structure is performed for the expected dynamic loads acting on the extended structure. See Subsection 3.9.2.2 for further details.

The maximum stress limits allowed in these analyses confirm structural integrity and are the limits developed and accepted by the ASME for the particular ASME Class of valve analyzed.

[*The stress limits for operability are provided in footnote 12 of Table 3.9-2.*]*

Dynamic load qualification is accomplished in the following way:

(1) All the active valves are typically designed to have a fundamental frequency which is greater than the high frequency asymptote (ZPA) of the dynamic event. This is shown by suitable test or analysis.

* See Section 3.10.

(2) The actuator and yoke of the valve system is statically loaded to an amount greater than that due to a dynamic event. The load is applied at the center of gravity of the actuator alone in the direction of the weakest axis of the yoke. The simulated operational differential pressure is simultaneously applied to the valve during the static deflection tests.

(3) The valve is then operated while in the deflected position (i.e., from the normal operating position to the safe position). The valve is verified to perform its safety-related function within the specified operating time limits.

(4) Motor operators and other electrical appurtenances necessary for operation are qualified as operable during a dynamic event by appropriate qualification tests prior to installation on the valve. Alternately, the valve including the motor operator and all other accessories is qualified by a shake table test.

The piping, stress analysis, and pipe support design maintain the motor operator accelerations below the qualification levels with adequate margin of safety.

If the fundamental frequency of the valve, by test or analysis, is less than that for the ZPA, a dynamic analysis of the valve is performed to determine the equivalent acceleration to be applied during the static test. The analysis provides the amplification of the input acceleration considering the natural frequency of the valve and the frequency content of the applicable plant floor response spectra. The adjusted accelerations have been determined using the same conservatism contained in the horizontal and vertical accelerations used for rigid valves. The adjusted acceleration is then used in the static analysis and the valve operability is assured by the methods outlined in Steps (2) through (4), using the modified acceleration input. Alternatively, the valve including the actuator and all other accessories is qualified by shake table test.

Valves which are safety-related but can be classified as not having an overhanging structure, such as check valves and pressure-relief valves, are considered as follows:

3.9.3.2.5.1.2.1 Active Check Valves

Due to the particular simple characteristics of the check valves, the active check valves are qualified by a combination of the following tests and analysis:

(1) Stress analyses including the dynamic loads where applicable

(2) In-shop hydrostatic tests

(3) In-shop seat leakage test

(4) Periodic in-situ valve exercising and inspection to assure the functional capability of the valve

3.9.3.2.5.1.2.2 Active Pressure-Relief Valves

The active pressure-relief valves (RVs) are qualified by the following procedures. These valves are subjected to test and analysis similar to check valves, stress analyses including the dynamic loads, in-shop hydrostatic seat leakage, and performance tests. In addition to these tests, periodic in-situ valve inspections, as applicable, and periodic valve removal, refurbishment, performance testing, and reinstallation are performed to assure the functional capability of the valve. Tests of the RV under dynamic loading conditions demonstrate that valve actuation can occur during application of the loads. The tests include pressurizing the valve inlet with nitrogen and subjecting the valve to accelerations equal to or greater than the dynamic event (SSE plus other RBV) loads.

3.9.3.2.5.1.2.3 Qualification of Electrical and Instrumentation Components Controlling Valve Actuation

A practical problem arises in attempting to describe tests for devices (relays, motors, sensors, etc.) as well as for complex assemblies such as control panels. It is reasonable to assume that a device, as an integral part of an assembly, can be subjected to dynamic loads tests while in an operating condition and its performance monitored during the test. However, in the case of complex panels, such a test is not always practical. In such a situation, the following alternate approach is recommended.

The individual devices are tested separately in an operating condition and the test levels recorded as the qualification levels of the devices. The panel, with similar devices installed but inoperative, is vibration tested to determine if the panel response accelerations, as measured by accelerometers installed at the device attachment locations, are less than the levels at which the devices were qualified. Note that the purpose of installing the nonoperating devices is to assure that the panel has the structural characteristics it will have when in use. If the acceleration levels at the device locations are found to be less than the levels to which the device is qualified, then the total assembly is considered qualified. Otherwise, either the panel is redesigned to reduce the acceleration level to the device locations and retested, or the devices is requalified to the higher levels.

3.9.3.2.5.2 Documentation

All of the preceding requirements (Subsection 3.9.3.2.5.1) are satisfied to demonstrate that functionality is assured for active valves. The documentation is prepared in a format that clearly shows that each consideration has been properly evaluated and tests have been validated by a designated quality assurance representative. The analysis is included as a part of the certified stress report for the assembly.

3.9.3.3 Design and Installation of Pressure Relief Devices

3.9.3.3.1 Main Steam Safety/Relief Valves

SRV lift in a main steam (MS) piping system results in a transient that produces momentary unbalanced forces acting on the MS and SRV discharge piping system for the period from opening of the SRV until a steady discharge flow from the reactor pressure vessel to the suppression pool is established. This period includes clearing of the water slug from the end of the discharge piping submerged in the suppression pool. Pressure waves traveling through the MS and discharge piping following the relatively rapid opening of the SRV cause this piping to vibrate.

The analysis of the MS and discharge piping transient due to SRV discharge consists of a stepwise time-history solution of the fluid flow equation to generate a time history of the fluid properties at numerous locations along the pipe. The fluid transient properties are calculated based on the maximum set pressure specified in the steam system specification and the value of ASME Code flow rating increased by a factor to account for the conservative method of establishing the rating. As a conservative approach, it is assumed that all SRVs mounted on a MS line actuate simultaneously. Simultaneous actuation of all SRVs is considered to induce maximum stress in the MS piping. Further, a subsequent actuation condition rather than initial actuation for all SRVs is conservatively assumed. This is a conservative approach, considering that all SRVs will not actuate simultaneously with subsequent actuation condition in the SRV piping, because individual SRVs have different relief set pressure values. These features should preclude simultaneous subsequent actuation of all SRVs. The methodology for calculating hydrodynamic loading on SRV discharge piping due to subsequent SRV actuations is consistent with previously approved methodology for earlier BWR (Mark II/III) designs.The effect of subsequent valve actuation is included by assuming hot SRV discharge pipe condition before valve actuation which results in higher loads on the piping. SRV loads are calculated assuming initial SRV pipe metal temperature to be 149°C for the pipe in the drywell region and 93°C for the pipe in the wetwell region, consistent with that used for earlier BWRs. These temperature values are based on measured data from in-plant SRV blowdown tests. Reaction loads on the pipe are determined at each location corresponding to the position of an elbow. These loads are composed of pressure-times-area, momentum-change, and fluid-friction terms.

The method of analysis applied to determine response of the MS piping system, including the SRV discharge line, to relief valve operation is time-history integration. The forces are applied at locations on the piping system where fluid flow changes direction, thus causing momentary reactions. The resulting loads on the SRV, the main steamline, and the discharge piping are combined with loads due to other effects as specified in Subsection 3.9.3.1. In accordance with Tables 3.9-1 and 3.9-2, the Code stress limits for service levels corresponding to load combination classification as normal, upset, emergency, and faulted are applied to the main steam and discharge pipe.

3.9.3.3.2 Other Safety/Relief Valves

An SRV is identified as a pressure relief valve or vacuum breaker. SRVs in the reactor components and subsystems are described and identified in Subsection 5.4.13.

The operability assurance program discussed in Subsection 3.9.3.2.5 applies to safety/relief valves. The qualification of active relief valves is specifically outlined in Subsection 3.9.3.2.5.1.2.2.

ABWR SRVs (safety valves with auxiliary actuating devices and pilot operated valves) are designed and manufactured in accordance with ASME Code Section III Division 1 requirements. Specific rules for pressure relieving devices are as specified in Article NB-7000 and NB-3500 (pilot-operated and power-actuated pressure relief valves).

The design of ABWR SRVs incorporates SRV opening and pipe reaction load considerations required by ASME Section III, Appendix O, including the additional criteria of SRP Section 3.9.3, Paragraph II.2 and those identified under Subsection NB-3658 for pressure and structural integrity. SRV operability is demonstrated either by dynamic testing or analysis of similarly tested valves or a combination of both in compliance with the requirements of SRP Subsection 3.9.3.

3.9.3.3.3 Rupture Disks

There are no rupture disks in the ABWR plant design that must function during and after a dynamic event (SSE including other RBV loads) at design basis conditions. However, the rupture disk in the containment overpressure protection system may operate following severe accident seismic conditions.

3.9.3.4 Component Supports

The design of bolts for component supports is specified in ASME Code Section III, Subsection NF. Stress limits for bolts are given in NF-3225. The rules and stress limits which must be satisfied are those given in NF-3324.6 multiplied by the appropriate stress limit factor for the particular service loading level and stress category specified in Table NF-3225.2-1.

Moreover, on equipment which is to be, or may be, mounted on a concrete support, sufficient holes for anchor bolts are provided to limit the anchor bolt stress to less than 68.6 MPa on the nominal bolt area in shear or tension.

Concrete anchor bolts (including under-cut type anchor bolts) which are used for pipe support base plates will be designed to the applicable factors of safety defined in IE Bulletin 79-02, “Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts”, Revision 2, November 8, 1979.[*Justification shall be provided for the use of safety factors for concrete anchor bolts other than those specified in IE Bulletin 79-02. This justification shall be submitted to the NRC staff for review and approval prior to the installation of the concrete*

anchor bolts. Pipe support base plate flexibilities are accounted for in the calculation of concrete anchor bolt loads, in accordance with IE Bulletin 79-02.]†

3.9.3.4.1 Piping

[*Supports and their attachments for essential ASME Code Section III, Class 1,2, and 3 piping are designed in accordance with Subsection NF* * *up to the interface of the building structure, with jurisdictional boundaries as defined by Subsection NF.*]† The loading combinations for the various operating conditions correspond to those used for design of the supported pipe. The component loading combinations are discussed in Subsection 3.9.3.1. The stress limits are per ASME III Code Section, Subsection NF and Appendix F. Supports are generally designed either by load rating method per Code Paragraph NF-3280 or by the stress limits for linear supports per paragraph NF-3143. The critical buckling loads for the Class 1 piping supports subjected to faulted loads, which are more severe than normal, upset and emergency loads, are determined by using the methods discussed in Appendices F and XVII of the Code. To avoid buckling in the piping supports, the allowable loads are limited to two-thirds of the determined critical buckling loads.

[*Maximum calculated static and dynamic deflections at support locations are checked to confirm that the support has not rotated beyond the vendor's recommended cone of action or the recommended arc of loading.*]†

[*Supports for ASME Code Section III instrumentation lines are designed and analyzed in accordance with ASME Code Section III, Subsection NF.*]†

The design of all supports for non-nuclear piping satisfies the requirements of ANSI/ASME B31.1, Power Piping Code, Paragraphs 120 and 121.

For the major active valves identified in Subsection 3.9.3.2.4, the valve operators are not used as attachment points for piping supports.

The design criteria and dynamic testing requirements for the ASME III Subsection NF piping supports are as follows:

(1) **Piping Supports—**All piping supports are designed, fabricated, and assembled so that they cannot become disengaged by the movement of the supported pipe or equipment after they have been installed.[*All piping supports are designed in*

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* [*Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet "Standard for Steel Support Design", analyses will be performed in accordance with torsional analysis methods such as: "Torsional Analysis of Steel Members, USS Steel Manual", Publication T114-2/83.*]†

† See Subsection 3.9.1.7.

accordance with the rules of Subsection NF of the ASME Code up to the building structure interface as defined by the jurisdictional boundaries in Subsection NF.]*

(2) **Spring Hangers—**The operating load on spring hangers is the load caused by dead weight. The hangers are calibrated to ensure that they support the operating load at both their hot and cold load settings. Spring hangers provide a specified down travel and up travel in excess of the specified thermal movement. Deflections due to dynamic loads are checked to confirm that they do not bottom out.

(3) **Snubbers—[***Mechanical and hydraulic type snubbers will be used when required as shock arrestors for nuclear safety-related piping systems. Snubbers are designed in accordance with ASME Section III, Subsection NF, Component Standard Supports.*]* Snubbers consist of a velocity-limiting or acceleration-limiting cylinder pinned to a pipe clamp at the pipe end and pinned to a clevis attached to the building structure at the other end. Snubbers operate as structural supports during dynamic events such as an earthquake, but during normal operation act as passive devices which accommodate normal expansions and contractions without resistance. The operating loads on snubbers are the loads caused by dynamic events (e.g., seismic, RBV due to LOCA and SRV discharge, discharge through a relief valve line or valve closure) during various operating conditions. Snubbers restrain piping against response to the vibratory excitation and to the associated differential movement of the piping system support anchor points. The criteria for locating snubbers and ensuring adequate load capacity, the structural and mechanical performance parameters used for snubbers and the installation and inspection considerations for the snubbers are as follows:

(a) Required Load Capacity and Snubber Location

[*The loads calculated in the piping dynamic analysis, described in Subsection 3.7.3.8, cannot exceed the snubber load capacity for design, normal, upset, emergency and faulted conditions.*

For hydraulic snubbers with load ratings greater than 222.4 kN, dynamic cyclic load tests will be conducted to verify the performance of the control valve. These hydraulic snubbers will be subjected to dynamic cyclic load tests at loads greater than or equal to one-half the calculated safe shutdown earthquake load on the snubbers.]†

Snubbers are generally used in situations where dynamic support is required because thermal growth of the piping prohibits the use of rigid supports. The snubber locations and support directions are first decided by estimation so that the stresses in the piping system will have acceptable values. The snubber locations and support directions are refined by performing the dynamic

* See Subsection 3.9.1.7.

† See Subsection 3.9.1.7.

analysis of the piping and support system as described above in order that the piping stresses and support loads meet the Code requirements.

[*The pipe support design specification requires that snubbers be provided with position indicators to identify the rod position.*]* This indicator facilitates the checking of hot and cold settings of the snubber, as specified in the installation manual, during plant preoperational and startup testing.

(b) I*nspection, Testing, Repair and/or Replacement of Snubbers*

[The pipe support design specification requires that the snubber supplier prepare an installation instruction manual. This manual is required to contain complete instructions for the testing, maintenance, and repair of the snubber. It also contains inspection points and the period of inspection.

The pipe support design specification requires that hydraulic snubbers be equipped with a fluid level indicator so that the level of fluid in the snubber can be ascertained easily.

The spring constant achieved by the snubber supplier for a given load capacity snubber is compared against the spring constant used in the piping system model. If the spring constants are the same, then the snubber location and support direction become confirmed. If the spring constants are not in agreement, they are brought in agreement, and the system analysis is redone to confirm the snubber loads. This iteration is continued until all snubber load capacities and spring constants are reconciled.]*

(c) Snubber Design and Testing

To assure that the required structural and mechanical performance characteristics and product quality are achieved, the following requirements for design and testing are imposed by the design specification:

(i) [*The snubbers are required by the pipe support design specification to be designed in accordance with all of the rules and regulations of ASME Code Section III, Subsection NF. This design requirement includes analysis for the normal, upset, emergency, and faulted loads. These calculated loads are then compared against the allowable loads to make sure that the stresses are below the code allowable limit.*

(ii) The snubbers are tested to insure that they can perform as required during the seismic and other RBV events, and under anticipated operational transient loads or other mechanical loads associated with the design requirements for the plant. The following test requirements are included:

— *Snubbers are subjected to force or displacement versus time loading at frequencies within the range of significant modes of the piping system.*

— *Dynamic cyclic load tests are conducted for large bore hydraulic snubbers to determine the operational characteristics of the snubber control valve.*

— *Displacements are measured to determine the performance characteristics specified.*

— *Tests are conducted at various temperatures to ensure operability over the specified range.*

— *Peak test loads in both tension and compression are required to be equal to or higher than the rated load requirements.*

— *The snubbers are tested for various abnormal environmental conditions. Upon completion of the abnormal environmental transient test, the snubber is tested dynamically at a frequency within a specified frequency range. The snubber must operate normally during the dynamic test.*]*

(d) Snubber Installation Requirements

[*An installation instruction manual is required by the pipe support design specification. This manual is required to contain instructions for storage, handling, erection, and adjustments (if necessary) of snubbers. Each snubber has an installation location drawing which contains the installation location of the snubber on the pipe and structure, the hot and cold settings, and additional information needed to install the particular snubber.*]†

(e) Snubber Pre-service Examination

[*The pre-service examination plan of all snubbers covered by the Chapter 16 technical specifications will be prepared. This examination will be made after snubber installation but not more than 6 months prior to initial system pre-operational testing. The pre-service examination will verify the following:*

(i) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

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* See Subsection 3.9.1.7.

† See Subsection 3.9.1.7.

(ii) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifications.

(iii) Snubbers are not seized, frozen or jammed.

(iv) Adequate swing clearance is provided to allow snubber movements.

(v) If applicable, fluid is to be at recommended level and not leaking from the snubber system.

(vi) Structural connections such as pins, fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational tests exceeds 6 months because of unexpected situations, re-examination of Items i, iv, and v will be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements will be repaired or replaced and re-examined in accordance with the above criteria.]*

(4) **Struts—**Struts are defined as ASME Section III, Subsection NF, Component Standard Supports. They consist of rigid rods pinned to a pipe clamp or lug at the pipe and pinned to a clevis attached to the building structure or supplemental steel at the other end. Struts, including the rod, clamps, clevises, and pins are designed in accordance with ASME Code Section III, Subsection NF-3000.

Struts are passive supports, requiring little maintenance and inservice inspection, and will normally be used instead of snubbers where dynamic supports are required and the movement of the pipe due to thermal expansion and/or anchor motions is small. Struts will not be used at locations where restraint of pipe movement to thermal expansion will significantly increase the secondary piping stress ranges or equipment nozzle loads. Increases of thermal expansion loads in the pipe and nozzles will normally be restricted to less than 20%.

Because of the pinned connections at the pipe and structure, struts carry axial loads only. The design loads on struts may include those loads caused by thermal expansion, dead weight, and the inertia and anchor motion effects of all dynamic loads. As in the case of other supports, the forces on struts (obtained from an analysis) do not exceed the design loads for various operating conditions.

(5) **Frame Type (Linear) Pipe Supports—**Pipe Supports: Frame type supports are linear supports as defined as ASME Section III, Subsection NF, Component Standard Supports. They consist of frames constructed of structural steel elements that are not attached to the pipes. They act as guides to allow axial and rotational movement of

the pipe, but also as rigid restraints to lateral movement in either one or two directions. [*Frame type pipe supports are designed in accordance with ASME Code Section III, Subsection NF-3000.*]*

Frame type supports are passive supports, requiring little maintenance and inservice inspection, and will normally be used instead of struts when they are more economical or where environmental conditions are not suitable for the ball bushings at the pinned connections of struts. Similar to struts, frame type supports will not be used at locations where restraint of pipe movement to thermal expansion will significantly increase the secondary piping stress ranges or equipment nozzle loads. Increases of thermal expansion loads in the pipe and nozzles will normally be restricted to less than 20%.

The design loads on frame type pipe supports include those loads caused by thermal expansion, dead weight, and the inertia and anchor motion effects of all dynamic loads. As in the case of other supports, the forces on frame type supports are obtained from an analysis and assured not to exceed the design loads for various operating conditions.

(6) **Special Engineered Pipe Supports**—In an effort to minimize the use and application of snubbers, there may be instances where special engineered pipe supports can be used where either struts or frame-type supports cannot be applied. Examples of special engineered supports are Energy Absorbers and Limit Stops.

(a) **Energy Absorbers** are linear energy absorbing support parts designed to dissipate energy associated with dynamic pipe movements by yielding. [*When energy absorbers are used they will be designed to meet the requirements of ASME Section III, Code Case N-420, Linear Energy Supports for Subsection NF, Classes 1, 2, and 3 Construction, Section III, Division 1.The information required by Regulatory Guide 1.84 will be provided to the regulatory agency.*]† The restrictions on location and application of struts and frame-type supports, discussed in (4) and (5) above, are also applicable to energy absorbers since energy absorbers allow thermal movement of the pipe only in its design directions.

(b) **Limit Stops** are passive seismic pipe support devices consisting of limit stops with gaps sized to allow for thermal expansion while preventing large seismic displacements.[*Limit stops are linear supports as defined as ASME Section III, Subsection NF, and are designed in accordance with ASME Code Section III, Subsection NF-3000.*]* They consist of either special component standard supports with a configuration, size and end to end dimensions similar to

* See Subsection 3.9.1.7.

† See Subsection 3.9.1.7.

snubbers, or box frames constructed of structural steel elements that are not attached to the pipe. The box frames allow free movement in the axial direction but limits large displacement in the lateral direction.

If these special devices are used, the modeling and analytical methodology will be in accordance with the methodology accepted by the regulatory agency at the time of the certification or at the time of application, per the direction of the COL applicant.

3.9.3.4.2 Reactor Pressure Vessel Support Skirt

The ABWR RPV support skirt is designed as an ASME Code Class 1 component per the requirements of ASME Code Section III, Subsection NF*. The loading conditions and stress criteria are given in Tables 3.9-1 and 3.9-2, and the calculated stresses meet the Code allowable stresses in the critical support areas for various plant operating conditions. The stress level margins assure the adequacy of the RPV support skirt. An analysis for buckling shows that the support skirt complies with Subparagraph F-1332.5 of ASME III, Appendix F, and the loads do not exceed two-thirds of the critical buckling strength of the skirt. The permissible skirt loads at any elevation, when simultaneously applied, are limited by the following interaction equation:

$$\left(\frac{P}{P_{crit}}\right) + \left(\frac{q}{q_{crit}}\right) + \left(\frac{\tau}{\tau_{crit}}\right) < \left(\frac{1}{S.F.}\right) \qquad (3.9\text{-}1)$$

where:

q	=	Longitudinal load
P	=	External pressure
τ	=	Transverse shear stress

* Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet "Standard for Steel Support Design", analyses will be performed in accordance with torsional analysis methods such as: "Torsional Analysis of Steel Members, USS Steel Manual", Publication T114-2/83.

S.F. = Safety factor

= 3.0 for design, testing, service levels A & B

= 2.0 for Service Level C

= 1.5 for Service Level D.

3.9.3.4.3 Reactor Pressure Vessel Stabilizer

The RPV stabilizer is designed as a Safety Class 1 linear type component support in accordance with the requirements of ASME Code Section III, Subsection NF. The stabilizer provides a reaction point near the upper end of the RPV to resist horizontal loads due to effects such as earthquake, pipe rupture and RBV. The design loading conditions and stress criteria listed in Tables 3.9-1 and 3.9-2 show that calculated stresses meet the Code allowable stresses in the critical support areas for various plant operating conditions.

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

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,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.

(2) Simulate all static and dynamic loadings on the pump.

(3) Monitor pump operability during testing.

(4) The normal operation of the pump during and after the test indicates that the supports are adequate (any deflection or deformation of the pump supports which precludes the operability of the pump is not accepted).

(5) Supports are inspected for structural integrity after the test. Any cracking or permanent deformation is not accepted.

Dynamic qualification of component supports by analysis is generally accomplished as follows:

(1) Stresses at all support elements and parts such as pump holddown and baseplate holddown bolts, pump support pads, pump pedestal, and foundation are checked to

be within the allowable limits as specified in ASME Code Section III, Subsection NF.

(2) For normal and upset conditions, the deflections and deformations of the supports are assured to be within the elastic limits, and to not exceed the values permitted by the designer based on design verification tests. This ensures the operability of the pump.

(3) For emergency and faulted plant conditions, the deformations do not exceed the values permitted by the designer to ensure the operability of the pump. Elastic/plastic analyses are performed if the deflections are above the elastic limits.

3.9.3.5 Other ASME III Component Supports

The ASME III component supports and their attachments (other than those discussed in preceding subsection) are designed in accordance with Subsection NF of ASME Code Section III* up to the interface with the building structure. The building structure component supports are designed in accordance with the AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings. The loading combinations for the various operating conditions correspond to those used to design the supported component. The component loading combinations are discussed in Subsection 3.9.3.1. Active component supports are discussed in Subsection 3.9.3.2. The stress limits are per ASME Code Section III, Subsection NF and Appendix F. The supports are evaluated for buckling in accordance with ASME Code Section III.

3.9.4 Control Rod Drive (CRD)

The Control Rod Drive (CRD) System is equipped with electro-hydraulic fine motion control rod drives (FMCRD), hydraulic control units (HCU), the condensate supply system, and power for FMCRD motors. The system extends inside the RPV to the coupling interface with the control rod blades.

3.9.4.1 Descriptive Information on CRD System

Descriptive information on the FMCRDs, as well as the entire CRD System, is contained in Section 4.6.

* Augmented by the following: (1) application of Code Case N-476, Supplement 89.1 which governs the design of single angle members of ASME Class 1,2,3 and MC linear component supports; and (2) when eccentric loads or other torsional loads are not accommodated by designing the load to act through the shear center or meet “Standard for Steel Support Design”, analyses will be performed in accordance with torsional analysis methods such as: “Torsional Analysis of Steel Members, USS Steel Manual”, Publication T114-2/83.

3.9.4.2 Applicable CRD System Design Specification

The CRD System is designed to meet the functional design criteria outlined in Section 4.6 and consists of the following:

(1) Fine motion control rod drive

(2) Hydraulic control unit

(3) Hydraulic power supply (pumps)

(4) Electric power supply (for FMCRD motors)

(5) Interconnecting piping

(6) Flow and pressure valves

(7) Instrumentation and electrical controls

Those components of the CRD System forming part of the primary pressure boundary are designed according to ASME Code Section III Class 1 requirements.

The quality group classification of the components of the CRD System is outlined in Table 3.2-1 and they are designed to the codes and standards, per Table 3.2-2, in accordance with their individual quality groups.

Pertinent aspects of the design and qualification of the CRD System components are discussed in the following locations: transients in Subsection 3.9.1.1, faulted conditions in Subsection 3.9.1.4, seismic testing in Subsection 3.9.2.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

The ASME III Code components of the CRD System are evaluated analytically and the design loading conditions, and stress criteria are as given in Tables 3.9-1 and 3.9-2. For the non-Code components, the ASME III Code requirements are used as guidelines and experimental testing is used to determine the CRD performance under all possible conditions as described in Subsection 3.9.4.4.

3.9.4.4 CRD Performance Assurance Program

The following CRD tests are described in Subsection 4.6.1.

(1) Development tests

(2) Factory quality control tests

(3) Functional tests

(4) Operational tests

(5) Acceptance tests

(6) Surveillance tests

3.9.5 Reactor Pressure Vessel Internals

This subsection identifies and discusses the structural and functional integrity of the major reactor pressure vessel (RPV) internals, including core support structures.

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.

3.9.5.1 Design Arrangements

The core support structures and reactor vessel internals (exclusive of fuel, control rods, and incore nuclear instrumentation) are:

(1) Core Support Structures

Shroud

Shroud support (including the internal pump deck)

Core plate (and core plate hardware)

Top guide

Fuel supports (orificed fuel supports and peripheral fuel supports)

Control rod guide tubes

Non-pressure boundary portion of control rod drive housings

(2) Reactor Internals

Shroud head* and steam separators assembly*

Steam dryers assembly*

Feedwater spargers

* These are non-nuclear safety category components as defined in Subsection 3.2.5.1. In Subsection 3.9.5, such components are called non-safety class components, and the safety-related internals (Safety Class 2 or 3) are called safety class components.

RHR/ECCS low pressure flooder spargers

ECCS high pressure core flooder spargers and piping

Core differential pressure lines

RPV vent and head spray assembly

Internal pump differential pressure lines*

Incore guide tubes and stabilizers

Non-pressure boundary portion of in-core housings

Surveillance sample holders*

A general assembly drawing of the important reactor components is shown in Figures 5.3-2a and 5.3-2b and Table 5.3-2.

The floodable inner volume of the reactor pressure vessel can be seen in Figure 3.9-2. It is the volume up to the level of the core flooder sparger.

The design arrangement of the reactor internals, such as the shroud, steam separators and guide tubes, is such that one end is unrestricted and thus free to expand.

The ECCS core flooder couplings incorporate vertically-oriented slip-fit joints to allow free thermal expansion.

3.9.5.1.1 Core Support Structures

The core support structures consist of those items listed in Subsection 3.9.5.1(1) and are Safety Class 3 as defined in Section 3.2. These structures form partitions within the reactor vessel to sustain pressure differentials across the partitions, direct the flow of the coolant water, and laterally locate and support the fuel assemblies. Figures 3.9-2 and 3.9-3 show the reactor vessel internal flow paths.

3.9.5.1.1.1 Shroud

The shroud support, shroud, and top guide make up a stainless steel 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 volume enclosed by this assembly is characterized by three regions. The upper portion surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide plate below. The central portion of the shroud surrounds the active fuel and forms the longest section of the assembly.

This section is bounded at the top by the top guide plate and at the bottom by the core plate. The lower portion, surrounding part of the lower plenum, is welded to the RPV shroud support. The shroud provides the horizontal support for the core by supporting the core plate and top guide.

3.9.5.1.1.2 Shroud Support

The RPV shroud support is designed to support the shroud, and includes the internal pump deck that locates and supports the pumps. The pump discharge diffusers penetrate the deck to introduce the coolant to the inlet plenum below the core. The RPV shroud support is a horizontal structure welded to the vessel wall to provide support to the shroud, pump diffusers, and core and pump deck differential pressure lines. The structure is a ring plate welded to the vessel wall and to a vertical cylinder supported by vertical stilt legs from the bottom head.

3.9.5.1.1.3 Core Plate

The core plate consists of a circular stainless steel plate with round openings and is stiffened with a rim and beam structure. The core plate provides lateral support and guidance for the control rod guide tubes, incore flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The last two items are also supported vertically by the core plate.

The entire assembly is bolted to a support ledge in the lower portion of the shroud.

3.9.5.1.1.4 Top Guide

The top guide consists of a circular plate with square openings for fuel with a cylindrical side forming an upper shroud extension and having a top flange for attaching the shroud head. 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 of the support intersections to anchor the incore flux monitors and startup neutron sources. The top guide is mechanically attached to the top of the shroud.

3.9.5.1.1.5 Fuel Supports

The fuel supports (Figure 3.9-4) are of two basic types: peripheral supports and orificed fuel supports. 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 fuel assembly and has an orifice designed to assure proper coolant flow to the peripheral fuel assembly. Each orificed fuel support holds four fuel assemblies vertically upward and horizontally and has four orifices to provide proper coolant flow distribution to each rod-controlled fuel assembly. The orificed fuel supports rest on the top of the CRGTs, which are supported laterally by the core plate. The control rods pass through cruciform openings in the center of the orificed fuel support. This locates the four fuel assemblies surrounding a control rod. A control rod and the four adjacent fuel assemblies represent a core cell (Section 4.4).

3.9.5.1.1.6 Control Rod Guide Tubes

The control rod guide tubes (CRGTs) located inside the vessel extend from the top of the CRD housings up through holes in the core plate. Each guide tube is designed as the guide for the lower end of a control rod and as the support for an orificed fuel support. This locates the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the CRD housing, which, in turn, transmits the weight of the guide tube, fuel support, and fuel assemblies to the reactor vessel bottom head. The control rod guide tubes (CRGTs) also contain holes, near the top of the control rod guide tube and below the core plate, for coolant flow to the orificed fuel supports.

3.9.5.1.2 Reactor Internals

The reactor internals consist of those items listed in Subsection 3.9.5.1(2). These components direct and control coolant flow through the core or support safety-related and non-safety-related function.

3.9.5.1.2.1 Shroud Head and Steam Separators Assembly

The shroud head and standpipes/steam separators are non-safety class internal components. The assembly is discussed here to describe the coolant flow paths in the reactor pressure vessel. The shroud head and steam separators assembly includes the upper flanges and bolts, and forms the top of the core discharge mixture plenum together with the separators and their connecting standpipes. The discharge plenum provides a mixing chamber for the steam/water mixture before it enters the steam separators. Individual stainless steel axial flow steam separators are supported on and attached to the top of standpipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam/water mixture rising through the standpipe passes vanes that impart a spin to establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the downcomer annulus. The assembly is removable from the reactor pressure vessel as a single unit on a routine basis.

3.9.5.1.2.2 Reactor Internal Pump (RIP)/Diffusers

The pump assembly (impeller, diffuser and pump shaft) is a non-safety class component and is discussed here to describe coolant flow paths (Figure 3.9-3) in the vessel. The pump provides a means for forced circulation of the reactor coolant through the core, including the mixing of feedwater and annulus water from the steam separators and distribution of this fluid to the vessel lower plenum and up through the lower grid to the core.

The pump assemblies are mounted vertically into pump nozzles arranged in an equally-spaced ring pattern on the bottom head of the RPV and are located inside the downcomer annulus between the core shroud and the reactor vessel wall. The design and performance of the pump assemblies is covered in detail in Subsection 5.4.1. Each pump consists of three major hardware

sections: an internal pump (IP) section; a recirculation motor (RM) section; and a stretch tube section (Figure 5.4-1).

The IP section of the RIP is located inside the RPV, in an opening through the RPV pump deck—the latter being the horizontal ring-plate enclosing the bottom of the downcomer annulus and thus separating the lower pressure annulus region from the higher-pressure lower plenum region. The IP, in turn, is comprised of a vertical axis single-stage, mixed-flow impeller driven from underneath by a pump shaft, with the impeller being encircled by a diffuser assembled into the pump deck opening.

The RM section of the RIP is located underneath, and at the periphery of, the RPV bottom head inside a pressure-retaining housing termed the motor casing. The motor casing itself is not part of the RM, but is instead a part of and welded into an RPV nozzle (pump nozzle). The motor casing thus comprises part of the RCPB and is a Safety Class 1 component.

The principal element of the stretch tube section is a thin-walled tube configured as a hollow bolt fitting around the pump shaft and within the pump nozzle. It has an external lip (bolt head) at its upper end and an external threaded section at this lower end. A stretch tube provides tight clamping of the IP diffuser to the gasketed, internal-mount end of the RPV pump nozzle, for all extremes of thermal transients and pump operating conditions.

3.9.5.1.2.3 Steam Dryer Assembly

The steam dryer assembly is a non-safety class component. It is discussed here to describe coolant flow paths in the vessel. The steam dryer 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.

The steam dryer assembly consists of multiple banks of dryer units mounted on a common structure which is removable from the RPV as an integral unit. The assembly includes the dryer banks, dryer supply and discharge ducting, drain collecting trough, drain piping, and a skirt which forms a water seal extending below the separator reference zero elevation. Upward and radial movement of the dryer assembly under the action of blowdown and seismic loads are limited by reactor vessel internal stops which are arranged to permit differential expansion growth of the dryer assembly with respect to the reactor pressure vessel. The assembly is arranged for removal from the vessel as an integral unit on a routine basis.

3.9.5.1.2.4 Feedwater Spargers

These are Safety Class 2 components. They are discussed here to describe coolant flow paths in the vessel and their safety function. Each of two feedwater lines is connected to three spargers via three RPV nozzles. One line is utilized by the RCIC System, the other by the RHR shutdown cooling system. During the ECCS mode, the two groups of spargers support diverse type of flooding of the vessel. The RCIC System side supports high pressure flooding and the RHR System side supports low pressure flooding, as required during the ECCS operation.

The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate sparger in two halves is fitted to each feedwater nozzle via a tee and is shaped to conform to the curve of the vessel wall. The sparger tee inlet is connected to the RPV nozzle safe end by a double thermal sleeve arrangement, with all connections made by full penetration welds. Sparger end brackets are pinned to vessel brackets to support the spargers. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The feedwater also serves to condense steam in the region above the downcomer annulus and to subcool water flowing to the recirculation internal pumps.

3.9.5.1.2.5 RHR/ECCS Low Pressure Flooder Spargers

These are Safety Class 2 components. The design features of these two spargers of the RHR shutdown cooling system are similar to those of the six feedwater spargers, three of which belonging to one feedwater line support additionally the same RHR (and ECCS) function. During the ECCS mode, these spargers support low pressure flooding of the vessel. The feedwater spargers are described in Subsection 3.9.5.1.2.4.

Two lines of the RHR shutdown cooling system enter the reactor vessel through the two diagonally opposite nozzles and connect to the spargers. The sparger tee inlet is connected to the RPV nozzle safe end by a thermal sleeve arrangement with all connections made by full penetration welds.

3.9.5.1.2.6 ECCS High Pressure Core Flooder Spargers and Piping

The core flooder spargers and piping are Safety Class 2. The spargers and piping are the means for directing high pressure ECCS flow to the upper end of the core during accident conditions.

Each of two High Pressure Core Flooder (HPCF) System lines enters the reactor vessel through a diagonally opposite nozzle in the same manner as an RHR low pressure flooder line, except that the curved sparger including the connecting tee is routed around the inside of and is supported by the cylindrical portion of the top guide. A flexible coupling is interposed between the sparger tee inlet and the sleeved inlet connector inside the nozzle. The two spargers are supported so as to accommodate thermal expansion.

3.9.5.1.2.7 RPV Vent and Head Spray Assembly

This is designed as a Safety Class 1 component. However, only the nozzle portion of the assembly is a reactor coolant pressure boundary, and the assembly function is not a safety-related operation. The reactor water cleanup return flow to the reactor vessel, via feedwater lines, can be diverted partly to a spray nozzle in the reactor head in preparation for refueling cooldown. The spray maintains saturated conditions in the reactor vessel head volume by condensing steam being generated by the hot reactor vessel walls and internals. The head spray subsystem is designed to rapidly cool down the reactor vessel head flange region for refueling and to allow installation of steamline plugs before vessel floodup for refueling.

The head vent side of the assembly passes steam and noncondensable gases from the reactor head to the steamlines during startup and operation. During shutdown and filling for hydrotesting, steam and noncondensable gases may be vented to the drywell equipment sump while the connection to the steamline is blocked. When draining the vessel during shutdown, air enters the vessel through the vent.

3.9.5.1.2.8 Core and Internal Pump Differential Pressure Lines

These lines comprise the core flow measurement subsystem of the Recirculation Flow Control System (RFCS) and provide two methods of measuring the ABWR core flow rates. The core DP lines (Safety Class 3) and internal pump DP lines (non-safety class) enter the reactor vessel separately through reactor bottom head penetrations. Four pairs of the core DP lines enter the head in four quadrants through four penetrations and terminate immediately above and below the core plate to sense the pressure in the region outside the bottom of the fuel assemblies and below the core plate during normal operation.

Similarly, four pairs of the internal pump DP lines terminate above and below the pump deck and are used to sense the pressure across the pump during normal pump operation. Each pair is routed concentrically through a penetration and upward along a shroud support leg in the lower plenum.

3.9.5.1.2.9 Incore Guide Tubes and Stabilizers

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 withdrawal 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 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.

3.9.5.1.2.10 Surveillance Sample Holders

This is a non-safety class component. The surveillance sample holders are welded baskets containing impact and tensile specimen capsules. The holders have brackets that are attached to the inside of the reactor vessel wall and are located in the active core beltline region. The radial and azimuthal positions are chosen to expose the specimens to the same environment and the maximum neutron fluxes experienced by the reactor vessel wall.

3.9.5.2 Loading Conditions

3.9.5.2.1 Events to be Evaluated

Examination of the spectrum of conditions for which the safety design bases (Subsection 3.9.5.3.1) must be satisfied by core support structures and safety-related internal components reveals four significant faulted events:

(1) **Feedwater Line Break—**A break in a feedwater line between the reactor vessel and the primary containment penetration; (the accident results in significant annulus pressurization and reactor building vibration (RBV) due to suppression pool dynamics).

(2) **Steamline Break Accident—**A break in one main steamline between the reactor vessel nozzle and the main steam isolation valve (the accident results in significant pressure differentials across some of the structures within the reactor and reactor building vibration due to suppression pool dynamics).

(3) **Earthquake—**subjects the core support structures and reactor internals to significant forces as a result of ground motion and consequent RBV.

(4) **Safety/relief valve discharge—**RBV due to suppression pool dynamics and structural feedback.

Analysis of other conditions existing during normal operation, abnormal operational transients, and accidents show that the loads affecting core support structures and other safety-related reactor internals are less severe than those affected by the four postulated events.

The faulted conditions for the reactor pressure vessel internals are discussed in Subsection 3.9.1.4. Loading combinations and analyses for safety-related reactor internals, including core support structures, are discussed in Subsections 3.9.3.1, 3.9.5.3.5, and 3.9.5.3.6.

3.9.5.2.2 Pressure Differential During Rapid Depressurization

A digital computer code is used to analyze the transient conditions within the reactor vessel following the main steamline break between the vessel nozzle and main steam isolation valve. The analytical model of the vessel consists of nine nodes which are connected to the necessary adjoining nodes by flow paths having the required resistance and inertial characteristics. The program solves the energy and mass conservation equations for each node to give the depressurization rates and pressures in the various regions of the reactor. Figure 3.9-5 shows the nine reactor nodes. The computer code used is the GE Short-Term Thermal-Hydraulic Model described in Reference 3.9-4. This model has been approved for use in ECCS conformance evaluation under 10CFR50 Appendix K. In order to adequately describe the blowdown pressure effect on the individual assembly components, three features are included

in the model that are not applicable to the ECCS analysis and are therefore not described in Reference 3.9-4. These additional features are as follows:

(1) The liquid level in the steam separator region and in the annulus between the dryer skirt and the pressure vessel is tracked to more accurately determine the flow and mixture quality in the steam dryer and in the steamline.

(2) The flow path between the bypass region and the shroud head is more accurately modeled, since the fuel assembly pressure differential is influenced by flashing in the guide tubes and bypass region for a steamline break. In the ECCS analysis, the momentum equation is solved in this flow path but its irreversible loss coefficient is conservatively set at an arbitrary low value.

(3) The enthalpies in the guide tubes and the bypass are calculated separately since the fuel assembly pressure differential is influenced by flashing in these regions. In the ECCS analysis, these regions are lumped.

3.9.5.2.3 Feedwater Line and Main Steamline Break

3.9.5.2.3.1 Accident Definition

Both a feedwater line break (the largest liquid line break) and a Main Steamline (MSL) break (the largest steamline break) upstream of the MSIV are considered in determining the design basis accident for the safety-related reactor internals, including the core support structures.

The feedwater line break is the same as the design basis LOCA described in Subsection 6.2.1.1.3.3.1. A sudden, complete circumferential break is assumed to occur in one feedwater line. The pressure differentials on the reactor internals and core support structures are in all cases lower than those for the MSL break.

The analysis for the MSL break assumes a sudden, complete circumferential break of one main steamline at the reactor vessel nozzle, downstream of the limiting flow area (Subsection 6.2.1.1.3.3.2).

The steamline break accident produces significantly higher pressure differential across the reactor internal structures than does the feedwater line break. This results from the higher reactor depressurization rate associated with the steamline break. Therefore, the steamline break is the design basis accident for internal pressure differentials.

3.9.5.2.3.2 Effects of Initial Reactor Power and Core Flow

The maximum internal pressure loads can be considered to be composed of two parts: steady-state and transient pressure differentials. For a given plant, the core flow and power are the two major factors which influence the reactor internal pressure differentials. The core flow essentially affects only the steady-state part. For a fixed power, the greater the core flow, the larger will be the steady-state pressure differentials. On the other hand, the core power affects

both the steady-state and the transient parts. As the power is decreased, there is less voiding in the core and, consequently, the steady-state core pressure differential is less. However, less voiding in the core also means that less steam is generated in the RPV and, thus, the depressurization rate and the transient part of the maximum pressure load is increased. As a result, the total loads on some components are higher at low power.

To ensure that calculated pressure differences bound those which could be expected if a steamline break should occur, an analysis is conducted at a low power high-recirculation flow condition in addition to the standard safety analysis condition at high power, rated recirculation flow. The power chosen for analysis is the minimum value permitted by the recirculation system controls at rated recirculation drive flow (i.e., the drive flow necessary to achieve rated core flow at rated power).

This condition maximizes those loads which are inversely proportional to power. It must be noted that this condition, while possible, is unlikely; first, because the reactor will generally operate at or near full power; second, because high core flow is neither required nor desirable at such a reduced power condition.

Table 3.9-3 summarizes the maximum pressure differentials. Case 1 is the safety analysis condition; Case 2 is the low power high-flow condition.

3.9.5.2.4 Seismic and Other Reactor Building Vibration Events

The loads due to earthquake and other reactor building vibration (RBV) acting on the structure within the reactor vessel are based on a dynamic analysis described in Sections 3.7, 3.8, and Subsection 3.9.2.5. Dynamic analysis is performed by coupling the lumped-mass model of the reactor vessel and internals with the building model to determine the system natural frequencies and mode shapes. The relative displacement, acceleration, and load response is then determined by either the time-history method or the response-spectrum method.

3.9.5.3 Design Bases

3.9.5.3.1 Safety Design Bases

The reactor internals, including core support structures, shall meet the following safety design bases:

(1) The reactor vessel nozzles and internals shall be so arranged as to provide a floodable volume in which the core can be adequately cooled in the event of a breach in the nuclear system process barrier external to the reactor vessel.

(2) Deformation of internals shall be limited to assure that the control rods and core standby cooling systems can perform their safety-related functions.

(3) Mechanical design of applicable structures shall assure that safety design bases (1) and (2) are satisfied so that the safe shutdown of the plant and removal of decay heat are not impaired.

3.9.5.3.2 Power Generation Design Bases

The reactor internals, including core support structures, shall be designed to the following power generation design bases:

(1) The internals shall provide the proper coolant distribution during all anticipated normal operating conditions to full power operation of the core without fuel damage.

(2) The internals shall be arranged to facilitate refueling operations.

(3) The internals shall be designed to facilitate inspection.

3.9.5.3.3 Design Loading Categories

The basis for determining faulted dynamic event loads on the reactor internals is shown in Sections 3.7, 3.8 and Subsections 3.9.2.5, 3.9.5.2.3 and 3.9.5.2.4. Table 3.9-2 shows the load combinations used in the analysis.

Core support structures are Seismic Category I and ASME Code Class CS structures, which meet the stress limits of the ASME Code Section III, Subsection NG. For these components and the safety-related internals, Level A, B, C, and D service limits are applied to the normal, upset, emergency, and faulted loading conditions, respectively, as defined in the design specification. Stress intensity and other design limits are discussed in Subsections 3.9.5.3.5 and 3.9.5.3.6

3.9.5.3.4 Response of Internals Due to Steamline Break Accident

As described in Subsection 3.9.5.2.3.2, the maximum pressure loads acting on the reactor internal components result from steamline break upstream of the main steam isolation valve and, on some components, the loads are greatest with operation at the minimum power associated with the maximum core flow (Table 3.9-3, Case 2). This has been substantiated by the analytical comparison of liquid versus steamline breaks and by the investigation of the effects of core power and core flow.

It has also been pointed out that, although possible, it is not probable that the reactor would be operating at the rather abnormal condition of minimum power and maximum core flow. More realistically, the reactor would be at or near a full power condition and thus the maximum pressure loads acting on the internal components would be as listed under Case 1 in Table 3.9-3.

3.9.5.3.5 Stress and Fatigue Limits for Core Support Structures

The design and construction of the core support structures are in accordance with ASME Code Section III, Subsection NG.

3.9.5.3.6 Stress, Deformation, and Fatigue Limits for Safety Class and Other Reactor Internals (Except Core Support Structures)

For safety class reactor internals, the stress deformation and fatigue criteria listed in Tables 3.9-4 through 3.9-7 are based on the criteria established in applicable codes and standards for similar equipment, by manufacturer's standards, or by empirical methods based on field experience and testing. For the quantity SF_{min} (minimum safety factor) appearing in those tables, the following values are used:

Service Level	Service Condition	SF_{min}
A	Normal	2.25
B	Upset	2.25
C	Emergency	1.5
D	Faulted	1.125

Components inside the reactor pressure vessel such as control rods which must move during accident condition have been examined to determine if adequate clearances exist during emergency and faulted conditions. No mechanical clearance problems have been identified. The forcing functions applicable to the reactor internals are discussed in Subsection 3.9.2.5.

The design criteria, loading conditions, and analyses that provide the basis for the design of the safety class reactor internals other than the core support structures meet the guidelines of NG-3000 and are constructed so as not to adversely affect the integrity of the core support structures (NG-1122).

The design requirements for equipment classified as non-safety (other) class internals (e.g., steam dryers and shroud heads) are specified with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate. Where Code design requirements are not applicable, accepted industry or engineering practices are used.

3.9.6 Testing of Pumps and Valves

Inservice testing of safety-related pumps and valves will be performed in accordance with the requirements of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1, 6, and 10. Table 3.9-8 lists the inservice testing parameters and frequencies for the safety-related pumps and valves. The reason for each code defined testing exception or justification for each code exemption request is noted in the description of the affected pump or valve. Valves having a containment isolation function are also noted in the listing. Inservice inspection is discussed in Subsection 5.2.4 and Section 6.6.

Details of the inservice testing program, including test schedules and frequencies, will be reported in the inservice inspection and testing plan to be provided by the applicant referencing the ABWR design. The plan will integrate the applicable test requirements for safety-related pumps and valves, including those listed in the technical specifications, Chapter 16, and the containment isolation system, Subsection 6.2.4. 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 3.4.4.1. This plan will include baseline pre-service testing to support the periodic inservice testing of the components. Depending on the test results, the plan will provide a commitment to disassemble and inspect the safety-related pumps and valves when limits of the OM Code are exceeded, as described in the following paragraphs. The primary elements of this plan, including the requirements of Generic Letter 89-10 for motor operated valves, are delineated in the subsections to follow. (See Subsection 3.9.7.3 for COL license information requirements.)

3.9.6.1 Testing of Safety-Related Pumps

For each pump, the design basis and required operating conditions (including tests) under which the pump will be required to function will be established. These designs (design basis and required operating) conditions include flow rate and corresponding head for each system mode of pump operation and the required operating time for each mode, acceptable bearing vibration levels, seismic/dynamic loads, fluid temperature, ambient temperature, and pump motor minimum voltage.

The COL applicant will establish the following design and qualification requirements and will provide acceptance criteria for these requirements. For each size, type, and model the COL applicant will perform testing encompassing design conditions that demonstrate acceptable flow rate and corresponding head, bearing vibration levels, and pump internals wear rates for the operating time specified for each system mode of pump operation. From these tests the COL applicant will also develop baseline (reference) hydraulic and vibration data for evaluating the acceptability of the pump after installation. The COL applicant will ensure that the pump specified for each application is not susceptible to inadequate minimum flow rate and inadequate thrust bearing capacity. With respect to minimum flow pump operation, the sizing of each minimum recirculation flow path is evaluated to assure that its use under all analyzed conditions will not result in degradation of the pump. The flow rate through minimum recirculation flow paths can also be periodically measured to verify that flow is in accordance with the design specification.

The ABWR safety-related pumps and piping configurations accommodate in-service testing at a flow rate at least as large as the maximum design flow for the pump application. The safety-related pumps are provided with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation. These pumps can be disassembled for evaluation when Part 6 testing results in a deviation which falls within the “required action range.” The Code provides criteria limits for the test

parameters identified in Table 3.9-8. A program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related pumps, including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (See Subsection 3.9.7.3(1) for COL license information requirements.)

3.9.6.2 Testing of Safety-Related Valves

3.9.6.2.1 Check Valves

(1) Design and Qualification

For each check valve with an active safety-related function, the design basis and required operating conditions (including testing) under which the check valve will be required to perform will be established.

The COL applicant will establish the following design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model the COL applicant will ensure the design adequacy of the check valve under design (design basis and required operating) conditions. These design conditions include all the required system operating cycles to be experienced by the valve (numbers of each type of cycle and duration of each type cycle), environmental conditions under which the valve will be required to function, severe transient loadings expected during the life of the valve such as waterhammer or pipe break, life-time expectation between major refurbishments, sealing and leakage requirements, corrosion requirements, operating medium with flow and velocity definition, operating medium temperature and gradients, maintenance requirements, vibratory loading, planned testing and methods, test frequency and periods of idle operation. The design conditions may include other requirements as identified during detailed design of the plant systems. This testing of each size, type and model shall include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test (such as prototype) of similar valves that support qualification of the required valve where similarity must be justified by technical data. The COL applicant will ensure proper check valve application including selection of the valve size and type based on the system flow conditions, installed location of the valve with respect to sources of turbulence, and correct orientation of the valve in the piping (i.e., vertical vs horizontal) as recommended or required by the manufacturer. The COL applicant will ensure that valve design features, material, and surface finish will accommodate non-intrusive diagnostic testing methods available in the industry or as specified. The COL applicant will also ensure that flow through the valve is determinable from installed instrumentation and that the valve disk positions are determinable without disassembly such as by use of non-intrusive diagnostic methods. Valve internal parts

are designed with self-aligning features for purpose of assured correct installation. The COL applicant will compare the maximum loading on the check valve under design basis and the required operating conditions to the allowable structural capability limits for the individual parts of the check valve. The qualification acceptance criteria noted above will include baseline data developed during qualification testing and will be used for verifying the acceptability of the check valves after installation.

(2) Pre Operational Testing

The COL applicant will test each check valve in the open and/or close direction, as required by the safety function, under all normal operating system conditions. To the extent practical, testing of the valves as described in this section will be performed under fluid temperature conditions that would exist during a cold shutdown as well as under fluid temperature conditions that would be experienced by the valve during other modes of plant operation. The testing will identify the flow needed to open the valve to the full-open position. The testing will include the effects of rapid pump starts and stops as required by expected system operating conditions. The testing will include any other reverse flow conditions that may be required by expected system operating conditions. The COL applicant will examine the disk movement during valve testing and verify the leak-tightness of valve when fully closed. By using methods such as non-intrusive diagnostic equipment, the COL applicant will examine the open valve disk stability under the flow conditions during normal and other required system operating conditions.

The parameters and acceptance criteria for demonstrating that the above functional performance requirements have been met are as follows:

(a) During all test modes that simulate expected system operating conditions, the valve disk fully opens or fully closes as expected based on the direction of the differential pressure across the valve.

(b) Leak-tightness of valve when fully closed is within established limits, as applicable.

(c) Valve disk positions are determinable without disassembly.

(d) Valve testing must verify free disk movement whenever moving to and from the seat.

(e) The disk is stable in the open position under normal and other required system operating fluid flow conditions.

(f) The valve is correctly sized for the flow conditions specified, i.e., the disk is in full open position at normal full flow operating condition.

(g) Valve design features, material, and surfaces accommodate non-intrusive diagnostic testing methods available in the industry or as specified.

(h) Piping system design features accommodate all the applicable check valve testing requirements as described in Table 3.9-8.

All ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the check valves under design conditions. Inservice testing will incorporate the use of advance non-intrusive techniques to periodically assess degradation and the performance characteristics of the check valves. The Part 10 tests will be performed, and check valves that fail to exhibit the required performance can be disassembled for evaluation. The Code provides criteria limits for the test parameters identified in Table 3.9-8. A program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related check valves, including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience (see Subsection 3.9.7.3 for COL license information requirements).

3.9.6.2.2 Motor-Operated Valves

For each motor-operated valve assembly (MOV) with an active safety related function, the design basis and required operating conditions (including testing) under which the MOV will be required to perform are established for the development and implementation of the design, qualification and preoperational testing.

[*Table 5 of DCD/Introduction identifies the commitments of design, qualification, and preoperational testing for MOVs, which, if changed, requires NRC Staff review and approval prior to implementation. The applicable portions of the Tier 2 sections and tables, identified on Table 5 of DCD/Introduction for this restriction, are italicized on the sections and tables themselves.*]*

(1) [*Design and Qualifications*

The COL applicant will establish the following design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model the COL applicant will determine the torque and thrust (as applicable to the type of MOV) requirements to operate the MOV and will ensure the adequacy of the torque and thrust that the motor-operator can deliver under design (design basis and required operating) conditions. The COL applicant will also test each size, type, and model under a range of differential pressure and flow conditions up to the design conditions. These design conditions include fluid flow, differential pressure (including pipe break), system pressure, fluid temperature, ambient temperature, minimum voltage, and minimum and maximum stroke time

* See Section 3.5 of DCD/Introduction.

requirements. This testing of each size, type and model shall include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test (such as prototype) of similar valves that support qualification of the required valve where similarity must be justified by technical data. From this testing the COL applicant will demonstrate that the results of testing under in situ or installed conditions can be used to ensure the capability of the MOV to operate under design conditions. The COL applicant will ensure that the structural capability limits of the individual parts of the MOV will not be exceeded under design conditions. The COL applicant will ensure that the valve specified for each application is not susceptible to pressure locking and thermal binding.]*

(2) [*Pre-operational Testing*

The COL applicant will test each MOV in the open and close directions under static and maximum achievable conditions using diagnostic equipment that measures torque and thrust (as applicable to the type of MOV), and motor parameters. The COL applicant will test the MOV under various differential pressure and flow to maximum achievable conditions and perform a sufficient number of tests to determine the torque and thrust requirements at design conditions. The COL applicant will determine the torque and thrust requirements to close the valve for the position at which there is diagnostic indication of hard seat contact. The determination of design torque and thrust requirements will be made for such parameters as differential pressure, fluid flow, undervoltage, temperature and seismic dynamic effects for MOVs that must operate during these transients. The design torque and thrust requirements will be adjusted for diagnostic equipment inaccuracies. For the point of control switch trip, the COL applicant will determine any loss in torque produced by the actuator and thrust delivered to the stem for increasing differential pressure and flow conditions (referred to as load sensitive behavior). The COL applicant will compare the design torque and thrust requirements to the control switch trip torque and thrust subtracting margin for load sensitive behavior, control switch repeatability, and degradation. The COL applicant will measure the total thrust and torque delivered by the MOV under static and dynamic conditions (including diagnostic equipment inaccuracy and control switch repeatability) to compare to the allowable structural capability limits for the individual parts of the MOV. The COL applicant will test for proper control room position indication of the MOV.

The parameters and acceptance criteria for demonstrating that the above functional performance requirements have been met are as follows:

(a) *As required by the safety function: the valve must fully open; the valve must full close with diagnostic indication of hard seat contact.*

* See Subsection 3.9.6.2.2.

(b) *The control switch settings must provide adequate margin to achieve design requirements including consideration of diagnostic equipment inaccuracy, control switch repeatability, load sensitive behavior, an margin for degradation.*

(c) *The motor output capability at degraded voltage must equal or exceed the control switch setting including consideration of diagnostic equipment inaccuracy, control switch repeatability, load sensitive behavior and margin for degradation.*

(d) *The maximum torque and thrust (as applicable for the type MOV) achieved by the MOV including diagnostic equipment inaccuracies and control switch repeatability must not exceed the allowable structural capability limits for the individual parts of the MOV.*

(e) *The remote position indication testing must verify that proper disk position is indicated in the control room.*

(f) *Stroke time measurements taken during valve opening and closing must meet minimum and maximum stroke time requirements.*]*

The inservice testing of MOVs will rely on diagnostic techniques that are consistent with the state of the art and which will permit an assessment of the performance of the valve under actual loading. Periodic testing per GL89-10 Paragraphs D and J will be conducted under adequate differential pressure and flow conditions that allow a justifiable demonstration of continuing MOV capability for design basis conditions. The COL applicant will determine the optimal frequency of this periodic verification. The frequency and test conditions will be sufficient to demonstrate continuing design basis and required operating capability. See Subsection 3.9.7.3 for COL license information requirements. The Code provides criteria limits for the test parameters identified in Table 3.9-8 for code inservice testing.

A program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related MOVs, including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (See Subsection 3.9.7.3 for COL license information requirements.)

3.9.6.2.3 Power Operated Valves

(1) Design and Qualification

For each power-operated (includes pneumatic- hydraulic-, piston-, and solenoid-operated) valve assembly (POV) with an active safety-related function, the design

* See Subsection 3.9.6.2.2.

basis and required operating conditions (including testing) under which the POV will be required to perform will be established.

The COL applicant will establish the following design and qualification requirements and will provide acceptance criteria for these requirements. By testing each size, type, and model the COL applicant will determine the force (as applicable to the type of POV) requirement to operate the POV and will ensure the adequacy of the force that the operator can deliver under design (design basis and required operating) conditions. The COL applicant will also test each size, type, and model under a range of differential pressure and flow conditions up to the design conditions. These design conditions include fluid flow, differential pressure (including pipe break), system pressure, fluid temperature, ambient temperature, minimum air supply system (or accumulator) pressure, spring force, and minimum and maximum stroke time requirements. This testing of each size, type and model shall include test data from the manufacturer, field test data for dedication by the COL applicant, empirical data supported by test, or test (such as prototype) of similar valves that support qualification of the required valve where similarity must be justified by technical data. From this testing, the COL applicant will demonstrate that the results of testing under in-situ conditions can be used to ensure the capability of the POV to operate under design conditions. The COL applicant will ensure that the structural capability limits of the assembly and the individual parts of the POV will not be exceeded under design conditions. The COL applicant will ensure that packing adjustment limits are specified for the valve for each application such that it is not susceptible to stem binding.

(2) Pre-operational Testing

The COL applicant will test each POV in the open and close directions under static and maximum achievable conditions using diagnostic equipment that measures or provides information to determine total friction, stroke time, seat load, spring rate, and travel under normal pneumatic or hydraulic pressure (as applicable to the type of POV), and minimum pneumatic or hydraulic pressure. The COL applicant will test the POV under various differential pressure and flow up to maximum achievable conditions and perform a sufficient number of tests to determine the force requirements at design conditions. The COL applicant will determine the force requirements to close the valve for the position at which there is a diagnostic indication of full valve closure (as required for the safety function of the applicable valves). The determination of design force requirements will be made for such parameters as differential pressure, fluid flow, minimum pneumatic or hydraulic pressure, power supply, temperature, and seismic/dynamic effects for POVs that must operate during these transients. The design force requirements will be adjusted for diagnostic equipment inaccuracies.

The COL applicant will measure the total force delivered by the POV under static and dynamic conditions (including diagnostic equipment inaccuracies) to compare to the

allowable structural capability limits for the assembly and individual parts of the POV. The COL applicant will test for proper control room position indication of the POV.

The parameters and acceptance criteria for demonstrating that the above functional performance requirements have been met are a follows:

(a) As required by the safety function, the valve must fully open and/or the valve must fully close with diagnostic indication of hard seat contact.

(b) The assembly must demonstrate adequate margin to achieve design requirements including consideration of diagnostic equipment inaccuracies and margin for degradation.

(c) The assembly must demonstrate adequate output capability of the power-operator at minimum pneumatic or hydraulic pressure of electrical supply (or loss of motive force for fail-safe positioning) with consideration of diagnostic equipment inaccuracies and margin for degradation.

(d) The maximum force (as applicable for the type of POV) achieved by the POV including diagnostic equipment inaccuracies must not exceed the allowable structural capability limits for the assembly and individual part of the POV.

(e) The remote position indication testing must verify that proper disk position is indicated in the control room and other remote locations relied upon by operators in any emergency situation.

(f) Stroke-time measurements taken during valve opening and closing must meet minimum and maximum stroke-time requirements.

(g) For solenoid-operated valves (SOVs), the Class 1E electrical requirements are to be verified. The SOV should be verified to be capable of performing for design requirements for energized or denergized and rated appropriately for the electrical power supply amperage and voltage.

(h) Provide leak-tight seating which must meet specified maximum leakage rate, or meet leakage rate to ensure an overall containment maximum leakage.

All ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the POVs under design conditions. Inservice testing will incorporate the use of advance non-intrusive techniques to periodically assess degradation and the performance characteristics of the POVS. The Part 10 tests will be performed, and valves that fail to exhibit the required performance can be disassembled for evaluation. The Code provides criteria limits for the test parameters identified in Table 3.9-8. A program will be developed by the COL applicant to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety-related POVs including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize

disassembly based on past disassembly experience. (See Subsection 3.9.7.3 for COL license information requirements.)

3.9.6.2.4 Isolation Valve Leak Tests

The leaktight integrity will be verified for each valve relied upon to provide a leaktight function. These valves include:

(1) Pressure isolation valves—valves that provide isolation of pressure differential from one part of a system from another or between systems.

(2) Temperature isolation valves—valves whose leakage may cause unacceptable thermal loading on supports or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps.

(3) Containment isolation valves—valves that perform a containment isolation function in accordance with the Evaluation Against Criterion 54, Subsection 3.1.2.5.5.2, including valves that may be exempted from Appendix J, Type C, testing but whose leakage may cause loss of suppression pool water inventory.

Leakage rate testing for valve group (1) is addressed in Subsection 3.9.6. Valve groups (2) and (3) will be tested in accordance with Part 10, Paragraph 4.2.2.3. The fusible plug valves that provide a lower drywell flood for severe accidents are described in Subsection 9.5.12. The valves are safety-related due to the function of retaining suppression pool water as shown in Figure 9.5-3. The fusible plug valve is a nonreclosing pressure relief device and the Code requires replacement of each at a maximum of 5-year intervals.

3.9.7 COL License Information

3.9.7.1 Reactor Internals Vibration Analysis, Measurement and Inspection Program

The first COL applicant will provide, at the time of application, the results of the vibration assessment program for the ABWR prototype internals. These results will include the following information specified in Regulatory Guide 1.20.

RG 1.20	Subject
C.2.1	Vibration Analysis Program
C.2.2	Vibration Measurement Program
C.2.3	Inspection Program
C.2.4	Documentation of Results

NRC review and approval of the above information on the first COL applicant's docket will complete the vibration assessment program requirements for prototype reactor internals.

In addition to the information tabulated above, the first COL applicant will provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals.

Subsequent COL applicants need only provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals (Subsection 3.9.2.4).

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

COL applicants will identify 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, either provide an appropriate analysis to demonstrate the required design life or provide designs to mitigate the magnitude or duration of the cyclic loads (Subsection 3.9.3.1.)

3.9.7.3 Pump and Valve Testing Program

COL applicants will provide plant specific environmental parameters for the equipment qualification program in accordance with Subsection 3.9.3.2.

COL applicants will provide a plan for the detailed pump and valve inservice testing and inspection program. This plan will:

(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 (Subsections 3.9.6, 3.9.6.1, 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3).

(2) Provide a study to determine the optimal frequency of the periodic verification of the continuing MOV capability for design basis conditions (Subsections 3.9.6.2.1, 3.9.6.2.2, and 3.9.6.2.3).

The COL applicant will include 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 in the development of the respective safety-related pump and valve design specifications.

The COL applicant will address the design, qualification, and preoperational testing for MOVs as discussed in Subsection 3.9.6.2.2 prior to plant startup.

3.9.7.4 Audit of Design Specification and Design Reports

COL applicants will make available to the NRC staff design specification and design reports required by ASME Code for vessels, pumps, valves and piping systems for the purpose of audit (Subsection 3.9.3.1).

The COL applicant shall ensure that the piping system design is consistent with the construction practices, including inspection and examination methods, of the ASME Code edition and addenda as endorsed in 10CFR50.55a in effect at the time of application.

The COL applicant shall identify ASME Code editions and addenda other than those listed in Tables 1.8-21 and 3.2-3, that will be used to design ASME Code Class 1, 2 and 3 pressure retaining components and supports. The applicable portions of the ASME Code editions and addenda shall be identified to the NRC staff for review and approval with the COL application (Subsection 3.9.3.1).

3.9.8 References

3.9-1 "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors - Supplement 1 to CENPD-287", WCAP-15942-P-A, March 2006.

3.9-2 "ABB Seismic/LOCA Evaluation Methodology for Boiling Water Reactor Fuel," Westinghouse Report CENPD-288-P-A (proprietary), CENPD-288-NP-A (non-proprietary), July 1996.

3.9-3 NEDE-24057-P (Class III) and NEDO-24057 (Class I) Assessment of Reactor Internals. Vibration in BWR/4 and BWR/5 Plants, November 1977. Also NEDO-24057, Amendment 1, December 1978, and NEDE-2-P 24057 Amendment 2, June 1979.

3.9-4 "General Electric Company, Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K", NEDE-20566P, Proprietary Document, November 1975.

3.9-5 "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", NUREG-0619.

3.9-6 ["*General Electric Environmental Qualification Program", NEDE-24326-1-P, Proprietary Document, January 1983.*]*

3.9-7 [*Functional Capability of Piping Systems, U.S. Nuclear Regulatory Commission, NUREG-1367, November 1992.*]†

3.9-8 Toshiba Report A98-1101-0007, "Generic Criteria for High Frequency Cutoff of BWR Equipment", Revision 0, June, 30, 2009.

* See Section 3.10 and Appendix 3K. This reference is same as Reference 3.11-2 (Subsection 3.11.7).

† See Subsection 3.9.1.7.

3.9-9 [*General Electric Company, Plain Carbon Steels, 408HA414, Rev. 1*.]†

3.9-10 [*EPRI NP-5639, "Guidelines for Piping System Reconciliation", May 1988*.]†

3.9-11 [*NUREG/CR-6049, Piping Benchmark Problems for the GE ABWR, August 1993*.]†

3.9-12 NCIG-01, Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants, Revision 2.

3.9-13 "STP 3 ABWR Prototype Reactor Internals Flow-Induced Vibration Assessment Program," WCAP-17256, Rev. 2, June 2011.

Table 3.9-1 Plant Events

A. Plant Operating Events	ASME Code Service Limit[1]	No. of Cycles/Events[2]
1. Boltup[2]	A	45
2. Hydrostatic Test (two test cycles for each boltup cycle)	Testing	90
3. Startup (56°C/h Heatup Rate)[3]	A	260
4. Daily and Weekly Reduction to 50% Power[2]	A	18,000
5. Control Rod Pattern Change[2]	A	600
6. Loss of Feedwater Heaters	B	120
7. Scram:		
a. Turbine Generator Trip, Feedwater On, and Other Scrams	B	125
b. Loss of Feedwater Flow, Loss of Auxiliary Power	B	139
c. Turbine Bypass, Single Safety or Relief Valve Blowdown	B	8
8. Reduction to 0% Power, Hot Standby, Shutdown (56°C/h Cooldown Rate)[3]	A	252
9. Refueling Shutdown with Head Spray and Unbolt[2]	A	45
10. Scram:		
a. Reactor Overpressure with Delayed Scram (Anticipated Transient Without Scram, ATWS)	C	1[4]
b. Automatic Blowdown	C	1[4]
11. Improper or Sudden Start of Recirculation Pump with Cold Bottom Head or Hot Standby—Drain Shut Off—Pump Restart	C	1[4]

See next page for footnotes

Table 3.9-1 Plant Events (Continued)

B. Dynamic Loading Events[5]	ASME Code Service Limit[1]	No. of Cycles/Events[2]
12. [*Safe Shutdown Earthquake (SSE) Event at Rated Power Operating Conditions*	*B*[6]	*2 Events* [7] *10 Cycles/event*]*
13. Safe Shutdown Earthquake (SSE) at Rated Power Operating Conditions	D[8]	1 Event[4]
14. Turbine Stop Valve Full Closure (TSVC) [9] During Event 7a and Testing	B	320 Events 3 Cycles/ Event
15. Safety/Relief Valve (SRV) Actuations During Events 7a and 7b		
— One plus Two Adjacent	B	2536 Events[10]
— All plus Automatic Depressurization System	B	264 Events[10]
16. Loss-of-Coolant Accident (LOCA)		
Small Break LOCA (SBL) or	D[8]	1[4]
Intermediate Break LOCA (IBL) or	D[8]	1[4]
Large Break LOCA (LBL)	D[8]	1[4]

1 These ASME Code Service Limits apply to ASME Code Class 1, 2 and 3 components, component supports and Class CS structures. Different limits apply to Class MC and CC containment vessels and components, as discussed in Section 3.8.

2 Some events apply to reactor pressure vessel (RPV) only. The number of events/cycles applies to RPV as an example.

3 Bulk average vessel coolant temperature change in any one-hour period.

4 The annual encounter probability of a single event is $<10^{-2}$ for a Level C event and $<10^{-4}$ for a Level D event (Subsection 3.9.3.1.1.5).

5 Table 3.9-2 shows the evaluation basis combination of these dynamic loadings.

6 [*The effects of displacement-limited, seismic anchor motions (SAM) due to SSE shall be evaluated for safety-related ASME Code Class 1, 2, and 3 components and component supports to ensure their functionality during and following an SSE. The SAM effects shall include relative displacements of piping between building floors and slabs, at equipment nozzles, at piping penetrations and at connections of small diameter piping to large diameter piping. See Table 3.9-2 and Note 7 of Table 3.9-2 for stress limits to be used to evaluate the SAM effects.*]*

7 Use 20 peak SSE cycles for evaluation of ASME Class I components and core support structures for Service Level B fatigue analysis. Alternatively, an equivalent number of fractional SSE cycles may be used in accordance with Subsection 3.7.3.2.

8 Appendix F or other appropriate requirements of the ASME Code are used to determine the service Level D limits, as described in Subsection 3.9.1.4.

9 Applicable to main steam piping system only.

10.The number of Reactor Building vibratory load cycles on the reactor vessel and internal components is 19,600 cycles of varying amplitude during the 264 events of SRV actuation. The number of Reactor Building vibratory load cycles on the piping systems inside the containment is 2536 events of single SRV actuation, with 3 stress cycles per event and 264 events of SRV actuation of all valves or the Automatic Depressurization System valves, with 3 stress cycles per event.

* See Subsection 3.9.1.7. The change restriction applies only to piping design.

Table 3.9-2 Load Combinations and Acceptance Criteria for Safety-Related, ASME Code Class 1, 2 and 3 Components, Component Supports, and Class CS Structures

Plant Event	Service Loading Combination [1 2 3]	ASME Service Level[4]
1. Normal Operation (NO)	N	A
2. Plant/System Operating Transients (SOT)	(a) N + TSVC (b) N + SRV[5]	B B
3. [*NO + SSE*	*N + SSE*	*B]**[6 7]
4. Not Used		
5. Infrequent Operating Transient (IOT), ATWS	N[8] + SRV[5]	C[9]
6. SBL	N + SRV[10] + SBL	C[9]
7. SBL or IBL + SSE	N + SBL (or IBL) + SSE + SRV[10]	D[9 12]
8. LBL + SSE	N + LBL + SSE	D[9 12]
9. NLF	N + SRV[5]+ TSVC	D[9]

1 See Legend on the following pages for definition of terms. See Table 3.9-1 for plant events and cycles information.
The service loading combination also applies to Seismic Category I Instrumentation and electrical equipment (Section 3.10).

2 For vessels and pumps, loads induced by the attached piping are included as identified in their design specification.
For piping systems, water (steam) hammer loads are included as identified in their design specification.

3 The method of combination of the loads is in accordance with NUREG-0484, Revision 1.

4 The service levels are as defined in appropriate subsection of ASME Section III, Division 1.

5 The most limiting load combination case among SRV(1), SRV(2) and SRV (ALL). For main steam and branch piping evaluation, additional loads associated with relief line clearing and blowdown into the suppression pool are included.

6 Applies only to fatigue evaluation of ASME Code Class I components and core support structures. See Dynamic Loading Event No. 12, Table 3.9-1, and Note 7 of Table 3.9-1 for number of cycles.

7 [*For ASME Code 1,2 and 3 piping the following changes and additions to ASME Code Section III Subsections NB-3600, NC-3600 and ND-3600 are necessary and shall be evaluated to meet the following stress limits:*

(a) *ASME Code Class 1 Piping:*

$$S_{SAM} = C_2 \frac{D_0}{2I} M_c \le 6.0Sm$$

where: S_{SAM} *is the nominal value of seismic anchor motion stress*

M_c *is the combined moment range equal to the greater of (1) the resultant range of thermal and thermal anchor movements plus one-half the range of the SSE anchor motion, or (2) the resultant range of moment due to the full range of the SSE anchor motions alone.*

C_2, D_0 *and* I *are defined in ASME Code Subsection NB-3600*

SSE inertia and seismic anchor motion loads shall be included in the calculation of ASME Code Subsection NB-3600 equations (10) and (11).

(b) *ASME Code Class 2 and 3 Piping:*

$$S_{SAM} = i\frac{M_c}{Z} \le 3.0S_h \quad (\le 2.0S_y)$$

where: S_{SAM} *and* M_c *are as defined in (a) above, and*

i and Z are defined in ASME Code Subsections NC/ND-3600

SSE inertia and seismic anchor motion loads shall not be included in the calculation of ASME Code Subsections NC/ND-3600 Equation (9), Service Levels A and B and Equations (10) and (11).]*

8 The reactor coolant pressure boundary is evaluated using in the load combination the maximum pressure expected to occur during ATWS.

9 [*All ASME Code Class 1,2 and 3 Piping systems which are essential for safe shutdown under the postulated events are designed to meet the requirements of NUREG-1367 (Reference 3.9-7).*]* Piping system dynamic moments can be calculated using an elastic response spectrum or time history analysis.

10 The most limiting load combination case among SRV(1), SRV(2) and SRV (ADS). See Note (5) for main steam and branch piping.

11 Not used

12 [*For active Class 2 and 3 pumps (and active Class 1,2 and 3 valves), the stresses are limited by criteria:* $\sigma m < 1.2S$ *(or 0.75 Sy), and* (σm *or* σL) + $\sigma b \le 1.8S$ *(or 1.1 Sy), where the notations are as defined in the ASME Code, Section III, Subsections NB and NC or ND, respectively.*]†

* See Subsection 3.9.1.7.

† See Section 3.10.

Load Definition Legend:

Normal (N)	—	Normal and/or abnormal loads associated with the system operating conditions, including thermal loads, depending on acceptance criteria.
SOT	—	System Operational Transient (Subsection 3.9.3.1).
IOT	—	Infrequent Operational Transient (Subsection 3.9.3.1).
ATWS	—	Anticipated Transient Without Scram.
TSVC	—	Turbine stop valve closure induced loads in the main steam piping and components integral to or mounted thereon.
RBV Loads	—	Dynamic loads in structures, systems and components because of reactor building vibration (RBV) induced by a dynamic event.
NLF	—	Non-LOCA faulted.
SSE	—	RBV loads induced by safe shutdown earthquake.
SRV(l), SRV(2)	—	RBV loads induced by SRV discharge of one or two adjacent valves, respectively.
SRV(ALL)	—	RBV loads induced by actuation of all SRVs which activate within milliseconds of each other (e.g., turbine trip operational transient).
SRV(ADS)	—	RBV loads induced by the actuation of SRVs associated with the Automatic Depressurization System, which actuate within milliseconds of each other during the postulated small or intermediate break LOCA, or SSE.
LOCA	—	The loss-of-coolant accident associated with the postulated pipe failure of a high-energy reactor coolant line. The load effects are defined by $LOCA_1$ through $LOCA_7$. LOCA events are grouped in three categories, SBL, IBL or LBL, as defined here.
$LOCA_1$	—	Pool swell (PS) drag/fallback loads on essential piping and components located between the main vent discharge outlet and the suppression pool water upper surface.
$LOCA_2$	—	Pool swell (PS) impact loads acting on essential piping and components located above the suppression pool water upper surface.

$LOCA_3$ — (a) Oscillating pressure-induced loads on submerged essential piping and components during main vent clearing (VLC), condensation oscillations (C)0, or chugging (CHUG), or

(b) Jet impingement (JI) load on essential piping and components as a result of a postulated IBL or LBL event.

Piping and components are defined essential, if they are required for shutdown of the reactor or to mitigate consequences of the postulated pipe failure without offsite power (see introduction to Subsection 3.6).

$LOCA_4$ — RBV load from main vent clearing (VLC).

$LOCA_5$ — RBV loads from condensation oscillations (CO).

$LOCA_6$ — RBV loads from chugging (CHUG).

$LOCA_7$ — Annulus pressurization (AP) loads due to a postulated line break in the annulus region between the RPV and shieldwall. Vessel depressurization loads on reactor internals (Subsection 3.9.2.5) and other loads due to reactor blowdown reaction and jet impingement and pipe whip restraint reaction from the broken pipe are included with the AP loads.

SBL — Loads induced by small break LOCA (Subsections 3.9.3.1.1.3 and 3.9.1.1.4); the loads are: $LOCA_3$(a), $LOCA_4$ and $LOCA_6$. See Note (11).

IBL — Loads induced by intermediate break LOCA (Subsection 3.9.3.1.1.4); the loads are: $LOCA_3$(a) or $LOCA_3$(b), $LOCA_4$, $LOCA_5$ and $LOCA_6$. See Note (11).

LBL — Loads induced by large break LOCA (Subsection 3.9.1.1.4); the loads are: $LOCA_1$ through $LOCA_7$. See Note (11).

Table 3.9-3 Pressure Differentials Across Reactor Vessel Internals

Reactor Component[1]		Maximum Pressure Differences Occurring During a Steamline Break (kPaD)	
		Case 1[2]	Case 2[3]
1.	Core plate and guide tube	184.08	162.01
2.	Shroud support ring and lower shroud (beneath the core plate)	242.04	260.67
3.	Shroud head (at marked elevation)	77.97	149.65
4.	Upper shroud (just below top guide)	90.22	152.4
5.	Core averaged power fuel bundle (bulge at bottom of bundle)	98.07	89.63
5.	Core averaged power fuel bundle (collapse at bottom of top guide)	81.40	79.34
6.	Maximum power fuel bundle (bulge at bottom of bundle)	111.8	96.50
7.	Top guide	42.76	64.72
8.	Steam Dryer	47.56	74.53
–	Shroud head to water level, from points (a) to (b), irreversible pressure drop	92.38	159.85
–	Shroud head to water level, from points (a) to (b), elevation pressure drop	10.40	15.20

1 Item numbers in this column correspond to the location (node) numbers identified in Figure 3.9-5.

2 Instantaneous break initiated at 102% rated core power, 102.4% rated steam flow, and 111.1% rated recirculation flow.

3 Instantaneous break initiated at 54.5% rated core power, 49.8% rated steam flow, and 114.8% rated recirculation flow.

Table 3.9-4 Deformation Limit for Safety Class Reactor Internal Structures Only

Either One of (Not Both)	General Limit
a. Permissible Deformation, DP Analyzed Deformation Causing Loss of Function, DL	$\leq \frac{0.9}{SF_{Min}}$
b. Permissible Deformation, DP Experiment Deformation Causing Loss of Function, DE	$\leq \frac{1.0}{SF_{Min}}$ [1]

Where:

DP = Permissible deformation under stated conditions of Service levels A, B, C or D (normal, upset, emergency or fault)

DL = Analyzed deformation which could cause a system loss of functions[2]

DE = Experimentally determined deformation which could cause a system loss of function

SF_{Min}= Minimum safety factor (see Subsection 3.9.5.3.6)

1 Equation will not be used unless supporting data are provided to the NRC.

2 "Loss of Function" can only be defined quite generally until attention is focused on the component of interest. In cases of interest, where deformation limits can affect the function of equipment and components, they will be specifically delineated. From a practical viewpoint, it is convenient to interchange some deformation condition at which function is assured with the loss of function condition if the required safety margins from the functioning conditions can be achieved. Therefore, it is often unnecessary to determine the actual loss of function condition because this interchange procedure produces conservative and safe designs. Examples where deformation limits apply are: control rod drive alignment and clearances for proper insertion, reactor internal pump wear, or excess leakage of any component.

Table 3.9-5 Primary Stress Limit for Safety Class Reactor Internal Structures Only

	Any One of (No More than One Required)	General Limit
a.	$\dfrac{\text{Elastic evaluated primary stresses, PE}}{\text{Permissible primary stresses, PN}}$	$\leq \dfrac{2.25}{SF_{Min}}$
b.	$\dfrac{\text{Permissible Load, LP}}{\text{Largest lower bound limit load, CL}}$	$\leq \dfrac{1.5}{SF_{Min}}$
c.	Elastic evaluated primary stress, PE	$\leq \dfrac{0.75}{SF_{Min}}$
d.	$\dfrac{\text{Elastic-plastic evaluated nominal primary stress, EP}}{\text{Conventional ultimate strength at temperature, US}}$	$\leq \dfrac{0.9}{SF_{Min}}$
e.	$\dfrac{\text{Permissible Load, LP}}{\text{Plastic instability load, PL}}$	$\leq \dfrac{0.9}{SF_{Min}}$ [1]
f.	$\dfrac{\text{Permissible Load, LP}}{\text{Ultimate load from fracture analysis, UF}}$	$\leq \dfrac{0.9}{SF_{Min}}$ [1]
g.	$\dfrac{\text{Permissible Load, LP}}{\text{Ultimate load or loss of function loaf from test, LE}}$	$\leq \dfrac{1.0}{SF_{Min}}$ [1]

Where:

PE = Primary stresses evaluated on an elastic basis. The effective membrane stresses are to be averaged through the load carrying section of interest. The simplest average bending, shear or torsion stress distribution which will support the external loading will be added to the membrane stresses at the section of interest.

PE = Permissible primary stress levels under service level A or B (normal or upset) conditions under ASME Boiler and Pressure Vessel Code, Section III.

CL = Lower bound limit load with yield point equal to 1.5 Sm, where Sm is the tabulated value of allowable stress at temperature of the ASME III code or its equivalent. The “lower bound limit load” is herein defined as that produced from the analysis of an ideally plastic (non-strain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yield to the uniaxial case.

US = Conventional ultimate strength at temperature or loading which would cause a system malfunction, whichever is more limiting.

Table 3.9-5 Primary Stress Limit for Safety Class Reactor Internal Structures Only (Continued)

EP =	Elastic-plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress curve, which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule may be used.
PL =	Plastic instability loads. The “Plastic Instability Load” is defined here as the load at which any load bearing section begins to diminish its cross-sectional area at a faster rate than the strain hardening can accommodate the loss in area. This type analysis requires a true stress-true strain curve or a close approximation based on monotonic loading at the temperature of loading.
UF =	Ultimate load from fracture analyses. For components which involve sharp discontinuities (local theoretical stress concentration) the use of a “Fracture Mechanics” analysis where applicable utilizing measurements of plane strain fracture toughness may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects as well as gross yielding may be necessary. The methods of linear elastic stress analysis may be used in the fracture analysis where its use is clearly conservative or supported by experimental evidence. Examples where “Fracture Mechanics” may be applied are for fillet welds or end of fatigue life crack propagation.
LE =	Ultimate load or loss of function load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate tensile strength of the actual part and the tested parts. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.
SF_{Min}=	Minimum safety factor (Subsection 3.9.5.3.6).

1 Do not use unless supporting data are provided to the NRC.

Table 3.9-6
Buckling Stability Limit for Safety Class Reactor Internal Structures Only

	Any One Of (No More Than one Required)	General Limit
a.	$\dfrac{\text{Permissible Load, LP}}{\text{Service Level A (normal) permissible load, PN}}$	$\leq \dfrac{2.25}{SF_{Min}}$
b.	$\dfrac{\text{Permissible Load, LP}}{\text{Stability analysis load, SL}}$	$\leq \dfrac{0.9}{SF_{Min}}$
c.	$\dfrac{\text{Permissible Load, LP}}{\text{Ultimate buckling collapse load from test, SET}}$	$\leq \dfrac{1.0}{SF_{Min}}$ [1]

Where:

LP = Permissible load under stated conditions of service levels A, B, C or D (normal, upset, emergency or faulted).

PN = Applicable service level A (normal) event permissive load.

SL = Stability analysis load. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects shall be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity on column members.

SET= Ultimate buckling collapse load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the actual part and the tested part. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

SF_{Min}=Minimum safety factor (Subsection 3.9.5.3.6)

1 Equation C will not be used unless supporting data are provided to the NRC.

Table 3.9-7 Fatigue Limit for Safety Class Reactor Internal Structures Only

Summation of fatigue damage usage following Miner hypotheses[1]:	
Cumulative Damage in Fatigue	**Limit for Service Levels A & B (Normal and Upset Conditions)**
Design fatigue cycle usage from analysis using the method of the ASME Code	≤1.0

1 Miner, M. A., “Cumulative Damage in Fatigue,” Journal of Applied Mechanics, Vol. 12, ASME, Vol. 67, pp A159-A164, September 1945.

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves[1]

MPL	System	Pump Page No.	Valve Page No.
B21	Nuclear Boiler		3.9-96
B31	Reactor Recirculation		3.9-99
C12	Control Rod Drive		3.9-100
C41	Standby Liquid Control	3.9-95	3.9-100
C51	Neutron Monitoring (ATIP)		3.9-101
D23	Containment Atmospheric Monitoring		3.9-101
E11	Residual Heat Removal	3.9-95	3.9-102
E22	High Pressure Core Flooder	3.9-95	3.9-107
E31	Leak Detection & Isolation		3.9-109
E51	Reactor Core Isolation Cooling	3.9-95	3.9-109
G31	Reactor Water Cleanup		3.9-114
G41	Fuel Pool Cooling & Cleanup		3.9-115
G51	Suppression Pool Cleanup		3.9-117
K17	Radwaste		3.9-117
P11	Makeup Water (Purified)		3.9-117
P21	Reactor Building Cooling Water	3.9-95	3.9-117
P24	HVAC Normal Cooling Water		3.9-122
P25	HVAC Emergency Cooling Water	3.9-95	3.9-123
P41	Reactor Service Water	3.9-95	3.9-126
P51	Service Air		3.9-127
P52	Instrument Air		3.9-127
P54	High Pressure Nitrogen Gas Supply		3.9-128
T22	Standby Gas Treatment		3.9-128
T31	Atmospheric Control		3.9-130
T49	Flammability Control		3.9-133
U41	Heating, Ventilating and Air Conditioning		3.9-133
Y52	Oil Storage and Transfer	3.9-95	3.9-134

See page 3.9-134 for notes.

1 This table responds to NRC Questions 210.47, 210.48 and 210.49 regarding provisions for inservice testing of safety-related pumps and valves within the scope of the ABWR Standard Plant in accordance with the ASME Code. The information is presented separately for each system for the MPL number.

Table 3.9-8 Inservice Testing Safety-Related Pumps and Valves[1] (Continued)

No.	Qty	Description (h) (i)	Safety Class (a)	Test Param (b)	Test Freq. (f)	Tier 2 Fig. (g)
System Pumps						
C41-C001	2	Standby Liquid Control System Pump	2	Pd,Vd, Q	3 mo	9.3-1
E11-C001	3	Residual Heat Removal System Pump	2	Pd, Pi, Q, Vv	3 mo	5.4-10 (Sh. 3, 4, 6)
E11-C002	3	Residual Heat Removal System fill pump (i1)	2	Pd,Pi, Vv	E10	5.4-10 (Sh. 3, 4, 6)
E22-C001	2	High Pressure Core Flooder pump	2	Pd,Pi, Q,Vv	3 mo	6.3-7(Sh. 2)
E51-C001	1	Reactor Core Isolation Cooling pump	2	N,Pd,Pi Q,Vv	3 mo	5.4-8(Sh. 1)
P21-C001	6	Reactor Building Cooling Water pump	3	Pd, Pi, Q, Vv	E10	9.2-1 (Sh. 1, 4, 7)
P25-C001	6	HVAC Emergency Cooling Water System pump	3	Pd, Pi, Q, Vv	E10	9.2-3 (Sh. 1, 2, 3)
P41-C001	6	Reactor Service Water System pump	3	Pd, Pi, Q, Vv	E10	9.2-7 (Sh. 1, 2, 3)
Y52-C001	6	Standby D/G Fuel Oil Transfer Pump	3	Pd, Pi, Q, Vv	3 mo	9.5-6

1 This table responds to NRC Questions 210.47, 210.48 and 210.49 regarding provisions for inservice testing of safety-related pumps and valves within the scope of the ABWR Standard Plant in accordance with the ASME Code. The information is presented separately for each system for the MPL number.

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)
B21 Nuclear Boiler System Valves								
F001	2	Feedwater line motor-operated valve (MOV)	2	B	P	S	RO	5.1-3 sh. 4
F002	2	Upstream (First) FW line check valve (h3)	2	A,C	A	L,S	RO	5.1-3 sh. 4
F003	2	FW line outboard check valve-air-operated (AO) (h1)	1	A,C	I,A	L,P, S	RO	5.1-3 sh. 4
F004	2	FW line inboard check valve (h1)	1	A,C	I,A	L,S	RO	5.1-3 sh. 4
F005	2	FW line inboard maintenance valve	1	B	P		E1	5.1-3 sh. 4
F006	2	RWCU (or CUW) System injection line check valve (h3)	2	A,C	A	L,S	RO	5.1-3 sh. 4
F007	2	RWCU (or CUW) System injection line MOV	2	B	P	S	E1	5.1-3 sh. 4
F008	4	Inboard main Steam isolation valve. (MSIV)(h1)	1	A	I,A	L,P S	RO 3 mo	5.1-3 sh. 3
F009	4	Outboard Main Steam isolation valve (MSIV)(h1)	1	A	I,A	L,P S	RO 3 mo	5.1-3 sh. 3
F010	18	Safety/Relief Valve (SRV) (h1), (h2)	1	A,C	A	R P,S	5 yr RO	5.1-3 sh. 2
F011	1	MSL bypass/drain line inboard isolation valve (h1)	1	A	I,A	L,P S	RO 3 mo	5.1-3 sh. 3
F012	1	MSL bypass/drain line outboard isolation valve (h1)	1	A	I,A	L,P S	RO 3 mo	5.1-3 sh. 3
F013	1	MSL warmup line valve	2	B	P		E1	5.1-3 sh. 3
F016	1	MSL downstream drain line header valve	2	B	P		E1	5.1-3 sh. 3
F017	1	MSL downstream drain line header bypass	2	B	A	P S	RO 3 mo	5.1-3 sh. 3
F018	1	RPV non-condensable gas removal line	1	B	P		E1	5.1-3 sh. 2
F019	1	RPV head vent inboard shutoff valve (h1)	1	B	A	P,S	RO	5.1-3 sh. 2

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)
F020	1	RPV head vent outboard shutoff valve (h1)	1	B	A	P,S	RO	5.1-3 sh. 2
F021	18	SRV discharge line vacuum breaker(h1)	3	C	A	R,S	RO	5.1-3 sh. 2
F022	18	SRV discharge line vacuum breaker (h1)	3	C	A	R,S	RO	5.1-3 sh. 2
F024	4	Inboard MSIV nitrogen supply line check valve (h1)	3	C	A	S	RO	5.1-3 sh. 3
F025	4	Outboard MSIV air supply line check valve (h1)	3	C	A	S	RO	5.1-3 sh. 3
F026	8	SRV ADS pneumatic supply line check valve (h1)	3	A,C	A	L,S	RO	5.1-3 sh. 2
F029	18	SRV pneumatic supply check valve (h1)	3	C	A	S	RO	5.1-3 sh. 2
F031	2	Inboard valve on the outboard FW line check valve test line	2	B	P		E1	5.1-3 sh. 4
F033	4	Inboard shutoff valve on the outboard MSIV test line	2	B	P		E1	5.1-3 sh. 3
F035	1	Inboard test line valve for the MSL bypass/drain valve	2	B	P		E1	5.1-3 sh. 3
F039	2	Inboard test line valve for the inboard FW line check valve	2	B	P		E1	5.1-3 sh. 4
F040	2	Outboard test line valve for the FW line check valve	2	B	P		E1	5.1-3 sh. 4
F500	2	Inboard test line valve for the first FW line check valve	2	B	P		E1	5.1-3 sh. 4
F503	2	Outboard drain line valve for the FW line check valve	2	B	P		E1	5.1-3 sh. 4
F508	4	Inboard MSIV accumulator A001 drain valve	3	B	P		E1	5.1-3 sh. 3
F509	4	Outboard MSIV accumulator A002 drain valve	3	B	P		E1	5.1-3 sh. 3
F510	8	SRV ADS accumulator A003 drain valve	3	B	P		E1	5.1-3 sh. 2
F511	18	SRV accumulator A004 drain valve	3	B	P		E1	5.1-3 sh. 2

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)
F700	4	Manual Isolation valve—RPV water level instrument reference leg line	2	B	P		E1	5.1-3 sh. 5,6
F701	4	Excess flow check valve—RPV water level instrument reference leg line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 5,6
F702	4	Manual isolation valve—RPV narrow range water level instrument sensing line	2	B	P		E1	5.1-3 sh. 5,6
F703	4	Excess flow check valve—RPV narrow range water level instrument sensing line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 5,6
F704	4	Manual isolation valve—RPV wide range water level instrument sensing line	2	B	P		E1	5.1-3 sh. 5,6
F705	4	Excess flow check valve—RPV wide range water level instrument sensing line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 5,6
F706	1	Root valve—Reactor well water level instrument sensing line	2	B	P		E1	5.1-3 sh. 5
F709	1	Manual isolation valve—RPV shutdown range water level instrument reference leg line	2	B	P		E1	5.1-3 sh. 2
F710	1	Excess flow check valve—RPV shutdown range water level instrument reference leg line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 2
F711	1	Manual isolation valve—RPV head seal leakage instrument line	2	B	P		E1	5.1-3 sh. 8
F712	1	Excess flow check valve to RPV head seal leakage instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 8
F713	4	Manual isolation valve—RPV above pump deck instrument line	2	B	P		E1	5.1-3 sh. 7

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)
F714	4	Excess flow check valve—RPV above pump deck instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 7
F715	4	Manual isolation valve—RPV below pump deck instrument line	2	B	P		E1	5.1-3 sh. 7
F716	4	Excess flow check valve—RPV below pump deck instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 7
F717	4	Manual Isolation valve—RPV above core plate instrument line	2	B	P		E1	5.1-3 sh. 7
F718	4	Excess flow check valve—RPV above core plate instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 7
F719	4	Manual isolation valve—RPV below core plate instrument line	2	B	P		E1	5.1-3 sh. 7
F720	4	Excess flow check valve—RPV below core plate instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 7
F723	4	Manual isolation valve—MSL flow restrictor instrument line	2	B	P		E1	5.1-3 sh. 2
F724	4	Excess flow check valve—MSL flow restrictor instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 2
F725	4	Manual isolation valve—MSL flow restrictor instrument line	2	B	P		E1	5.1-3 sh. 2
F726	4	Excess flow check valve—MSL flow restrictor instrument line (h3)	2	A,C	I,A	L,S	RO	5.1-3 sh. 2
F727	2	MSL PX instrument line inboard root valve	2	B	P		E1	5.1-3 sh. 3
B31 Reactor Recirculation Internal Pump Valves								
F008	10	Excess flow check valve RIP pump motor purge water line (h3)	2	A,C	I,A	L,S	RO	5.4-4 sh. 2
F010	10	RIP pump motor purge water supply line valve	2	B	P		E1	5.4-4 sh. 1

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)
F011	10	RIP inflatable pressurized water line inboard valve	2	B	P		E1	5.4-4 sh. 1
F013	10	RIP seal equalizing line valve	2	B	P		E1	5.4-4 sh. 1
F015	10	Manual maintenance valve—RIP pump motor purge water line	2	B	P		E1	5.4-4 sh. 2
F500	10	RIP cooling water HX vent line inboard valve	2	B	P		E1	5.4-4 sh. 1
F502	10	RIP drain line inboard valve	2	B	P		E1	5.4-4 sh. 1
F505	10	RIP cooling water HX shell drain line inboard valve	2	B	P		E1	5.4-4 sh. 1
C12 Control Rod Drive System Valves								
F719	4	Root valve charging line header pressure instrument line	2	B	P		E1	4.6-8 sh. 2
F720	4	Root valve charging line header pressure instrument line	2	B	P		E1	4.6-8 sh. 2
C41 Standby Liquid Control System Valves								
F001	2	SLCS storage tank outlet line MOV	2	B	A	P S	RO 3 mo	9.3-1
F002	2	SLCS pump suction line maintenance valve	2	B	P		E1	9.3-1
F003	2	SLCS pump discharge line relief valve	2	C	A	R	5 yr	9.3-1
F004	2	SLCS pump discharge line check valve	2	C	A	S	3 mo	9.3-1
F005	2	SLCS pump discharge line maintenance valve	2	B	P		E1	9.3-1
F006	2	SLCS pump injection valve MOV	2	A	I,A	L,P S	2 yr 3 mo	9.3-1
F007	1	SLCS injection line outboard check valve (h5)	2	A,C	I,A	L,S	2 yrs	9.3-1
F008	1	SLCS injection line inboard check valve (h1)	2	A,C	I,A	L,S	2 yr	9.3-1

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)
F018	1	SLCS storage tank sample line inboard shutoff valve	2	B	P		E1	9.3-1
F025	1	SLCS injection line test/vent line inboard valve	2	B	P		E1	9.3-1
F500	1	SLCS pump suction line drain line	2	B	P		E1	9.3-1
F501	2	SLCS pump discharge line drain line valve	2	B	P		E1	9.3-1
C51 Neutron Monitoring System Valves								
J004	3	Isolation valve assembly: ATIP ball valve	2	A	I,A	L,P S	RO 3 mo	7.6-1 SH. 3
		Index shear valve	2	A,D	A	X	RO	7.6-1 SH. 3
J011	3	Purge isolation valve	2	A,C	P	L,P	2yr	7.6-1 sh. 3
D23 Containment Atmospheric Monitoring System Valves								
F001	2	CAMS drywell pressure instrument line outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F004	2	CAMS drywell sample line outboard containment isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F005	2	CAMS drywell return line outboard containment isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F006	2	CAMS wetwell sample line outboard containment isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F007	2	CAMS wetwell return line outboard containment isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F008	2	CAMS rack drain line outboard containment isolation valve	2	A	I,A	L,P S	RO 3 mo	7.6-7 sh. 2
F009	2	CAMS drywell pressure instrument line outboard isolation valve	2	B	P		E1	7.6-7 sh. 2

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)
F010	2	CAMS drywell sample line outboard valve	2	B	P		E1	7.6-7 sh. 2
F011	2	CAMS drywell return line outboard valve	2	B	P		E1	7.6-7 sh. 2
F012	2	CAMS wetwell sample line outboard valve	2	B	P		E1	7.6-7 sh. 2
F013	2	CAMS wetwell return line outboard valve	2	B	P		E1	7.6-7 sh. 2
F014	2	CAMS rack drain line outboard containment isolation valve	2	B	P		E1	7.6-7 sh. 2
E11 Residual Heat Removal System Valves								
F001	3	Suppression pool suction valve	2	A	I,A	L,P S	RO 3 mo	5.4-10 sh.3,4,6
F002	3	RHR pump discharge line check valve	2	C	A	S	3 mo	5.4-10 sh.3,4,6
F003	3	RHR pump discharge line maintenance valve	2	B	P		E1	5.4-10 sh.3,4,6
F004	3	Heat Exchanger flow control valve	2	B	A	P S	2 yr 3 mo	5.4-10 sh.3,4,6
F005	1	RPV injection valve, Loop A (h6)	2	A	A	L,P S	RO CS	5.4-10 sh. 3
F005	2	RPV injection valve, Loop B & C (h6)	1	A	I,A	L,P S	RO CS	5.4-10 sh. 5,7
F006	1	RPV injection line check valve, Loop A	2	A,C	A	L,P S	RO 3 mo	5.5-10 sh. 3
F006	2	RPV injection line check valve, Loop B & C	1	A,C	I,A	L,P S	RO 3 mo	5.4-10 sh. 5,7
F007	2	RPV injection line inboard maint. valve	1	B	P		E1	5.4-10 sh. 5,7
F008	3	Suppression pool return line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-10 sh. 3,4,6
F009	3	Shutdown Cooling suction line maintenance valve	1	B	P		E1	5.4-10 sh. 2
F010	3	Shutdown Cooling suction line inboard isolation valve (h6)	1	A	I,A	L,P S	RO CS	5.4-10 sh. 2

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)
F011	3	Shutdown Cooling suction line outboard isolation valve (h6)	1	A	I,A	L,P S	RO CS	5.4-10 sh. 2
F012	3	Shutdown Cooling suction line adm. valve	2	B	A	P S	2 yr, 3 mo	5.4-10 sh.3,4,6
F013	3	Heat exchanger bypass flow control valve	2	B	A	P S	2 yr, 3 mo	5.4-10 sh.3,4,6
F014	3	Fuel Pool Cooling supply line inboard MOV	2	B	A	P S	2 yr, 3 mo	5.4-10 sh. 5,7
F015	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	3	Gate valve-line from Fuel Pool Cooling (FPC)	2	B	A	S	3 mo	5.4-10 sh. 2
F017	2	Drywell spray line inboard valve	2	A	I,A	L,P S	RO 3 mo	5.4-10 sh. 5,7
F018	2	Drywell spray line outboard valve	2	A	I,A	L,P S	RO 3 mo	5.4-10 sh. 5,7
F019	2	Wetwell spray line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-10 sh. 5,7
F020	3	RHR pump min flow bypass line check valve	2	C	A	S	3 mo	5.4-10 sh.3,4,6
F021	3	RHR pump min flow bypass line MOV	2	A	I,A	L,P S	2 yr 3 mo	5.4-10 sh.3,4,6
F022	3	Discharge line fill pump suction line valve	2	B	P		E1	5.4-10 sh.3,4,6
F023	3	Fill pump discharge line check valve	2	C	A	S	3 mo	5.4-10 sh.3,4,6
F024	3	Fill pump discharge line stop check valve	2	C	A	S	3 mo	5.4-10 sh.3,4,6
F025	3	Fill pump minimum flow line globe valve	2	B	P		E1	5.4-10 sh.3,4,6
F026	3	RHR pump suction to High Conductivity Waste (HCW)	2	B	P		E1	5.4-10 sh.3,4,6
F027	3	Bypass line around the check valve MPL E11-F002	2	B	P		E1	5.4-10 sh.3,4,6
F028	3	Heat exchanger outlet line relief valve	2	C	A	R	5 yr	5.4-10 sh.3,4,6

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)
F029	3	Inboard reactor well drain line valve	2	B	P		E1	5.4-10 sh.3,4,6
F030	3	Drain to radwaste valve	2	B	P		E1	5.4-10 sh.3,4,6
F031	3	Outboard reactor well drain line valve (to SP)	2	A	I,P	L,P	RO	5.4-10 sh.3,4,6
F032	3	Shutoff valve—line from MUWC	2	B	P		E1	5.4-10 sh.3,4,6
F033	3	Check valve in the line from MUWC	2	C	A	S	3 mo	5.4-10 sh.3,4,6
F034	1	RPV injection line vent/test line inboard valve, Loop A	2	B	P		E1	5.4-10 sh. 3
F034	2	RPV injection line vent/test line inboard valve, Loop B&C	1	B	P		E1	5.4-10 sh. 5,7
F036	1	Press equal valve around check valve E11-F006, Loop A	2	A	P		E1	5.4-10 sh. 3
F036	2	Press equal valve around check valve E11-F006, Loop B&C	1	A	P		E1	5.4-10 sh. 5,7
F037	3	Shutdown cooling suction line test line	1	A	P		E1	5.4-10 sh. 2
F039	3	Relief valve around the MOV MPL E11-F011	1	C	A	R	5 yr	5.4-10 sh. 2
F040	3	Shutoff valve—line from MUWC	2	B	P		E1	5.4-10 sh. 2
F041	3	Check valve—line from Make-Up Water Condenser (MUWC)	2	C	A	S	3 mo	5.4-10 sh. 2
F042	3	Shutdown Cooling Mode suction line relief valve	2	C	A		E1	5.4-10 sh.3,4,5
F043	3	HX outlet to the Sampling System (SS) test inboard valve	2	B	P		E1	5.4-10 sh.3,6,7
F045	1	HX outlet to the PASS—inboard valve	2	B	A	P S	2 yr 3 mo	5.4-10 sh. 3
F046	1	HX outlet to the PASS—outboard valve	2	B	A	P S	2 yr 3 mo	5.4-10 sh. 3

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)
F047	2	Shutoff—line from MUWC	2	B	P		E1	5.4-10 sh. 5,7
F048	2	Check valve—line from MUWC	2	C	P		E1	5.4-10 sh. 5,7
F049	2	Drywell spray line vent & test line inboard valve	2	B	P		E1	5.4-10 sh. 5,7
F051	3	Fill pump discharge line relief valve	2	C	A	R	5 yr	5.4-10 sh.3,4,6
F052	1	Drain line for the suppression pool	2	B	P		E1	5.4-10 sh. 4
F101	1	AC independent water addition input valve	2	B	A	S	3 mo	5.4-10 sh. 7
F102	1	AC independent water addition input valve	2	B	A	S	3 mo	5.4-10 sh. 7
F500	3	Heat exchanger inlet drain line inboard valve	2	B	P		E1	5.4-10 sh.3,4,6
F502	3	HX outlet line drain line inboard valve	2	B	P		E1	5.4-10 sh.3,4,6
F504	3	RPV injection line vent line inboard valve	2	B	P		E1	5.4-10 sh.3,4,7
F506	1	RPV injection line drain line inboard valve	2	B	P		E1	5.4-10 sh. 3
F506	2	RPV injection line drain line inboard valve	1	B	P		E1	5.4-10 sh. 5,7
F508	3	Shutdown Cooling suction line vent line valve	2	B	P		E1	5.4-10 sh. 2
F509	2	Vent valve—FPC return line	2	B	P		E1	5.4-10 sh. 5,7
F511	2	Drywell spray line inboard drain line valve	2	B	P		E1	5.4-10 sh. 5,7
F513	2	Drywell spray line inboard drain line valve	2	B	P		E1	5.4-10 sh. 5,7
F515	2	Wetwell spray line inboard drain line valve	2	B	P		E1	5.4-10 sh. 5,7
F517	3	RHR pump min flow line drain line inboard valve	2	B	P		E1	5.4-10 sh.3,4,6

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)
F700	3	RHR pump suction line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F701	3	RHR pump suction line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F702	3	RHR pump discharge line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F704	3	RHR pump discharge line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F706	3	RHR pump discharge line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F707	3	RHR pump discharge line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F708	3	FT MPL E11-FT008 instrument line inboard root valve	2	B	P		E1	5.4-10 sh.3,4,6
F709	3	FT MPL E11-FT008 instrument line outboard root valve	2	B	P		E1	5.4-10 sh.3,4,6

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)
F710	3	FT MPL E11-FT008 instrument line inboard root valve	2	B	P		E1	5.4-10 sh.3,4,6
F711	3	FT MPL E11-FT008 instrument line outboard root valve	2	B	P		E1	5.4-10 sh.3,4,6
F712	3	Shutdown Cooling Mode suction line pressure instrument line	2	B	P		E1	5.4-10 sh.3,4,6
F713	3	Fill pump suction line instrument line valve	2	B	P		E1	5.4-10 sh.3,4,6
F714	1	Discharge to radwaste flow instrument line	2	B	P		E1	5.4-10 sh. 4
F716	1	Discharge to radwaste flow instrument line	2	B	P		E1	5.4-10 sh. 4
F718	3	Fill pump discharge line check valve test point	2	B	P		E1	5.4-10 sh. 3,4,6
F720	3	Fill pump discharge line check valve test point	2	B	P		E1	5.4-10 sh. 3,4,6
E22 High Pressure Core Flooder System Valves								
F001	2	Condensate Storage Tank (CST) suction line MOV	2	B	A	P S	2 yr 3 mo	6.3-7 sh. 2
F002	2	CST suction line check valve	2	C	A	S	3 mo	6.3-7 sh. 2
F003	2	HPCF System injection valve (h6)	1	A	I,A	L,P S	RO CS	6.3-7 sh. 1
F004	2	HPCF System inboard check valve	1	A,C	I,A	L,P S	RO 3 mo	6.3-7 sh. 1
F005	2	Pump discharge line inboard maintenance valve	1	B	P		E1	6.3-7 sh. 1
F006	2	Suppression pool suction line MOV	2	A	I,A	L,P, S	RO 3 mo	6.3-7 sh. 2
F007	2	Suppression pool suction line check valve	2	C	A	S	3 mo	6.3-7 sh. 2
F008	2	Test return line inboard valve	2	B	A	P S	2 yr 3 mo	6.3-7 sh. 2
F009	2	Test return line outboard valve	2	A	I,A	L,P S	RO 3 mo	6.3-7 sh. 2

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)
F010	2	Pump minimum flow bypass line MOV	2	A	I,A	L,P S	RO 3 mo	6.3-7 sh. 2
F011	2	Bypass line shutoff valve around check valve E22-F002	2	B	P		E1	6.3-7 sh. 2
F012	2	HPCI pump suction line drain line to HCW	2	B	P		E1	6.3-7 sh. 2
F014	2	Pump discharge line fill line outboard check valve	2	C	A	S	3 mo	6.3-7 sh. 1
F015	2	Pump discharge line fill line outboard check valve	2	C	A	S	3 mo	6.3-7 sh. 1
F017	2	Pump discharge line test and vent line inboard valve	1	A	P		E1	6.3-7 sh. 1
F019	2	Pressure equalizing valve around check valve E22-F004	1	A	P		E1	6.3-7 sh. 1
F020	2	Suppression pool suction line relief valve	2	C	A	R	5 yr	6.3-7 sh. 2
F021	2	Pump discharge check valve	2	C	A	S	3 mo	6.3-7 sh. 2
F022	2	Suppression pool suction line test line valve	2	B	P		E1	6.3-7 sh. 2
F023	2	Pump discharge line test line valve	2	B	P		E1	6.3-7 sh. 2
F500	2	Pump discharge line high point vent inboard valve	2	B	P		E1	6.3-7 sh. 1
F502	2	Pump discharge line drywell test line inboard valve	2	B	P		E1	6.3-7 sh. 1
F700	2	Pump suction line pressure instrument line root valve	2	B	P		E1	6.3-7 sh. 2
F701	2	Pump suction line pressure instrument line root valve	2	B	P		E1	6.3-7 sh. 2
F702	2	Pump discharge line pressure instrument line inboard valve	2	B	P		E1	6.3-7 sh. 2
F704	2	Pump discharge line pressure instrument line inboard valve	2	B	P		E1	6.3-7 sh. 2
F705	2	Pump discharge line pressure instrument line outboard valve	2	B	P		E1	6.3-7 sh. 2

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)
F706	2	Pump discharge line flow instrument line inboard valve	2	B	P		E1	6.3-7 sh. 1
F707	2	Pump discharge line flow instrument line outboard valve	2	B	P		E1	6.3-7 sh. 1
F708	2	Pump discharge line flow instrument line inboard valve	2	B	P		E1	6.3-7 sh. 1
F709	2	Pump discharge line flow instrument line outboard valve	2	B	P		E1	6.3-7 sh. 1
E31 Leak Detection and Isolation System Valves								
F001	1	Drywell fission product monitoring line maintenance valve	2	B	P		E1	5.2-8 sh. 9
F002	1	Drywell fission product monitoring line inboard isolation valve	2	A	I,A	L,P S	RO 3 mo	5.2-8 sh. 9
F003	1	Drywell fission product monitoring line outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	5.2-8 sh. 9
F004	1	Drywell fission product monitoring line outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	5.2-8 sh. 9
F005	1	Drywell fission product monitoring line inboard isolation valve	2	A	I,A	L,P S	RO 3 mo	5.2-8 sh. 9
F006	1	Drywell fission product monitoring line maintenance valve	2	B	P		E1	5.2-8 sh. 9
F701	4	RCIC instrument line manual maintenance valve	2	B	P		E1	5.2-8 sh. 6
F702	4	RCIC instrument line isolation excess flow check valve(h3)	2	A, C	I,A	L,S	RO	5.2-8 sh. 6
F703	4	RCIC instrument line manual maintenance valve	2	B	P		E1	5.2-8 sh. 6
F704	4	RCIC instrument line isolation excess flow check valve (h3)	2	A, C	I, A	L,S	RO	5.2-8 sh. 6
E51 Reactor Core Isolation Cooling System Valves								
F001	1	Condensate Storage Tank (CST) suction line MOV	2	B	A	P S	2 yr 3 mo	5.4-8 sh. 1

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)
F002	1	CST suction line check valve	2	C	A	S	3 mo	5.4-8 sh. 1
F003	1	RCIC pump discharge line check valve	2	C	A	P S	2 yr 3 mo	5.4-8 sh. 1
F004	1	RCIC System injection valve (h6)	2	A	A	L,P S	RO CS	5.4-8 sh. 1
F005	1	RCIC System discharge line testable check valve	2	C	A	L,P S	RO 3 mo	5.4-8 sh. 1
F006	1	Suppression Pool (CSP) suction line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 1
F007	1	Suppression Pool (CSP) suction line check valve	2	C	A	S	3 mo	5.4-8 sh. 1
F008	1	RCIC System suppression pool test return line MOV	2	A	A	P S	2 yr 3 mo	5.4-8 sh. 1
F009	1	RCIC System suppression pool test return line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 1
F010	1	RCIC System minimum flow bypass line check valve	2	C	A	P S	2 yr 3 mo	5.4-8 sh. 1
F011	1	RCIC System minimum flow bypass line MOV	2	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 1
F017	1	RCIC pump suction line relief valve	2	C	A	R	5 yr	5.4-8 sh. 1
F018	1	Valve in the bypass line around check valve E51-F003	2	B	P		E1	5.4-8 sh. 1
F019	1	Pump discharge line test line valve	2	B	P		E1	5.4-8 sh. 1
F020	1	Pump discharge line test line valve	2	B	P		E1	5.4-8 sh. 1
F021	1	Pump discharge line fill line shutoff valve	2	B	P		E1	5.4-8 sh. 1
F022	1	Pump discharge line fill line check valve	2	C	A	S	3 mo	5.4-8 sh. 1
F023	1	Pump discharge line fill line check valve	2	C	A	S	3 mo	5.4-8 sh. 1
F024	1	Pump discharge line test line valve	2	B	P		E1	5.4-8 sh. 1

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)
F025	1	Pump discharge line test line valve	2	B	P		E1	5.4-8 sh. 1
F026	1	Valve in pressure equalizing line around E51-F005	2	B	P		E1	5.4-8 sh. 1
F027	1	Suppression Pool (S/P) suction line test line valve	2	B	P		E1	5.4-8 sh. 1
F028	1	Minimum flow bypass line test line valve	2	B	P		E1	5.4-8 sh. 1
F029	1	Minimum flow bypass line test line valve	2	B	P		E1	5.4-8 sh. 1
F033	1	Discharge line fill line bypass line shutoff valve	2	B	P		E1	5.4-8 sh. 3
F035	1	Steam supply line isolation valve	1	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 2
F036	1	Steam supply line isolation valve	1	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 2
F037	1	Steam admission valve	2	B	A	P, S	2 yr 3 mo	5.4-8 sh. 2
F038	1	Turbine exhaust line check valve (h3)	2	A, C	I,A	L S	2 yr RO	5.4-8 sh. 2
F039	1	Turbine exhaust line MOV	2	A	I,A	L,P S	2 yr 3 mo	5.4-8 sh. 1
F048	1	Steam supply line warm-up line valve	1	A	I,A	L,P S	RO 3 mo	5.4-8 sh. 2
F049	1	Steam supply line test line valve	2	B	P		E1	5.4-8 sh. 2
F050	1	Steam supply line test line valve	2	B	P		E1	5.4-8 sh. 2
F053	1	Turbine exhaust line test line valve	2	B	P		E1	5.4-8 sh. 1
F054	1	Turbine exhaust line vacuum breaker (h1)	2	C	A	R	RO	5.4-8 sh. 1

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)
F055	1	Turbine exhaust line vacuum breaker (h1)	2	C	A	R	RO	5.4-8 sh. 1
F056	1	Steam supply line drain pot drain line test line valve	2	B	P		E1	5.4-8 sh. 1
F057	1	Steam supply line drain pot drain line test drain line	2	B	P		E1	5.4-8 sh. 2
F500	1	Pump discharge line vent line valve	2	B	P		E1	5.4-8 sh. 1
F501	1	Pump discharge line vent line valve	2	B	P		E1	5.4-8 sh. 1
F502	1	Pump discharge line drain line valve	2	B	P		E1	5.4-8 sh. 1
F503	1	Pump discharge line drain line valve	2	B	P		E1	5.4-8 sh. 1
F700	1	Pump suction line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F701	1	Pump suction line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F702	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F703	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F704	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F705	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P		E1	5.4-8 sh. 1
F706	1	Pump discharge line flow instrument root valve	2	B	P		E1	5.4-8 sh. 1
F707	1	Pump discharge line flow instrument root valve	2	B	P		E1	5.4-8 sh. 1

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)
F708	1	Pump discharge line flow instrument root valve	2	B	P		E1	5.4-8 sh. 1
F709	1	Pump discharge line flow instrument root valve	2	B	P		E1	5.4-8 sh. 1
F710	1	Pump discharge line pressure instrument root valve	2	B	P		E1	5.4-8 sh. 1
F711	1	Pump discharge line pressure instrument root valve	2	B	P		E1	5.4-8 sh. 1
F716	1	Steam supply line pressure instrument root valve	2	B	P		E1	5.4-8 sh. 2
F717	1	Steam supply line pressure instrument root valve	2	B	P		E1	5.4-8 sh. 2
F718	1	Steam supply line drain pot instrument root valve	2	B	P		E1	5.4-8 sh. 2
F719	1	Steam supply line drain pot instrument root valve	2	B	P		E1	5.4-8 sh. 2
F720	1	Steam supply line drain pot instrument root valve	2	B	P		E1	5.4-8 sh. 2
F721	1	Steam supply line drain pot instrument root valve	2	B	P		E1	5.4-8 sh. 2
F722	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F723	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F724	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F725	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F726	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F727	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F728	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F729	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3

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)
F730	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F731	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F732	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F733	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
F734	1	Turbine exhaust pressure instrument root valve	2	B	P		E1	5.4-8 sh. 3
G31 Reactor Water Cleanup System Valves								
F001	1	Line inside containment from RHR system maintenance valve	1	B	P		E1	5.4-12 sh. 1
F002	1	CUW System suction line inboard isolation valve	1	A	I,A	L,P S	RO 3 mo	5.4-12 sh. 1
F003	1	CUW System suction line outboard isolation valve	1	A	I,A	L,P S	RO 3 mo	5.4-12 sh. 1
F017	1	CUW System RPV head spray line outboard isolation valve (h3)	1	A	I,A	L,P S	RO CS	5.4-12 sh. 1
F018	1	CUW System RPV head spray line inboard check valve (h1)	1	A, C	I,A	L, S	RO	5.4-12 sh. 1
F019	1	CUW System bottom head drain line maintenance valve	1	B	P		E1	5.4-12 sh. 1
F026	1	CUW System suction line shutoff valve	1	B	P	P,S	RO	5.4-12 sh. 1
F050	1	Test line off the suction line outboard isolation valve G31-F003	2	B	P		E1	5.4-12 sh. 1
F058	1	Test line off RPV head spray line outboard isolation valve	2	B	P		E1	5.4-12 sh. 1
F060	1	RPV bottom head drain line sample line test line valve	2	B	P		E1	5.4-12 sh. 1
F070	1	RPV bottom head drain line sample line maintenance valve	2	B	P		E1	5.4-12 sh. 1

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)
F071	1	RPV bottom head drain line sample line inboard valve	2	A	I,A	L,P S	RO 3 mo	5.4-12 sh. 1
F072	1	RPV bottom head drain line sample line outboard valve	2	A	I,A	L,P S	RO 3 mo	5.4-12 sh. 1
F500	1	CUW System bottom head drain line drain valve	2	B	P		E1	5.4-12 sh. 1
F501	1	CUW System bottom head drain line drain valve	2	B	P		E1	5.4-12 sh. 1
F700	2	CUW System suction line FE upstream instrument manual maintenance valve	2	B	P		E1	5.4-12 sh. 1
F701	2	CUW System suction line FE downstream instrument manual maintenance valve	2	B	P		E1	5.4-12 sh. 1
F702	2	CUW System suction line FE upstream instrument excess flow check valve (h3)	2	A, C	I,A	L,S,P	RO	5.4-12 sh. 1
F703	2	CUW System suction line FE downstream instrument excess flow check valve (h3)	2	A, C	I,A	L,S,P	RO	5.4-12 sh. 1
G41 Fuel Pool Cooling and Cleanup Valves								
F015	2	FPC system heat exchanger outlet line maintenance valve	3	B	P		E1	9.1-1 sh. 2
F016	1	FPC system discharge line to spent fuel pool check valve	3	C	A	S	3 mo	9.1-1 sh. 2
F017	1	FPC system discharge line to spent fuel pool maintenance valve	3	B	P		E1	9.1-1 sh. 2
F018	1	FPC system discharge line to spent fuel pool check valve	3	C	A	S	3 mo	9.1-1 sh. 2
F019	2	FPC system discharge line to spent fuel pool valve	3	B	P		E1	9.1-1 sh. 1
F020	2	FPC system discharge line to spent fuel pool check valve	3	C	A	S	3 mo	9.1-1 sh. 1
F022	1	FPC system discharge line to reactor well maintenance valve	3	B	P		E1	9.1-1 sh. 2
F023	1	FPC system discharge line to reactor well check valve (h7)	3	C	A	S	RO	9.1-1 sh. 2

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)
F091	1	FPC system supply line from SPCU check valve	3	C	A	S	3 mo	9.1-1 sh. 2
F093	1	FPC system RHR return line valve to FPC	3	B	P		E1	9.1-1 sh. 2
F094	1	FPC system RHR return line check valve to FPC (h7)	3	C	A	S	RO	9.1-1 sh. 2
F095	1	FPC system discharge line to spent fuel pool sample line	3	B	P		E1	9.1-1 sh. 2
F506	1	FPC system line valve from RHR-to-FPC line to LCW	3	B	P		E1	9.1-1 sh. 2

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

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
G51 Suppression Pool Cleanup System Valves								
F001	1	SPCU suction line inboard isolation valve	2	A	I,A	L,P S	RO 3 mo	9.5-1
F002	1	SPCU suction line outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	9.5-1
F006	1	SPCU return line isolation valve	2	A	I,A	L, P S	RO 3 mo	9.5-1
F007	1	SPCU return line isolation valve	2	A	I,A	L,P S	RO 3 mo	9.5-1
K17 Radwaste System Valves								
F003	1	Drywell LCW sump pump inboard discharge line isolation valve	2	A	I,A	L,P S	RO 3 mo	11A.2-2 sh. 29
F004	1	Drywell LCW sump pump outboard discharge line isolation valve	2	A	I,A	L,P S	RO 3 mo	11A.2-2 sh. 29
F103	1	Drywell HCW sump pump inboard discharge line isolation valve	2	A	I,A	L,P S	RO 3 mo	11A.2-2 sh. 30
F104	1	Drywell HCW sump pump outboard discharge line isolation valve	2	A	I,A	L,P S	RO 3 mo	11A.2-2 sh. 30
P11 Makeup Water (Purified) System Valves								
F141	1	Outboard isolation valve	2	A	I,P	L	RO	9.2-5 sh. 2
F142	1	Inboard isolation valve	2	A, C	I,P	L	RO	9.2-5 sh. 2
P21 Reactor Building Cooling Water System Valves								
F001	6	Pump discharge line check valve	3	C	A	S	E2	9.2-1 sh. 1,4,7
F002	6	Pump discharge line maintenance valve	3	B	P		E1	9.2-1 sh. 1,4,7
F003	9	Heat exchanger inlet line valve	3	B	P		E1	9.2-1 sh. 1,4,7
F004	9	Heat exchanger outlet line MOV	3	B	P	P	2 yr	9.2-1 sh. 1,4,7
F005	3	Cold water line to hot/cold water blender	3	B	P		E1	9.2-1 sh. 1,4,7
F006	3	Hot/cold water blender valve—cold water	3	B	A	S	E2	9.2-1 sh. 1,4,7

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

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F007	3	Hot/cold water blender outlet line valve	3	B	P		E1	9.2-1 sh. 1,4,7
F008	3	Hot/cold water blender cold water bypass line	3	B	P		E1	9.2-1 sh. 1,4,7
F009	3	Hot water line to hot/cold water blender	3	B	P		E1	9.2-1 sh. 1,4,7
F010	3	Hot/cold water blender valve—hot water	3	B	A	S	E2	9.2-1 sh. 1,4,7
F011	3	Hot/cold water blender hot water bypass line	3	B	P		E1	9.2-1 sh. 1,4,7
F012	3	Cooling water supply line to RHR System maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F013	3	Cooling water return line from RHR System MOV	3	B	A	P S	2 yr 3 mo	9.2-1 sh. 2,5,8
F014	3	Cooling water return line from RHR Hx maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F015	6	Pump suction line maintenance valve	3	B	P		E1	9.2-1 sh. 1,4,7
F016	3	Surge tank outlet line to RCW pump suction	3	B	P		E1	9.2-1 sh. 2,5,8
F017	3	Surge tank makeup water line from SPCU	3	B	P		E1	9.2-1 sh. 2,5,8
F018	3	Surge tank makeup water line from SPCU	3	B	P	P	2 yr	9.2-1 sh. 2,5,8
F019	3	Surge tank makeup water from MUWP	3	B	P	P	2 yr	9.2-1 sh. 2,5,8
F020	3	Surge tank makeup water line from MUWP	3	B	P		E1	9.2-1 sh. 2,5,8
F021	2	Chemical addition tank inlet line valve	3	B	P		E1	9.2-1 sh. 1,4
F022	2	Chemical addition tank outlet line valve	3	B	P		E1	9.2-1 sh. 1,4
F024	6	Cooling water supply line to HECW chiller maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F025	6	Cooling water supply line to HECW chiller PCV	3	B	A	S	E2	9.2-1 sh. 2,5,8

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

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F026	6	Cooling water supply line to HECW chiller maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F027	6	Cooling water line to HECW chiller bypass line	3	B	P		E1	9.2-1 sh. 2,5,8
F028	6	Cooling water return line from HECW chiller	3	B	P		E1	9.2-1 sh. 2,5,8
F029	2	Cooling water supply line to FPC Hx	3	B	P		E1	9.2-1 sh. 2,5
F030	2	Cooling water return line from FPC Hx	3	B	P		E1	9.2-1 sh. 2,5
F031	2	Cooling water supply line to FPC pump room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F032	2	Cooling water return line from FPC pump room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F033	2	Cooling water line to PCV Atmospheric Monitoring System clr	3	B	P		E1	9.2-1 sh. 2,5
F034	2	Return line from PCV Atmospheric Monitoring System clr	3	B	P		E1	9.2-1 sh. 2,5
F035	2	Cooling water supply line to SGTS room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F036	2	Cooling water return line from SGTS room air conditioner	3	B	P		E1	9.2-1 sh. 2,5
F039	3	Cooling water supply line to RHR equipment room air conditioner	3	B	P		E1	9.2-1 sh. 2,5,8
F040	3	Cooling water return line from RHR equipment room air conditioner	3	B	P		E1	9.2-1 sh. 2,5,8
F041	3	Cooling water supply line to RHR pump motor	3	B	P		E1	9.2-1 sh. 2,5,8
F042	3	Cooling water return line from RHR pump motor	3	B	P		E1	9.2-1 sh. 2,5,8
F043	3	Cooling water supply line to RHR pump mechanical seals	3	B	P		E1	9.2-1 sh. 2,5,8
F044	3	Cooling water return line from RHR pump mechanical seals	3	B	P		E1	9.2-1 sh. 2,5,8
F045	1	Cooling water supply line to RCIC equipment room air conditioner	3	B	P		E1	9.2-1 sh. 2

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

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F046	1	Cooling water supply line from RCIC equipment room air conditioner	3	B	P		E1	9.2-1 sh. 2
F047	2	Cooling water supply line to HPCF equipment room air conditioner	3	B	P		E1	9.2-1 sh. 5,8
F048	2	Cooling water supply line from HPCF equipment room air conditioner	3	B	P		E1	9.2-1 sh. 5,8
F049	2	Cooling water supply line to HPCF pump motor bearing	3	B	P		E1	9.2-1 sh. 5,8
F050	2	Cooling water return line from HPCF pump motor bearing	3	B	P		E1	9.2-1 sh. 5,8
F051	2	Cooling water supply line to HPCF pump mechanical seals	3	B	P		E1	9.2-1 sh. 5,8
F052	2	Cooling water return from HPCF pump mechanical seals	3	B	P		E1	9.2-1 sh. 5,8
F053	2	Surge tank outlet line to HECW System	3	B	P		E1	9.2-1 sh. 2,5,8
F055	6	Cooling water return line from Emergency Diesel Generator	3	B	A	P S	2 yr 3 mo	9.2-1 sh. 5,8
F056	3	Cooling water return line from Emergency Diesel Generator maintenance valve	3	B	P		E1	9.2-1 sh. 2,5,8
F061	3	Cooling water line Emergency Diesel Generators	3	B	P		E1	9.2-1 sh. 2,5,8
F071	6	Cooling water supply line–to non-essential coolers	3	B	P		E1	9.2-1 sh. 2,5,8
F072	6	Cooling water supply line–to non-essential coolers	3	B	A	P S	2 yr 3 mo	9.2-1 sh. 2,5,8
F075	2	Cooling water supply line to PCV outboard isolation valve (h3)	2	A	I,A	L,P S	RO, CS	9.2-1 sh. 3,6
F076	2	Cooling water supply line to PCV inboard check isolation valve (h1)	2	A, C	I,A	L,S	RO	9.2-1 sh. 3,6
F080	2	Cooling water return line from PCV inboard isolation valve (h1)	2	A	I,A	L,P,S	RO	9.2-1 sh. 3,6
F081	2	Cooling water return line from PCV outboard isolation valve (h3)	2	A	I,A	L,P S	RO, CS	9.2-1 sh. 3,6

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

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F083	3	Cooling water return line from non-essential coolers (h4)	3	C	A	S	RO	9.2-1 sh. 2,5,8
F084	3	Cooling water return line from containment bypass line	3	B	P		E1	9.2-1 sh. 2,5,8
F175	3	Cooling water supply to RHR System Hx pressure relief valve	3	C	A	R	5 yr	9.2-1 sh. 2,5,8
F195	2	Cooling water supply line to FPC heat exchanger	3	B	A	P S	2 yr 3 mo	9.2-1 sh. 2,5
F220	9	Bypass line around RCW System outlet line MOV	3	B	P		E1	9.2-1 sh. 1,4,7
F251	2	Cooling water supply line to PCV test line	2	B	P		E1	9.2-1 sh. 3,6
F252	2	Cooling water return line from PCV test line	2	B	P		E1	9.2-1 sh. 3,6
F501	9	Heat exchanger shell side vent line	3	B	P		E1	9.2-1 sh. 1,4,7
F502	9	Heat exchanger shell side drain line	3	B	P		E1	9.2-1 sh. 1,4,7
F503	3	Surge tank drain line to SD	3	B	P		E1	9.2-1 sh. 2,5,8
F601	3	Cooling water supply line to RHR System drain line to SD	3	B	P		E1	9.2-1 sh. 2,5,8
F602	3	Cooling water supply line to RHR System drain line to HCW	3	B	P		E1	9.2-1 sh. 2,5,8
F603	3	Cooling water return line from RHR Hx drain line to SD	3	B	P		E1	9.2-1 sh. 2,5,8
F604	3	Cooling water return line from RHR Hx drain line to HCW	3	B	P		E1	9.2-1 sh. 2,5,8
F701	6	Pump discharge line pressure instrument line	3	B	P		E1	9.2-1 sh. 1,4,7
F702	9	Hx discharge line sample line valve	3	B	P		E1	9.2-1 sh. 1,4,7
F703	3	Cooling water supply line pressure instrument line	3	B	P		E1	9.2-1 sh. 1,4,7
F704	3	Cooling water supply line sample valve	3	B	P		E1	9.2-1 sh. 1,4,7

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

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F705	3	Cooling water supply line elbow tap instrument root valve	3	B	P		E1	9.2-1 sh. 1,4,7
F706	3	Cooling water supply line elbow tap instrument root valve	3	B	P		E1	9.2-1 sh. 1,4,7
F707	3	Cooling water supply line to RHR System FT instrument root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F708	3	Cooling water supply line to RHR System FT instrument root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F709	3	Cooling water return line from RHR Hx sample valve	3	B	P		E1	9.2-1 sh. 2,5,8
F710	6	Pump suction line PX instrument root valve	3	B	P		E1	9.2-1 sh. 1,4,7
F711	6	Pump suction line pressure instrument root valve	3	B	P		E1	9.2-1 sh. 1,4,7
F712	3	Surge tank level instrument root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F713	3	Surge tank level instrument line root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F714	3	Surge tank level instrument line root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F717	3	Cooling water line to DG instrument line	3	B	P		E1	9.2-1 sh. 2,5,8
F718	3	Return water line from DG instrument line	3	B	P		E1	9.2-1 sh. 2,5,8
F719	3	Cooling water line to DG instrument line	3	B	P		E1	9.2-1 sh. 2,5,8
F720	3	Return water line from DG instrument line	3	B	P		E1	9.2-1 sh. 2,5,8
F721	3	Cooling water supply line to non-essential coolers FT instrument root valve	3	B	P		E1	9.2-1 sh. 2,5,8
F722	3	Cooling water supply line to non-essential coolers FT instrument root valve	3	B	P		E1	9.2-1 sh. 2,5,8
P24 HVAC Normal Cooling Water System Valves								
F053	1	HNCW supply line outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	9.2-2

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

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F054	1	HNCW supply line inboard isolation check valve (h1)	2	A, C	I,A	L,S	RO	9.2-2
F141	1	HNCW return inboard isolation valve (h1)	2	A	I,A	L,P,S	RO	9.2-2
F142	1	HNCW return outboard isolation valve	2	A	I,A	L,P S	RO 3 mo	9.2-2
P25 HVAC Emergency Cooling Water System Valves								
F001	6	Pump discharge line check valve	3	C	P	S	E2	9.2-3 sh. 1,2,3
F002	6	Pump discharge line maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F003	6	Chiller outlet line maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F004	2	Maintenance valve at HECW supply to MCR cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F005	2	HECW supply to MCR cooler Temperature Control Valve (TCV)	3	B	A	S	E2	9.2-3 sh. 1,2,3
F006	2	Maintenance valve at HECW supply to MCR cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F007	6	Maintenance valve at HECW supply to MCR cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F008	6	Maintenance valve at HECW return from MCR cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F009	6	Pump suction line maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F010	2	TCV bypass at HECW discharge to MCR cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F011	3	Pump suction line/discharge line PCV maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F012	3	Pump suction line/discharge line PCV	3	B	A	S	E2	9.2-3 sh. 1,2,3
F013	3	Pump suction line/discharge line PCV maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F014	3	Pump suction line/discharge line PCV bypass line	3	B	P		E1	9.2-3 sh. 1,2,3

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

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F015	3	Maintenance valve at HECW supply to C/B Essential Electrical Equipment Room Cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F016	3	HECW supply to C/B Essential Electrical Equipment Room cooler TCV	3	B	A	S	E2	9.2-3 sh. 1,2,3
F017	3	Maintenance valve at HECW supply to C/B Essential Electrical Equipment Room Cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F018	6	HECW supply to C/B Essential Electrical Equipment Room cooler maintenance valve	3	B	P		E1	9.2-3 sh. 1,2,3
F019	6	Maintenance valve at HECW return from C/B Essential Electrical Equipment Room Cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F020	3	TCV bypass valve at HECW supply to C/B Essential Electrical Equipment Room cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F021	3	Maintenance valve at HECW supply to DG zone cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F022	3	HECW supply to DG zone cooler TCV	3	B	A	S	E2	9.2-3 sh. 1,2,3
F023	3	Maintenance valve at HECW supply to DG zone cooler TCV	3	B	P		E1	9.2-3 sh. 1,2,3
F024	6	Maintenance valve at HECW supply to DG zone cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F025	6	Maintenance valve at HECW return from DG zone cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F026	3	TCV bypass valve at HECW supply to DG zone cooler	3	B	P		E1	9.2-3 sh. 1,2,3
F030	3	Chemical addition tank return valve from HECW	3	B	P		E1	9.2-3 sh. 1,2,3
F031	3	Chemical addition tank feed valve to HECW	3	B	P		E1	9.2-3 sh. 1,2,3
F050	2	Make-up Water Purified (MUWP) line to pump suction check valve	3	C	A	S	E2	9.2-3 sh. 1,2,3
F070	6	Pump discharge line drain valve	3	B	P		E1	9.2-3 sh. 1,2,3

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

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func (d)	Test Para (e)	Test Freq (f)	Tier 2 Fig.(g)
F400	6	Pump drain line valve	3	B	P		E1	9.2-3 sh. 1,2,3
F401	6	Pump bearing cooling water needle valve	3	B	P		E1	9.2-3 sh. 1,2,3
F402	3	Chiller outlet line sample line valve	3	B	P		E1	9.2-3 sh. 1,2,3
F700	6	Pump discharge line pressure instrument line root valve	3	B	P		E1	9.2-3 sh. 1,2,3
F701	6	FE P25-FE003 upstream instrument line root valve	3	B	P		E1	9.2-3 sh. 1,2,3
F702	6	FE P25-FE003 downstream instrument line root valve	3	B	P		E1	9.2-3 sh. 1,2,3
F703	6	Pump suction pressure instrument line root valve	3	B	P		E1	9.2-3 sh. 1,2,3
F704	6	Pump suction/discharge line Δp instrument line root valve	3	B	P		E1	9.2-3 sh. 1,2,3

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)
P41 Reactor Service Water System Valves								
F001	6	Pump discharge line check valve	3	C	A	S	E2	9.2-7 sh. 1,2,3
F002	6	Pump discharge line maintenance valve	3	B	P		E1	9.2-7 sh. 1,2,3
F003	9	Service water inlet valve to RCW System heat exchanger	3	A	A	P S	2 yr E2	9.2-7 sh. 1,2,3
F004	6	Service water inlet valve to service water strainer	3	B	P	P	2 yr	9.2-7 sh. 1,2,3
F005	9	Service water outlet valve from RCW heat exchanger	3	A	A	P S	2 yr E2	9.2-7 sh. 1,2,3
F006	6	Service water strainer blowout valve	3	B	P	P	2 yr	9.2-7 sh. 1,2,3
F007	9	Supply line from Potable Water check valve	3	C	P		E1	9.2-7 sh. 1,2,3
F008	9	Supply line from Potable Water check valve	3	C	P		E1	9.2-7 sh. 1,2,3
F009	9	Supply valve from Potable Water System	3	B	A	P S	2 yr E2	9.2-7 sh. 1,2,3
F010	9	RCW Hx tube side (service water side) relief valve	3	C	P	R	5 yr	9.2-7 sh. 1,2,3
F011	9	Bypass line around RCW Hx outlet line outlet valve MOV P41-F005	3	B	P		E1	9.2-7 sh. 1,2,3
F012	9	Service water sampling valve	3	B	P		E1	9.2-7 sh. 1,2,3
F013	6	Service water strainer outlet valve	3	A	A	P S	2 yr E2	9.2-7 sh. 1,2,3
F014	3	Common service water strainer outlet valve	3	A	A	P S	2 yr E2	9.2-7 sh. 1,2,3
F015	3	Discharge line to discharge canal MOV	3	A	A	P S	E1 E2	9.2-7 sh. 1,2,3
F101	3	RSW line to HVAC Air Conditioning Condenser Manual Isolation valves	3	B	P		E1	9.2-7 sh 1,2,3.
F102	3	RSW blowdown line to Main Cooling Reservoir MOV	3	B	A	P S	2 yr 3 mo	9.2-7 sh. 1,2,3

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)
F109	3	RSW cold bypass to cooling tower basin MOV	3	B	A	P S	2 yr 3mo	9.2-7 sh. 1,2,3
F110	6	RSW return to cooling tower	3	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	B	P		E1	9.2-7 sh. 1
F114/ F117	2	Makeup water to UHS basin Check valves	3	C	A	S	3 mo	9.2-7 sh. 1
F115	1	Makeup water to UHS basin MOV	3	B	A	P S	2 yr 3mo	9.2-7sh. 1
F501	9	RCW Hx shell side drain valve to SWSD	3	B	P		E1	9.2-7 sh. 1,2,3
F502	9	RCW Hx shell side vent valve to SWSD	3	B	P		E1	9.2-7 sh. 1,2,3
F503	9	RCW Hx shell side drain valve to SWSD	3	B	P		E1	9.2-7 sh. 1,2,3
F504	9	RCW Hx shell side vent valve to SWSD	3	B	P		E1	9.2-7 sh. 1,2,3
F701	6	Pump discharge line pressure instrument line	3	B	P		E1	9.2-7 sh. 1,2,3
F702	3	Service water supply pressure instrument root valve	3	B	P		E1	9.2-7 sh. 1,2,3
F703	6	ΔP across service water strainer upstream instrument root valve	3	B	P		E1	9.2-7 sh. 1,2,3
F704	6	ΔP across service water strainer downstream instrument root valve	3	B	P		E1	9.2-7 sh. 1,2,3
F705	9	Service water ΔP across RCW Hx upstream instrument root valve	3	B	P		E1	9.2-7 sh. 1,2,3
F706	9	Service water ΔP across RCW Hx downstream instrument root valve	3	B	P		E1	9.2-7 sh. 1,2,3
P51 Service Air System Valves								
F131	1	Outboard isolation manual valve	2	A	I,P	L	RO	9.3-7
F132	1	Inboard isolation check valve (h1)	2	A,C	I,A	L,S	RO	9.3-7
P52 Instrument Air System Valves								
F276	1	Outboard isolation valve (h3)	2	A	I,A	L,P,S	RO	9.3-6
F277	1	Inboard isolation check valve (h3)	2	A,C	I,A	L,P,S	RO	9.3-6

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)
P54 High Pressure Nitrogen Gas Supply System Valves								
F002	4	Nitrogen bottles N_2 supply line valve	3	B	P		E1	6.7-1
F003	2	Nitrogen bottles N_2 supply line MOV	3	B	A	P S	2 yr 3 mo	6.7-1
F004	2	N_2 bottle supply line PCV maintenance valve	3	B	P		E1	6.7-1
F005	2	N_2 bottle supply line PCV	3	B	A		E1	6.7-1
F006	2	N_2 bottle supply line PCV maintenance valve	3	B	P		E1	6.7-1
F007	2	Safety grade N_2 supply line isolation valve	2	A	I,A	L,P S	RO 3 mo	6.7-1
F008	2	Safety grade N_2 supply line isolation check valve (h1)	2	A,C	I,A	L, S	RO	6.7-1
F009	8	Safety grade N_2 supply line to SRV	3	B	P		E1	6.7-1
F010	2	Bypass line around the N_2 bottle supply line PCV	3	B	P		E1	6.7-1
F011	2	N_2 bottle supply line relief valve	3	C	A	R	5 yr	6.7-1
F012	2	MOV at safety/non-safety boundary	3	A	A	P S	2 yr 3 mo	6.7-1
F200	1	Non-safety N2 supply line isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.7-1
F209	1	Non-safety N2 supply line isolation check valve	2	A,C	I,A	L,S	RO	6.7-1
T22 Standby Gas Treatment System Valves								
F001	2	Fuel handling floor inlet butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 1
F002	2	Filter train inlet butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 1
F003	2	Filter train exhaust gravity damper	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 2,3
F004	2	Filter train exhaust butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 2,3
F005	2	Cooling fan butterfly valve	3	B	A	P S	2 yr 3 mo	6.5-1 sh. 2,3
F006	2	Filter train R112 injection line valve	3	B	P		E1	6.5-1 sh. 2,3

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)
F007	2	Filter train DOP injection line valve to pre HEPA filter	3	B	P		E1	6.5-1 sh. 2,3
F008	2	Filter train DOP sampling line valve downstream of pre HEPA	3	B	P		E1	6.5-1 sh. 2,3
F009	2	Filter train DOP sampling line valve downstream of pre HEPA	3	B	P		E1	6.5-1 sh. 2,3
F010	2	Filter train DOP injection line valve downstream of charcoal absorbent	3	B	P		E1	6.5-1 sh. 2,3
F011	2	Filter train DOP sampling line valve downstream of charcoal absorbent	3	B	P		E1	6.5-1 sh. 2,3
F012	2	Filter train DOP sampling line valve downstream of after HEPA	3	B	P		E1	6.5-1 sh. 2,3
F014	2	SGTS sample line valve	3	B	P		E1	6.5-1 sh. 2,3
F015	2	PRM discharge to stack valve	3	B	P		E1	6.5-1 sh. 2,3
F500	2	Filter unit vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F501	2	Filter unit drain line valve	3	B	P		E1	6.5-1 sh. 2,3
F504	2	Filter unit vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F505	2	Exhaust fan vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F506	2	Filter train vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F507	2	Filter train vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F508	2	Filter train vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F509	2	Filter train vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F510	2	Filter train vent line valve	3	B	P		E1	6.5-1 sh. 2,3
F511	2	Exhaust stack drain line valve	3	B	P		E1	6.5-1 sh. 2,3

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)
F700	2	Filter unit demister dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F701	2	Filter unit demister dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F705	2	Filter train prefilter dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F706	2	Filter train prefilter dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F707	2	Filter train preHEPA dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F708	2	Filter train preHEPA dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F709	2	Filter train charcoal absorber dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F710	2	Filter train charcoal absorber dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F711	2	Filter train after HEPA dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F712	2	Filter train after HEPA dp instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F713	2	Filter train exhaust flow instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
F714	2	Filter train exhaust flow instrument line valve	3	B	P		E1	6.5-1 sh. 2,3
T31 Atmospheric Control System Valves								
F001	1	Purge supply line outboard isolation valve (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F002	1	Drywell purge line supply inboard isolation valve (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F003	1	Wetwell purge supply line inboard isolation valve (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F004	1	Drywell purge exhaust line inboard isolation valve (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F005	1	Drywell purge exhaust line bypass line valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-39 sh. 1
F006	1	Wetwell purge exhaust line inboard isolation valve (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1

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)
F007	1	Wetwell overpressure line valve (h2)	2	A	I,P	L, P S	2 yr RO	6.2-39 sh. 1
F008	1	Containment exhaust line to SGTS (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F009	1	Containment exhaust line to R/B HVAC (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F010	1	Wetwell overpressure line valve (h2)	2	A	I,P	L, P S	2 yr RO	6.2-39 sh. 1
F011	1	Containment exhaust line to SGTS (h2)	2	A	I,A	L, P S	2 yr RO	6.2-39 sh. 1
F025	1	Purge supply line from outboard containment isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-39 sh. 1
F039	1	N_2 makeup line from outboard containment isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-39 sh. 1
F040	1	N_2 makeup line from to drywell inboard isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-39 sh. 1
F041	1	N_2 makeup line from to wetwell inboard isolation valve	2	A	I,A	L, P S	2 yr 3 mo	6.2-39 sh. 1
F044	8	Drywell/wetwell vacuum breaker valve	2	C	A	P,R	RO	6.2-39 sh. 2
F050	1	Purge supply line from test line valve	2	B	P		E1	6.2-39 sh. 1
F051	1	Purge exhaust line test line valve	2	B	P		E1	6.2-39 sh. 1
F054	1	Makeup line test line valve	2	B	P		E1	6.2-39 sh. 1
F055	1	Drywell personnel air lock hatch test line valve	2	B	P		E1	6.2-39 sh. 2
F056	1	Wetwell personnel air lock hatch test line valve	2	B	P		E1	6.2-39 sh. 2
F057	1	Overpressure protection test line valve	2	B	P		E1	6.2-39 sh. 1
F058	1	Overpressure protection test line valve	3	B	P		E1	6.2-39 sh. 1
F059	1	Overpressure protection test line valve	3	B	P		E1	6.2-39 sh. 1

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)
F700	1	FE instrument line valve	2	B	P		E1	6.2-39 sh. 1
F701	1	FE instrument line valve	2	B	P		E1	6.2-39 sh. 1
F702	1	FE instrument line valve	2	B	P		E1	6.2-39 sh. 1
F703	1	FE instrument line valve	2	B	P		E1	6.2-39 sh. 1
F730	1	Drywell pressure instrument line isolation valve	2	B	P		E1	6.2-39 sh. 2
F731	1	Drywell pressure instrument line isolation valve	2	A	I,P	L,P	RO	6.2-39 sh. 2
F732	2	Drywell pressure instrument line valve	2	B	P		E1	6.2-39 sh. 2
F733	2	Drywell pressure instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F734	4	Drywell pressure instrument line valve	2	B	P		E1	6.2-39 sh. 2
F735	4	Drywell pressure instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F736	2	Wetwell pressure instrument line valve	2	B	P		E1	6.2-39 sh. 2
F737	2	Wetwell pressure instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F738	4	Suppression pool water level instrument line valve	2	B	P		E1	6.2-39 sh. 2
F739	4	Suppression pool water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F740	4	Suppression pool water level instrument line valve	2	B	P		E1	6.2-39 sh. 2
F741	4	Suppression pool water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F742	4	Suppression pool water level instrument line valve	2	B	P		E1	6.2-39 sh. 2

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)
F743	4	Suppression pool water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F744	4	Suppression pool water level instrument line valve	2	B	P		E1	6.2-39 sh. 2
F745	4	Suppression pool water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F800	2	Drywell water level instrument line isolation valve	2	B	P		E1	6.2-39 sh. 2
F801	2	Drywell water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F802	2	Drywell water level instrument line valve	2	B	P		E1	6.2-39 sh. 2
F803	2	Drywell water level instrument line isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
F804	2	DW/WW differential pressure instrument line valve	2	B	P		E1	6.2-39 sh. 2
F805	2	DW/WW differential pressure instrument isolation valve	2	A	I,P	L, P	RO	6.2-39 sh. 2
D001	1	Wetwell overpressure rupture disk	2	D	I,P	Rplc.	5 yr	6.2-39 sh. 1
D002	1	Wetwell rupture disk	3	D	I,P	Rplc.	5 yr	6.2-39 sh. 1
U41 Heating, Ventilating and Air Conditioning System Valves								
F001	2	Secondary containment supply isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F002	2	Secondary containment exhaust isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F003	3	Secondary Containment divisional supply isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F004	3	Secondary Containment divisional exhaust isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1
F007	4	MCR area HVAC bypass line isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2
F008	4	MCR area HVAC supply isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2

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)
F009	4	MCR area HVAC emergency HVAC supply	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2
F010	4	MCR area HVAC exhaust isolation valve	2	B	A	P S	2 yr 3 mo	9.4-3 sh. 1,2
Y52 Oil Storage Transfer System Valves								
F001	6	D/G transfer pump discharge line check valve	3	C	A	S	3 mo	9.5-6
F002	3	D/G transfer pump discharge line relief valve	3	C	A	R	5 yr	9.5-6
F003	3	D/G transfer pump discharge line ball (plug) valve	3	B	P		E1	9.5-6
F004	3	D/G fuel oil day tank return to storage tank valve	3	B	P		E1	9.5-6
F501	3	D/G transfer pump discharge line drain valve	3	B	P		E1	9.5-6
F502	3	D/G transfer pump discharge line vent valve	3	B	P		E1	9.5-6

Notes:

(a) 1, 2, or 3—Safety Classification, Subsection 3.2.3.

(b) Pump test parameters per ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM--1987, Part 6:
N - Speed
Pd- Discharge Pressure
Pi - Inlet Pressure
Q - Flow Rate
Vd -Peak-to-peak vibration displacement
Vv -Peak vibration velocity

(c) A, B, C or D—Valve category per ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, Parts 1 and 10.

(d) Valve function:
 I - Primary containment isolation, Subsection 6.2.4
 A or P -Active or passive per ASME Code in (c) above (Part 10, Paragraph 1.3).

(e) Valve test parameters per ASME Code in (c) above:
 L - Leakage rate (Part 10, Paragraph 4.2.2, Tier 2 Table 6.2-7 for valves with function I in (d) above)
 P - Local position verification (Part 10, Paragraph 4.1)
 R - Relief valve test including visual examination, set pressure and seat tightness testing (Part 10, Paragraph 4.3.1 and Part 1, Paragraph 1.3.3 and 1.3.4)
 S - Stroke exercise Category A or B (Part 10, Paragraphs 4.2.1.1, 4.2.1.2)
 Category C (Part 10, Paragraphs 4.3.2.1, 4.3.2.2, 4.3.2.4)
 X - Explosive charge test (Part 10, Paragraph 4.4.1)

(f) Pump or valve test exclusions, alternatives and frequency per ASME code in (b) or (c) above or Appendix I:
 CS -Cold shutdown
 RO - Refueling outage and/or no case greater than two years.
 E1 -Used for operating convenience (i.e., passive vent, drain, instrument test, maintenance valves, or a system control valve). Tests are not required (Part 10, Paragraph 1.2).
 E2 -In regular use. Test frequency is not required provided the test parameters are analyzed and recorded at an operation interval not exceeding three months.
 Category A or B, Stroke (Part 10, Paragraph 4.2.1.5).
 Category C, Stroke (Part 10, Paragraph 4.3.2.3).
 E3 -Operability test every six months. Set pressure and leak test every refueling outage. (Part 1, Paragraph 1.3.4.3).
 E10 -In regular use. Test frequency is not required provided the test parameters are recorded at least once every three months of operation (Part 6, Paragraph 5.3)
 E11 -Lacking required fluid inventory. Test shall be performed at least once every two years with required fluid inventory provided (Part 6, Paragraph 5.5).

(g) Piping and instrument symbols and abbreviations are defined in Figure 1.7-1. Figure page numbers are shown in parenthesis.

(h) Reasons for code defined testing exceptions (Part 10, Paragraphs 4.2.1.2, 4.3.2.2).
 (h1) Inaccessible inerted containment and/or steam tunnel radiation during power operations.
 (h2) Avoids valve damage and impacts on power operations.
 (h3) Avoids impacts on power operations.

(h4) A temporary crosstie is necessary to carry the ongoing cooling loads. A permanent crosstie would violate divisional separation.

(h5) Avoids cold/hot water injection to RPV during power operations.

(h6) Maintain pressure isolation during normal operation.

(h7) Inventory available only during refueling outage.

(h8) Not Used

(h9) Test connection size is insufficient for full flow test during operation. Therefore, test part stroke during plant operation and full stroke during refueling outage. A test connection size which would be sufficient for full flow tests would pressurize the secondary containment beyond specified limits, thus affecting power operation.

(i) Summary justification for code exemption request (Part 6, Paragraph 5.2, or Part 10, Paragraph 6.2).

(i1) The piping is maintained full by a small fraction of the pump's flow capacity. These pumps may be a constant speed centrifugal type with a cooling by-pass loop. Normal operation will be near minimum flow in the flat or constant region of the pressure/flow performance curve. Therefore, a flow measurement would not be useful. The pumps will be designed and analyzed to withstand low flow operation without significant degradation.

Table 3.9-9 Reactor Coolant System Pressure Isolation Valves

Standby Liquid Control System	
C41-F006 A,B	Injection Valves
C41-F008	Inboard Check Valve
Residual Heat Removal System	
E11-F005 A,B,C	Injection Valve Loops A,B&C
E11-F006 A,B,C	Testable Check Valve A,B&C
E11-F010 A,B,C	Shutdown Cooling Inboard Suction Isolation Valve Loops A,B&C
E11-F011 A,B,C	Shutdown Cooling Outboard Suction Isolation Valve Loops A,B&C
High Pressure Core Flooder System	
E22-F003 B,C	Injection Valve Loops B&C
E22-F004 B,C	Testable Check Valve Loops B&C
Reactor Core Isolation Cooling System	
E51-F004	Injection Valve
E51-F005	Testable Check Valve

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds

Component	Weld Type	NDE Requirements
Class 1 Components (1) (2) (3)		
Vessel	Category A (Longitudinal)	RT plus MT or PT
Vessel, Pipe, Pump, Valve	Category B (Circumferential)	RT plus MT or PT
Pipe, Pump, Valve	Butt weld Fillet and socket welds	RT plus MT or PT MT or PT
Vessels (9)	Category C and similar welds	RT plus MT or PT, RT must be multiple exposure
	Partial penetration and fillet welds	MT or PT on all accessible surfaces
Vessels (9) & Branched Connections	Category D	
	a) Butt welds, all	RT plus MT or PT
	b) Corner welded nozzles	RT plus MT or PT
	c) Corner welded branch and piping connection exceeding 100A nominal diameter	RT plus MT or PT
	d) Corner welds branch and piping 100A and less	MT or PT
	e) Weld buildup deposits at openings	UT plus a, b, c above if connected to nozzle or pipe
	f) Partial penetration	MT or PT progressive and final surface
	g) Oblique full penetration branch and piping connections	RT or UT plus MT or PT, In addition, UT of weld, fusion zone, and parent metal beneath attachment surface.
General	Fillet, partial penetration, socket welds	MT or PT
General	Structural attachment welds	MT or PT
Special Welds	1) Specially designed seals	MT or PT
	2) Weld metal cladding	PT
	3) Hard surfacing	PT
	a) Valves 100A or less	None
	4) Tube-tube sheet welds	PT
	5) Brazed joints	VT

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds (Continued)

Component	Weld Type	NDE Requirements
Class 2 Components (1) (2) (4)		
Vessel	Category A (Longitudinal)	
	a) Either of the members exceeds 4.8 mm	RT
	b) Each member 4.8 mm or less	MT, PT, or RT
Pipe, Pump, Valve	Longitudinal	RT
Vessel	Category B (Circumferential)	
	a) Either of the members exceeds 4.8 mm	RT
	b) Each member 4.8 mm or less	MT, PT, or RT
Pipe, Pump, Valve	Circumferential	
	a) Butt weld	RT
	b) Partial penetration and fillet weld	MT or PT
Vessels (9) and Similar Joints in Other Components	Category C	
	a) Corner joints, either of the members exceeds 4.8 mm	RT
	b) Each member 4.8 mm or less	MT, PT, or RT
	c) Partial penetration and fillet welds	MT or PT
Vessels (9) and Similar Welds in Other Components	Category D	
	a) Full penetration joints when either members exceed 4.8 mm of thickness	RT
	b) Full penetration corner joints when either member exceeds 4.8 mm	UT or RT
	c) Both members 4.8 mm or less	MT or PT
	d) Partial penetration and fillet weld joints	MT or PT
Branch Con. and Nozzles in Pipe, Valve Pump	a) Nominal size exceed 100A	RT
	b) Nominal size100A or smaller	MT or PT (External and accessible internal surfaces)
Vessels Designed to NC-3200	Category A	RT
	Category B	RT
	Category C, Butt weld	RT
	Category C, Full penetration corner	UT or RT
	Category C, Partial penetration corner and fillet welds	MT or PT both sides
	Category D, Full penetration (6)	RT
	Category D, Partial penetration	MT or PT both sides
	Fillet, partial penetration, socket, and structural attachment welds	MT or Pt

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds (Continued)

Component	Weld Type	NDE Requirements
Special Welds	a) Specially designed seals	MT or PT
	b) Weld metal cladding	MT or PT
	c) Hard surfacing	PT
	d) Hard surfacing for valves with inlet connection 100A nominal pipe size or less	None
	e) Tube-tube sheet welds	PT
	f) Brazed joints	VT
Storage Tanks (Atmospheric)	a) Side joints	RT
	b) Roof and roof-to-sidewall	VT
	c) Bottom joints	Vacuum box testing of at least 20.6 kPaG
	d) Bottom to sidewall	Vacuum box plus MT or PT
	e) Nozzle to tank side	MT or PT
	f) Nozzle to roof	VT
	g) Joints in nozzles	RT
	h) Others	Similar welds in vessels
Storage Tanks (0-103.42 kPaG)	a) Sidewall	RT
	b) Roof	RT
	c) Roof-to-sidewall	RT, if not possible, MT or PT
	d) Bottom & bottom-to-side	Vacuum box method plus MT or PT
	e) Nozzle tank	MT or PT
	f) Joints to nozzles	RT
	g) Others	Same as similar vessel joints

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds (Continued)

Component	Weld Type	NDE Requirements
Class 3 Components (1) (2) (5)		
Vessels	Category A (Longitudinal)	
	1) a) Thickness exceeding the limits of Table ND-5211.2-1	RT
	b) Welds based on joint efficiency permitted by ND-3352.1	RT
	c) Butt welds in nozzles attached to vessels in a or b above	RT
	2) Welds not included in 1 above	Spot RT each 15.24m of weld. Additional RT to cover each welders work.
	3) Nonferrous vessels exceeding 9.5 mm	RT
Pump, Valve, Pipe	Pipes greater than 50A nominal size	RT, MT, or PT
	Pumps & valves greater than 50A nominal	According to the product form
Vessel	Category B (Circumferential)	
	1) a) Thickness exceeds Table ND-5211.2-1 for ferrous metals	RT
	b) Thickness exceeds 9.5 mm for nonferrous metals	RT
	c) Joint efficiency according to ND-3352.1(a)	RT
	d) Attachments to vessels and exceeds nominal pipe size 254 mm or thickness 28.6 mm	RT
	2) Welds not involved in 1 above	RT 152 mm long sections plus the intersections of Category A welds
Pipe, Pump and Valve	Greater than 50A nominal pipe size	RT, PT, or MT
Vessel	Category C	
	1) a) Thickness exceeds Table ND-5211.2-1 or ND-5211.3	RT
	b) Attachments exceed 250A NPS or 28.6 mm wall thickness	RT
	2) Welds not involved in 1 above	Spot RT to cover each welders work
Pipe, Pump, Valves	Greater than 50A nominal pipe size	RT, PT, or MT

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds (Continued)

Component	Weld Type	NDE Requirements
Vessel	Category D	
	1) Full penetration butt welds designed for joint efficiency per ND.3352.1(a)	RT
	2) In nozzles or communicating chambers attached to vessels or heads requiring full RT	RT
	3) Welds not covered by 1 and 2 above	Spot RT to cover each welders work
Pipe, Pump and Valve	Greater than 50A nominal pipe size	RT, PT, or MT
Special Welds	a) Weld metal cladding	PT
	b) Hard surfacing	PT
	i) Hard surfacing for valves with inlet connection 100A nominal pipe size or less	None
	c) Tube-tube sheet welds	PT
	d) Brazed joints	VT
Storage Tanks (Atmospheric)	a) Sidewall joints	Same as Category A or B vessel joints
	b) Roof and roof-to-sidewall	VT
	c) Bottom joints	Vacuum box testing of at least 20.6 kPaG, or PT or MT plus VT during pressure test
	d) Bottom to sidewall	Same as bottom joints
	e) Nozzle to tank side	MT or PT
	f) Nozzle to roof	VT
	g) Joints in nozzles ex. roof nozzles	MT or PT
	h) Others	Similar welds in vessels
Storage Tanks (0–103.42 kPaG)	a) Sidewall	Same as Category A or B vessel joints
	b) Roof	Same as Category A vessel joints
	c) Roof-to-sidewall	Same as above, if possible, or MT or PT
	d) Bottom to bottom-to-side	Vacuum box testing at least 20.6 kPaG, or PT or MT plus VT during pressure test
	e) Nozzle to tank	MT or PT
	f) Joints in nozzles	MT or PT
	g) Others	Same as similar vessel joints

Table 3.9-10 Welding Activities and Weld Examination Requirements for ASME Code, Section III Welds (Continued)

Component	Weld Type	NDE Requirements
Components Supports (1) (2) (7)		
Class 1 Supports	Primary member, full penetration butt welds All other welds Secondary member welds	RT MT or PT VT
Class 2 and MC Supports	Primary member, full penetration butt welds Partial penetration or fillet welds throat greater than 25.4 mm All other welds Secondary member welds	MT or PT MT or PT VT VT
Class 3 Supports	Primary member, groove or throat greater than 25.4 mm All other welds Secondary member welds	MT or PT VT VT
Special Requirements, All Classes	Welds transmitting loads in the through thickness direction in members greater than 25.4 mm	UT base metal beneath the weld
Core Support Structures (1) (2) (8)		
Core Support Structures (Provide direct support or restraint of the fuel, etc. under normal operating conditions.)	Category A, longitudinal butt welds Category B, circumferential butt welds Category C, flange to shell welds Category D, nozzle to shell welds Category E, beam end connections to other structures	Examination may be by any technique or certain combinations of techniques, from simple VT to MT or PT plus RT or UT. Quality factor *n* and fatigue factor *f* are dependent on the technique(s) selected, in accordance with Table NG-3352-1.
	Repair welds under 9.5 mm or 10% deep Repair welds over 9.5 mm or 10% deep	MT or PT MT or PT plus RT or UT
Internal Structures (Can be any other structure within the reactor vessel.) nonmandatory	Same as above	Same as above
Temporary Attachments (Removed before operation.)	All	MT or PT

NOTES:

(1) The required confirmation that facility welding activities are in compliance with the certified commitments will include the following third-party verifications:

(a) Facility welding and the applicable ASME Code requirements

(b) Facility welding activities are performed with the applicable ASME Code requirements;

(c) Welding activities related records are prepared, evaluated and maintained in accordance withers the ASME Code Requirements.

(d) Welding processes used to weld dissimilar base metal and welding filler metal combinations for the intended applications.

(e) The facility has established procedures for qualifications of welders and welding operators in accordance with the applicable ASME Code requirements.

(f) Approved procedures are available and used for preheating and post heating of welds, and those procedures meet the applicable requirements of the ASME Code requirements.

(g) Completed welds are examined in accordance with the applicable examination method required by the ASME Code.

(2) Radiographic film will be reviewed and accepted by the licensee's nondestructive examination (NDE), Level III examiner prior to final acceptance.

(3) The NDE requirements for Class 1 components will be as stated in subarticle NB-5300 of Section III of the ASME Code.

(4) The NDE requirements for Class 2 components will be as stated in subarticle NC-5300 of Section III of the ASME Code.

(5) The NDE requirements for Class 3 components will be as stated in subarticle ND-5300 of Section III of the ASME Code.

(6) Deleted

(7) The NDE requirements for component supports will be as stated in subarticle NF-5300 of Section III of the ASME Code.

(8) The NDE requirements for Core Support structures will be as stated in subarticle NG-5300 of Section III of the ASME Code.

(9) For corner joints UT may be used instead of RT. For Type 2 full penetration corner weld joints, if RT is used, the fusion zone, and parent metal beneath the attachment surface shall be UT examined after welding.

LEGEND:

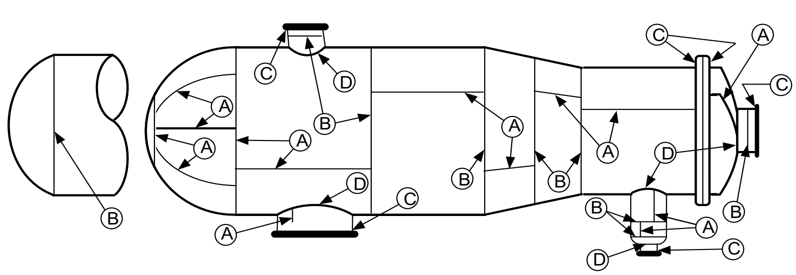
RT–Radiographic Examination

UT–Ultrasonic Examination

MT–Magnetic Particle Examination

PT–Liquid Penetrant Examination

VT–Visual Examination



Categories A, B, C, and D Welded Joint Typical Locations

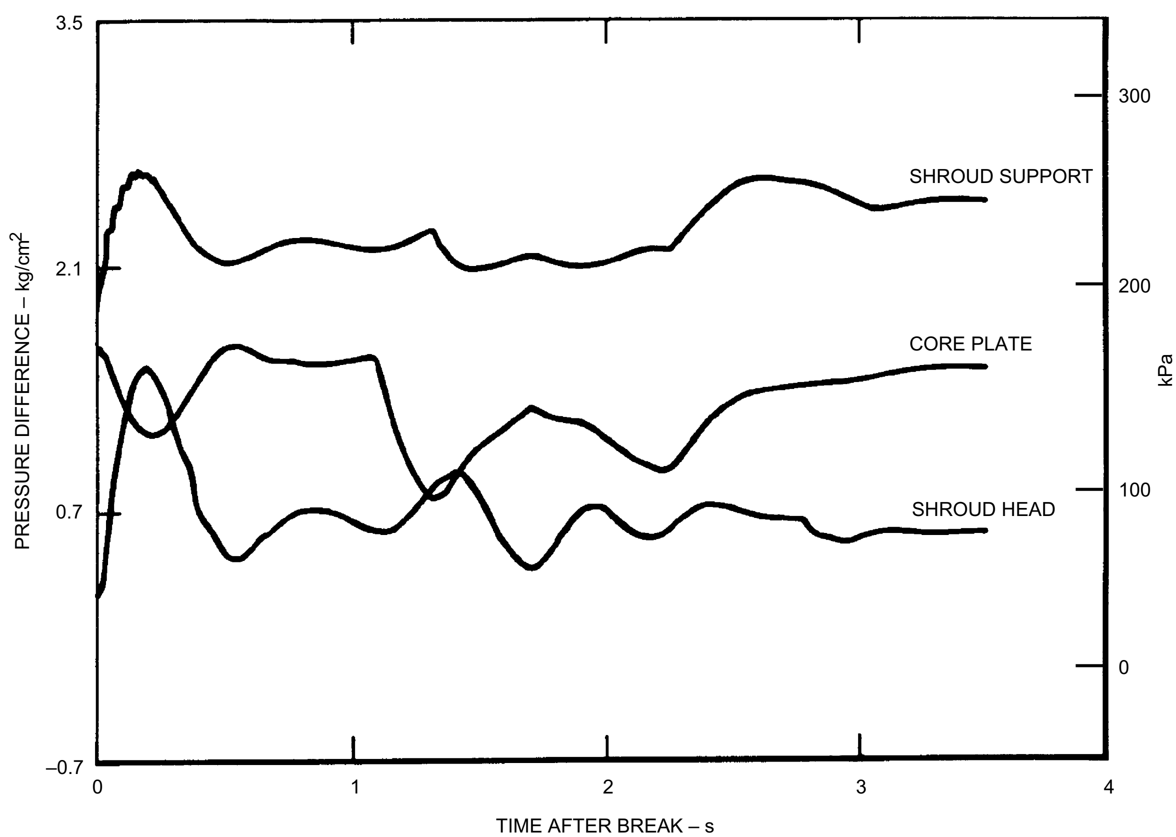


Figure 3.9-1 Transient Pressure Differentials Following a Steam Line Break

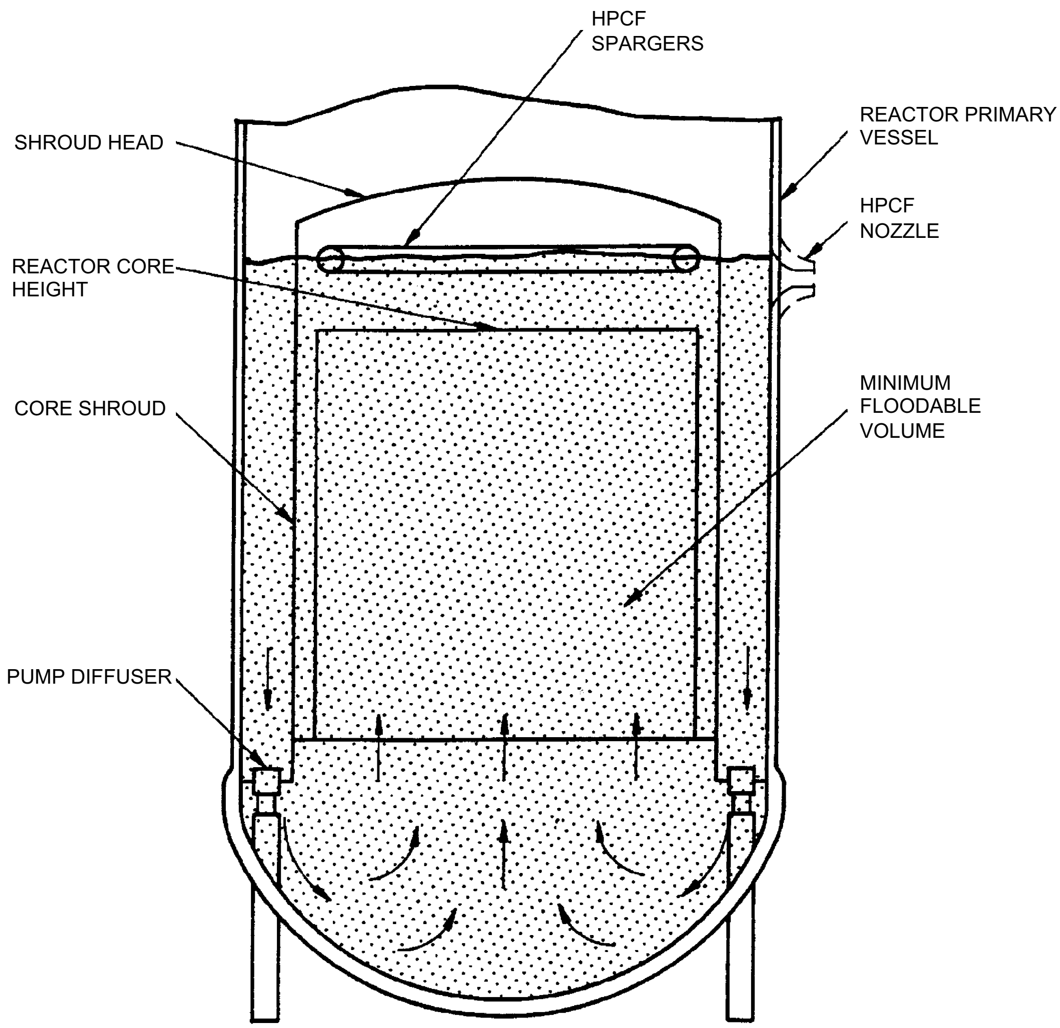


Figure 3.9-2 Reactor Internal Flow Paths and Minimum Floodable Volume

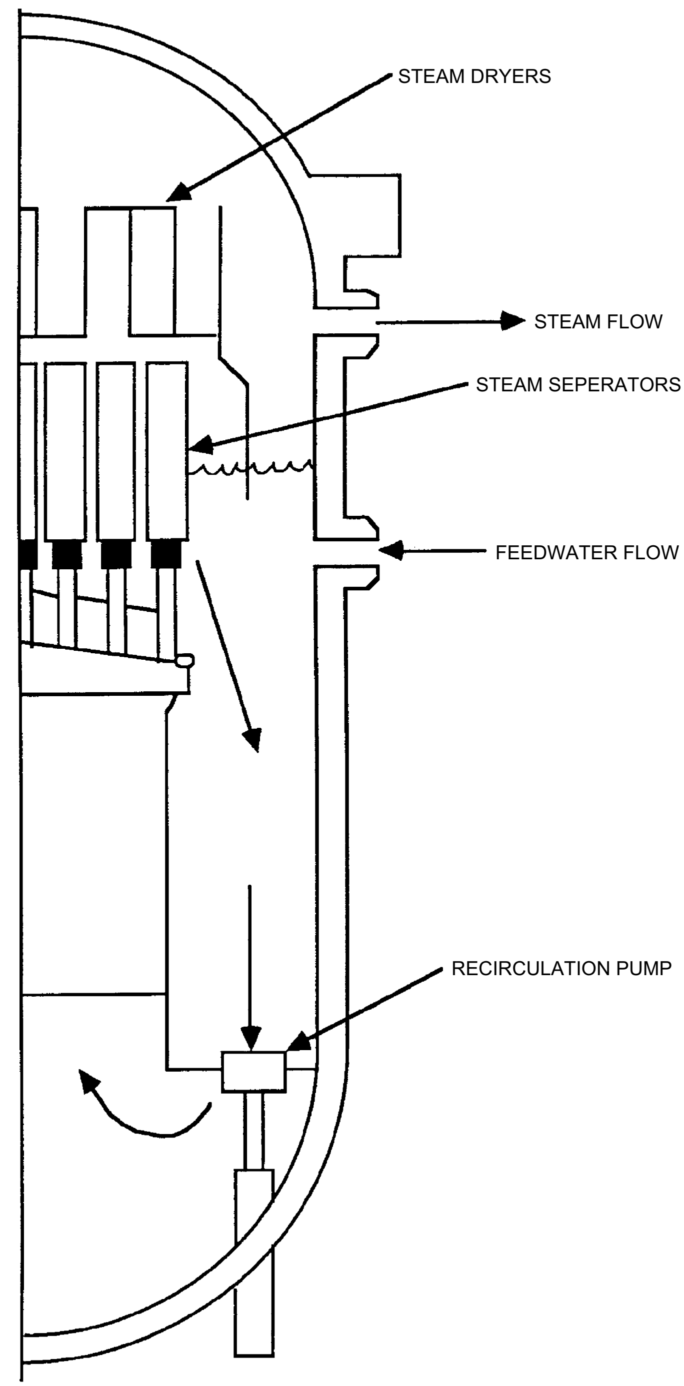


Figure 3.9-3 ABWR Recirculation Flow Path

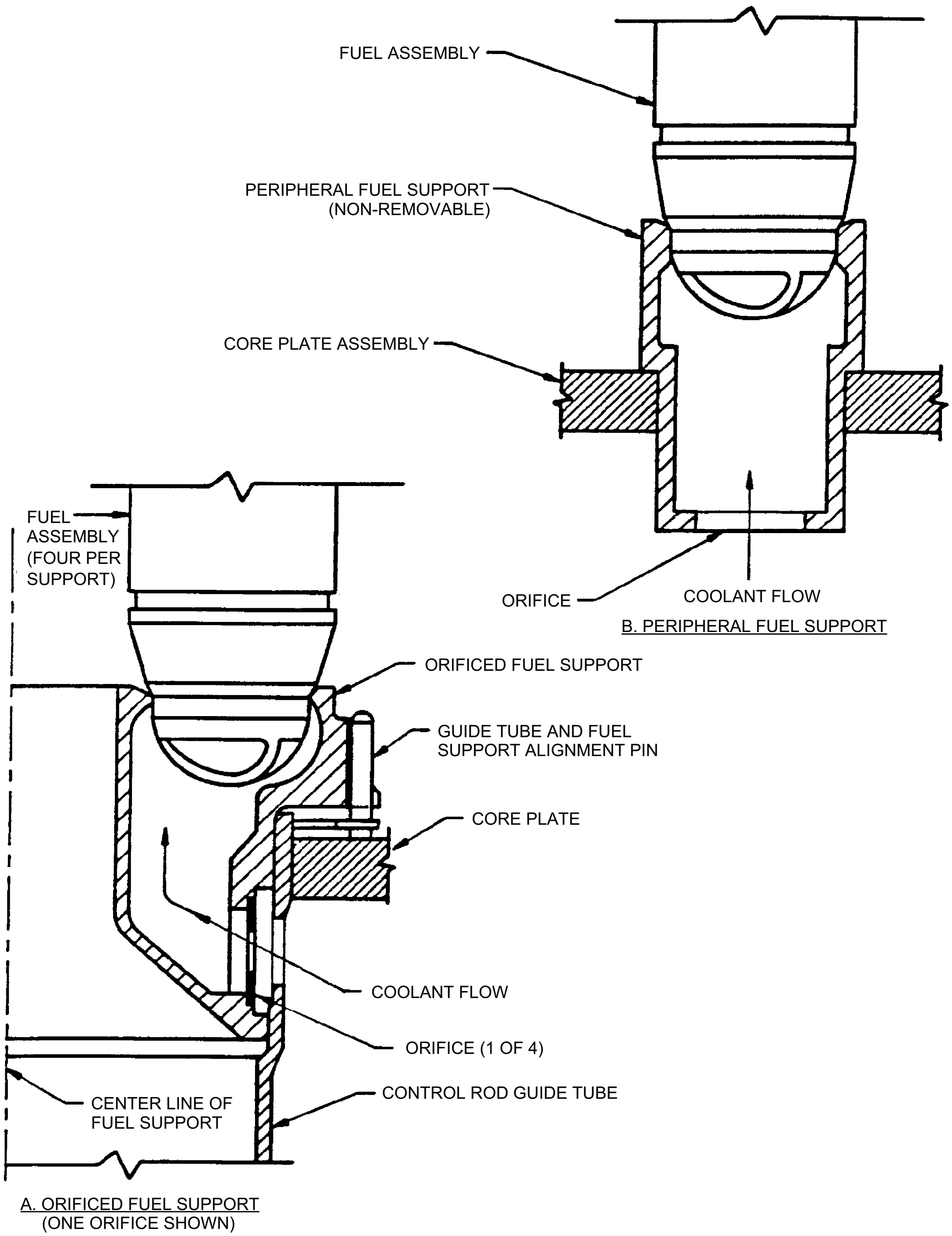


Figure 3.9-4 Fuel Support Pieces

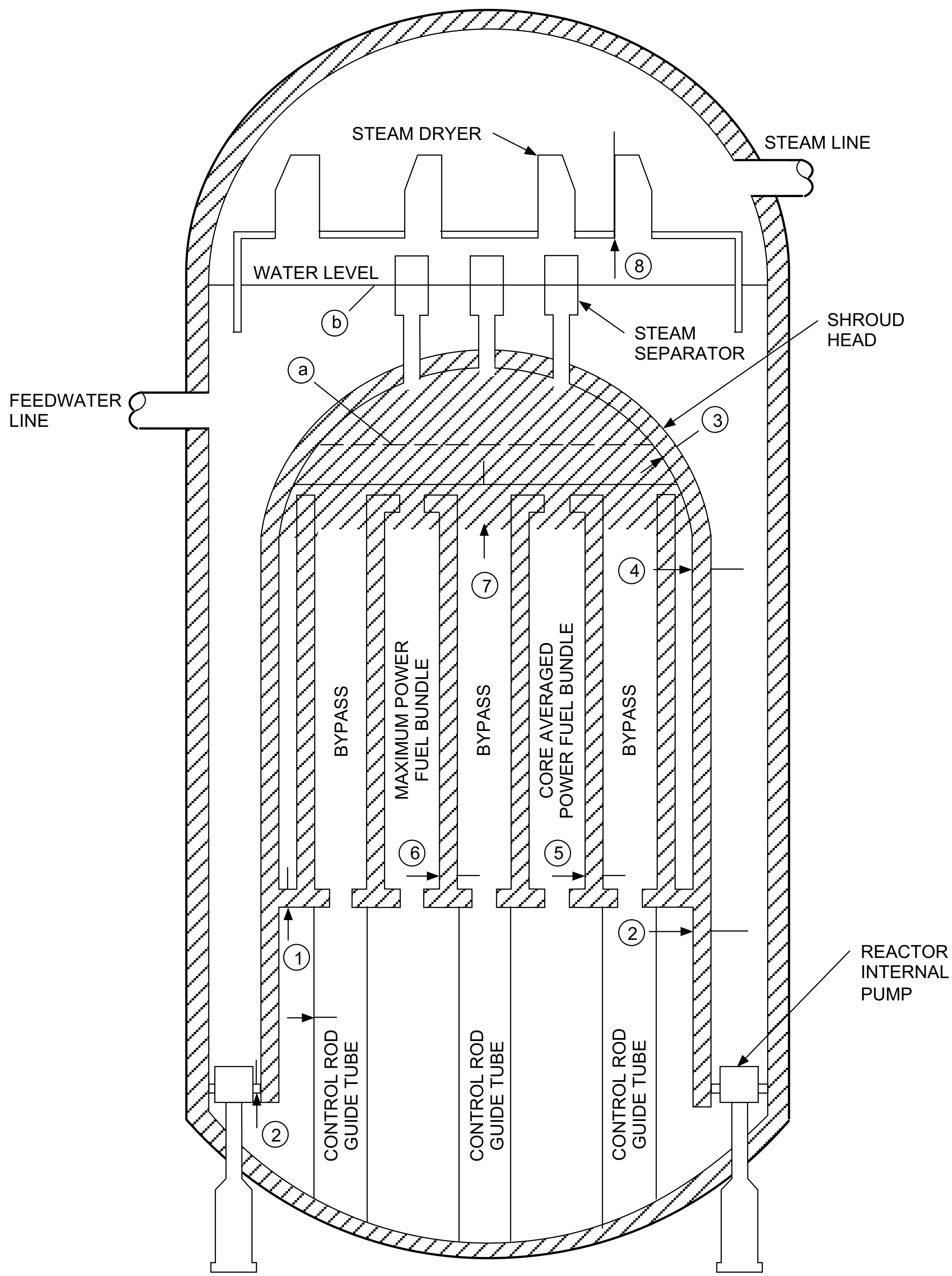


Figure 3.9-5 Pressure Nodes for Depressurization Analysis

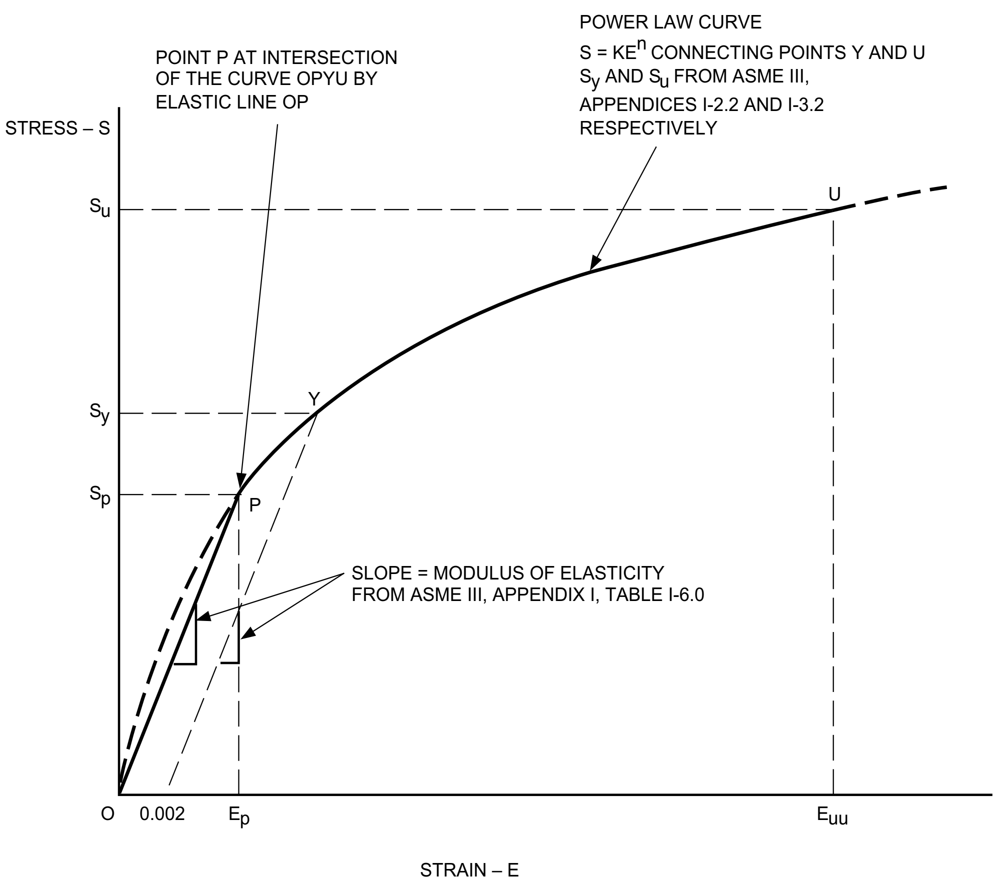


Figure 3.9-6 Stress-Strain Curve for Blowout Restraints