3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

The following information identifies the transients used in the design and fatigue analysis of ASME Code Class 1 components, reactor internals, and component supports. All transients are classified with respect to the component operating conditions identified as Level A (Normal), B (Upset), C (Emergency), and D (Faulted) and testing as defined in the ASME Section III. The transients specified below represent conservative estimates for design purposes only and are not intended to represent actual transients, nor necessarily reflect actual operating procedures; nevertheless, all envisaged actual transients are accounted for, and the number and severity of the design transients exceeds those that may be anticipated during the life of the plant.

Pressure, temperature, and flow rate resulting from the normal, test, upset, emergency, and faulted transients are computed by means of computer simulations of the NSSS (nuclear steam supply system) components. Design transients are detailed in the design specifications via time-history plots of the fluid temperature, pressure, and flow rate during plant events. The component designer then uses the transient data in the design specification as the basis for design and fatigue analysis. In support of the design of each Code Class 1 and core support (CS) component, a fatigue analysis for the combined effects of mechanical and thermal loads is performed in accordance with the requirements of ASME Section III. The purpose of the analysis is to demonstrate that fatigue failure will not occur when the components are subjected to typical dynamic events that may occur during the life of the plant.

ASME Section III, Division 1, Subsection NB, and NRC RG 1.207 are used for performing fatigue evaluations considering the effects of reactor coolant environment of the APR1400 components.

The fatigue analysis is based on a series of dynamic events depicted in the respective design specifications. Associated with each dynamic event is a mechanical, thermal-hydraulic transient presentation along with an assumed number of occurrences for the event.

The presentation is generally simple and straightforward because it is meant to envelop the actual plant responses. The intent is to present material for purposes of design.

The fundamental concept provides reasonable assurance that the consequences of the normal and upset conditions that are expected to occur in the power plant are enveloped by one or more of the dynamic event portrayals in the design specifications. The number of occurrences selected for each dynamic event is conservative so that in the aggregate, a 60-year useful life is provided by the design process.

Design loading combinations for ASME Code Class 1, 2, and 3 components are given in Subsection 3.9.3. Design loading combinations for reactor internals structures are presented in Subsection 3.9.5.2.

The principal design bases of the RCS and reactor internals structures are given in Section 5.2 and Subsection 3.9.5, respectively.

The APR1400 design basis initiating events and frequencies used in the stress analysis of ASME Code Class 1 and Class CS components of the primary system are shown in Table 3.9-1. The resulting APR1400 events and frequencies conservatively represent the 60-year design basis.

The design basis events (DBEs) are classified as normal, upset, emergency, faulted, and test. The normal and test events are planned operations that occur during the life of the plant. Upset events are occurrences that may occur during the life of the plant. Emergency and faulted events are not expected to occur but are included in the design basis for additional design margin. The normal and test events are selected by reviewing the expected plant operations. The upset, emergency, and faulted events are determined by reviewing industry databases (References 1, 2, 3, and 4) for events that have occurred, or that may be postulated to occur, based on observed plant behavior.

Normal and test event frequencies are determined by summing the number of expected plant operations over the 60-year design life. The frequencies for upset, emergency, and faulted events are determined on a probabilistic basis using industry databases (References 1, 2, 3, and 4). The 60-year design frequency of occurrence stated in Table 3.9-1 is always greater than the expected frequency of occurrence.

Conservative mathematical models and methodology are used to determine the thermal-hydraulic consequences of the DBEs on individual plant components. The design margin is further enhanced by enveloping similar events and using the most conservative thermal-hydraulic consequences to represent a composite group. The group frequency is then determined by algebraically summing the individual design frequencies.

Pressure and thermal stress variations associated with the design transients are considered in the design of supports, valves, and piping within the reactor coolant pressure boundary (RCPB).

In addition to the design transients listed above and included in the fatigue analysis, the loadings produced by seismic events are also applied in the design of components and support structures of the RCS. The number of cycles pertaining to fatigue effects of cyclic motion associated with the seismic events is provided in Subsection 3.7.3. Design loading combinations for ASME Code Class 1, 2, and 3 components are addressed in Subsection 3.9.3.

ASME Section III defines the plant conditions (Service Level A, B, C, and D, and Test Conditions) for the design of RCS Class 1 components, auxiliary Class 1 components, RCS component supports, and reactor internals as described below.

Normal Conditions (ASME Service Level A)

Normal conditions include any condition in the course of startup, operation in the design power range, hot standby, and system shutdown other than upset, emergency, faulted, or testing conditions.

Upset Conditions (Incidents of Moderate Frequency; ASME Service Level B)

Upset conditions include any deviations from normal conditions that are anticipated to occur often enough for the design to include a capability to withstand the conditions without operational impairment. Upset conditions include transients that result from any single operator error or control malfunction, transients caused by a fault in a system component requiring its isolation from the system, and transients due to loss of power. Upset conditions also include abnormal incidents not resulting in a forced outage as well as

those that cause forced outages for which the corrective action does not include any repair of mechanical damage. The estimated duration of an upset condition is included in the design specifications.

Emergency Conditions (Infrequent Incidents; ASME Service Level C)

Emergency conditions include deviations from normal conditions that require shutdown for correction of the conditions or repair of damage in the system. The emergency conditions have a low probability of occurrence, but are included to demonstrate that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system.

Faulted Conditions (ASME Service Level D)

Faulted conditions are combinations of conditions associated with low probability, postulated events whose consequences may impair the integrity and operability of the nuclear energy system to the extent that consideration of public health and safety are involved. Such considerations require compliance with safety criteria. The methods of analysis to calculate the stresses and deformations conform to the methods in ASME Section III, Division 1, Appendix F.

Testing Conditions

Testing conditions include hydrostatic pressure tests of individual components and the primary system as specified in this section.

In accordance with ASME Section III, emergency and faulted conditions are not included in fatigue evaluations, with the exception that any significant emergency cycles in excess of 25 are considered in the fatigue analyses.

3.9.1.1.1 Service Level A Conditions

Service Level A conditions consist of the following events.

a. Steady-state operation with normal NSSS parameter variations in the increasing and decreasing directions

The plant could experience primary and secondary process parameter variations because of secondary steam conditions. Power step changes of 10 percent are used to envelop this event. This is conservative since the 10 percent power step change produces more severe plant process parameter variations than those that occurred during any normal plant variations as the result of changing steam conditions.

Each event is assumed to occur 1,500,000 times during the 60-year plant design life.

b. Daily load follow operation

For daily load follow operation, the power is maintained at 100 percent for 10 through 16 hours, ramped down from 100 percent to 50 percent over a 2-hour period, operated for 4 through 10 hours at 50 percent, and then ramped up from 50 percent to 100 percent power over a 2-hour period.

Each event is assumed to occur 22,000 times during the 60-year plant design life.

c. Turbine power step changes of 10 percent power (15 to 100 percent power)

This event is a turbine power change from 100 to 90 percent power and from 90 to 100 percent power. The transients of step change from 25 to 15 percent power and step change from 15 to 25 percent power are also considered. These power step changes are representative of other power levels and serve to envelop smaller power step changes.

Each event is assumed to occur 3,200 times during the 60-year plant design life.

d. Turbine power step changes of 1 percent power (5-15 percent)

This power step change is from 15 to 14 percent power and from 14 to 15 percent power. The transients of step change from 6 to 5 percent power and step change from 5 to 6 percent power are also considered. These power steps are representative of other low power levels and serve to envelop other possible power steps.

Each event is assumed to occur 1,600 times during the 60-year plant design life.

e. Turbine load rejection up to 50 percent power (50-100 percent power)

This event is a load rejection up to 50 percent power. The load rejection from 100 percent to 50 percent power is more severe than any other smaller load rejection.

This event is assumed to occur 60 times during the 60-year plant design life.

f. Turbine generator runback to house load

This event is a loss of offsite load with the turbine running back to house load. The house load is about 5 percent of full power conditions.

This event is assumed to occur 60 times during the 60-year plant design life.

g. Reactor trip

This event is an uncomplicated reactor trip. The uncomplicated reactor trip is an event when the reactor trip is the event initiator. A reactor trip can occur at any power level and causes a turbine trip.

This event is assumed to occur 150 times during the 60-year plant design life.

h. Turbine trip

This event is a turbine trip caused by a mechanical or electrical problem.

This event is assumed to occur 150 times during the 60-year plant design life.

i. Turbine power ramp changes of 5 percent/min (15-100 percent power)

This event is a turbine power ramp change from 100 to 15 percent power and power ramp change from 15 to 100 percent power. The turbine power ramp changes of 5 percent/min between these power levels are more severe than power ramps from any other power levels and serve to envelop the less severe power ramps.

Each event is assumed to occur 3,200 times during the 60-year plant design life.

j. Turbine power ramp changes of 1 percent /min (5-15 percent power)

This event is a turbine power ramp change from 15 percent to 5 percent power and power ramp change from 5 percent to 15 percent power. The turbine power ramp changes of 1 percent/min between these power levels are more severe than power ramps from any other power levels and serve to envelop the less severe power ramps.

Each event is assumed to occur 1,600 times during the 60-year plant design life.

k. Loss of main FW pumps without reactor trip

The loss of main feedwater pumps without generating a reactor trip event is composed of two events: loss of a main feedwater pump and loss of two main feedwater pumps. The loss of one main feedwater pump results in a minor system transient. The two remaining feedwater pumps automatically increase their speeds to match the flow rate requirements of the original power level. The loss of two main feedwater pumps at 100 percent power envelops all other possible cases that may occur during part load operation.

This event is assumed to occur 60 times during the 60-year plant design life.

l. NSSS operations with manual control of the CEAs, turbine bypass valves, pressurizer spray/heaters, pressurizer level control, and feedwater flow (0 to 5 percent power)

   This event consists of several manual operations that can be expected to occur during low power conditions. These include the manual operation of the CEAs, turbine bypass valves, pressurizer spray/heaters, pressurizer level control, and the feedwater flow control.

   This event is assumed to occur 1,600 times during the 60-year plant design life.

m. Opening or closure of the economizer feedwater control valve

   The steam generator has two feedwater control valves to provide the necessary flow control over the full power range. The smaller downcomer valve controls flow between 0 percent and 20 percent power and the larger economizer feedwater valve controls flow between 20 percent and 100 percent power. The feedwater valve switch is performed at 20 percent reactor power during power increase and at 18 percent reactor power during power decrease.

   This event is assumed to occur 500 times during the 60-year plant design life.

n. NSSS operations with the NSSS control systems in the manual mode (5-100 percent power)

   This event consists of several manual operations that can be expected to occur during the 5 to 100 percent power range. It includes manual operation of the CEAs, turbine bypass valves, pressurizer spray/heaters, pressurizer level control, and the feedwater flow control. The control systems can be manually controlled within the normal automatic control bands.

   This event is assumed to occur 3,200 times during the 60-year plant design life.

o. Manual operation of the auxiliary spray system

The manual operation of the auxiliary spray system may be required during power operation to reduce primary pressure excursions when the main spray system is out-of-service. This transient may be especially severe for the pressurizer spray line since isolated fluid in the auxiliary spray line may cool down to low temperatures before being sprayed into the pressurizer.

This event is assumed to occur 250 times during the 60-year plant design life.

p. High capacity steam generator blowdown

The steam generator high capacity blowdown is performed with a flow rate of approximately 5 percent of the steam generator maximum steaming rate to maintain steam generator chemistry within control limit.

This event is assumed to occur 3,200 times during the 60-year plant design life.

q. Shift from normal to maximum CVCS flow rate

The chemical and volume control system (CVCS) letdown and charging flow rates may be increased to support daily load follow operations or to more rapidly reduce impurities in the RCS.

This event is assumed to occur 3,200 times during the 60-year plant design life.

r. Low-low VCT level and charging pump diversion to the boric acid storage tank

Low volume control tank (VCT) level results in diverting the charging flow sources from VCT to the boric acid storage tank.

This event is assumed to occur 60 times during the 60-year plant design life.

s. Spurious actuation of the pressurizer spray

The spurious actuation of the pressurizer spray may occur because of a mechanical or a control system failure. The failure is assumed to open both spray valves while the plant is operating at full power. This is the more severe event of the possible spray valve failures and envelops other possible failures.

This event is assumed to occur 60 times during the 60-year plant design life.

t. Spurious actuation of the pressurizer heaters

The spurious actuation of the pressurizer heaters may occur because of a mechanical or a control system failure. The failure is assumed to actuate all pressurizer heaters while the plant is operating at full power and at hot standby.

This event is assumed to occur 60 times during the 60-year plant design life.

u. Inadvertent closure of one economizer or downcomer FW control valve

The spurious closure of one economizer or downcomer feedwater control valve may occur because of a mechanical or a control system failure. The failure will result in the loss of feedwater flow to the economizer section of one steam generator.

This event is assumed to occur 60 times during the 60-year plant design life.

v. Inadvertent opening of one economizer or downcomer FW control valve

The spurious opening of one economizer or downcomer feedwater control valve may occur because of a mechanical or a control system failure. The failure will result in increase of feedwater flow to the economizer section of one steam generator.

This event is assumed to occur 60 times during the 60-year plant design life.

w. Inadvertent isolation of one main FW heater

The inadvertent isolation of one main feedwater heater may result from an operator error or mechanical failure. This event results in a decrease in feedwater temperature to the steam generator.

This event is assumed to occur 60 times during the 60-year plant design life.

x. Startup and coastdown of a reactor coolant pump at hot standby (HSB)

The startup and coastdown of a reactor coolant pump at HSB occur during each plant heatup and cooldown operation.

This event is assumed to occur 2,000 times during the 60-year plant design life.

y. Startup and shutdown of the shutdown cooling system at hot shutdown (HSD)

The startup and shutdown of shutdown cooling system (SCS) at HSD condition occur during each plant cooldown and heatup operations.

This event is assumed to occur 250 times during the 60-year plant design life.

z. Spurious startup of a safety injection pump during shutdown condition

Spurious startup of a safety injection pump during shutdown condition is considered to occur due to operator error or control system failure.

This event is assumed to occur 60 times during the 60-year plant design life.

aa. Spurious actuation of the pressurizer heaters at HSB

The spurious actuation of the pressurizer heaters may occur because of a mechanical or a control system failure. The failure is assumed to actuate all pressurizer heaters while the plant is operating at hot standby condition.

This event is assumed to occur 60 times during the 60-year plant design life.

ab. Plant heatup and cooldown

Plant heatup is defined as operations that bring the RCS from a condition where the reactor is subcritical and the RCS is at nearly ambient temperature and atmospheric pressure to a condition where the system temperature and pressure are at their normal operating zero power values. The temperature changes of 37.8 °C (100 °F) per hour that bound the heatup rate are conducted by four reactor coolant pumps (RCPs). The heatup rate is controlled by the shutdown cooling system and the steam generators. The heatup rate for the pressurizer is 93.3 °C (200 °F) per hour.

Plant cooldown is a series of operations that bring the RCS from a power operation condition to a cold shutdown condition in preparation of refueling or other maintenance operations. The cooldown operations represented by ramp changes in temperature of 37.8 °C (100 °F) per hour that bound the cooldown rate are performed by the steam generator steam dump and shutdown cooling operation. The cooldown rate for the pressurizer is allowed up 93.3 °C (200 °F) per hour.

Each event is assumed to occur 250 times each during the 60-year plant design life.

3.9.1.1.2 Service Level B Conditions

Service Level B Conditions consist of the following events.

a. Decrease in FW temperature

A decrease in main feedwater temperature results in an increase in heat removal by the secondary system. A postulated failure in the feedwater train is assumed to result in a decrease in feedwater enthalpy.

This event is assumed to occur 20 times during the 60-year plant design life.

b. Increase in FW flow rate

An increase in feedwater flow results in an increase in heat removal from the secondary system. A failure in the main feed train is assumed to cause an increase in feedwater flow up to a maximum flow rate of about 160 percent of rated flow.

This event is assumed to occur 20 times during the 60-year plant design life.

c. Increase in steam flow rate

An increase in main steam flow results in an increase in heat removal from the secondary system. A failure in the secondary system is assumed to cause an increase in steam flow up to about 11 percent of the full-power steaming rate.

This event is assumed to occur 20 times during the 60-year plant design life.

d. Inadvertent opening of a main steam safety valve

An inadvertent opening of a main steam safety valve results in an increase in heat removal from the secondary system. The event is postulated to result in the increase in main steam flow while operating at full power.

This event is assumed to occur 10 times during the 60-year plant design life.

e. Loss of external load

A loss of external load (event) occurs due to the separation of the turbine/generator from the electricity distribution grid. When house load operation is operable, the plant is controlled by reactor power cutback system (RPCS) and steam bypass control system without a reactor trip for loss of external load. When the house load operation is not operable, the turbine is tripped for loss of external load, the turbine stop valve is closed, and the steam flow from the steam generator to the turbine is blocked. This event shows the loss of external load for an inoperable case of house load operation.

This event is assumed to occur 20 times during the 60-year plant design life.

f. Loss of condenser vacuum

A loss of condenser vacuum event results in a decrease in heat removal from the secondary system. A loss of condenser vacuum can be caused by the failure of the circulating water system providing condenser cooling, failure of the main condenser evacuator system to remove non-condensable gases, or excessive in-leakage of air through a turbine gland. The turbine is assumed to trip immediately upon the loss of condenser vacuum.

This event is assumed to occur 20 times during the 60-year plant design life.

g. Loss of non-emergency AC power to the station auxiliaries

A loss of non-emergency AC power to the station auxiliaries event results in a decrease in heat removal by the secondary system. This event may result from a complete loss of the external grid or a loss of the onsite AC power distribution system. Emergency power is still supplied to the plant by emergency diesel generators. The loss of non-emergency AC power to the station results in the loss of the busses that power the reactor coolant pumps, thereby causing all the pumps to coast down.

This event is assumed to occur 20 times during the 60-year plant design life.

h. Main steam isolation valve closure

A main steam isolation valve (MSIV) closure event results in a decrease in heat removal by the secondary system. An MSIV closure event is initiated by the closure of all the MSIVs that is the result of a spurious closure signal (e.g., spurious main steam isolation signal). This results in termination of both main steam and main feedwater flow because the main steam isolation signal closes the main feedwater isolation valves.

This event is assumed to occur 20 times during the 60-year plant design life.

i. Loss of normal feedwater flow

A loss of main feedwater flow event results in a decrease in heat removal from the secondary system. The loss of main feedwater flow is caused by losing one or both main feedwater pumps or by spurious signals in the feedwater control system. The limiting loss of main feedwater event is defined as the total loss of main feedwater to both steam generators. The loss of main feedwater flow results in a decreasing steam generator level and an increasing secondary pressure. These secondary variations cause a corresponding increase in primary system temperatures and pressures.

This event is assumed to occur 20 times during the 60-year plant design life.

j. Loss of forced reactor coolant flow

A loss of forced reactor coolant flow event results in a decrease in the RCS flow. The loss of forced reactor coolant could be initiated by a failure in the RCP auxiliary system or loss of non-emergency AC power.

This event is assumed to occur 20 times during the 60-year plant design life.

k. Natural circulation cooldown (HSB to HSD)

If the AC power to all non-safety systems including the RCPs is lost during power operation, the plant is tripped and cooled down in a natural circulation mode. The amount of time needed to achieve plant shutdown conditions will be greater. Only safety-related systems are used to cool down the plant. A natural circulation cooldown operation is necessary to cool down the RCS from the HSB to HSD condition at which the operation of shutdown cooling system is allowed.

This event is assumed to occur 10 times during the 60-year plant design life.

l. Uncontrolled CEA withdrawal at low power

An uncontrolled control element assembly (CEA) withdrawal at low power or subcritical condition results in a reactivity or power distribution anomaly. The CEA withdrawal at low power or subcritical conditions adds positive reactivity to the reactor core by removing regulating CEAs. This causes an increase in core power and heat flux resulting in increasing core temperatures and pressures.

This event is assumed to occur five times during the 60-year plant design life.

m. Uncontrolled CEA withdrawal at high power

An uncontrolled CEA withdrawal at high power event causes a reactivity or power distribution anomaly.

This event is assumed to occur five times during the 60-year plant design life.

n. Control rod misoperation, RPCS inadvertent operation, or operator error

A single, full-length CEA drop event causes a reactivity or power distribution anomaly. The CEA drop is caused by failure in the CEA drive mechanism causing an initial insertion of negative reactivity.

This event is assumed to occur 50 times during the 60-year plant design life.

o. Loss of component cooling water to the letdown heat exchanger

The loss of component cooling water to the letdown heat exchanger event results in increase in RCS inventory. The letdown isolation valve closes on high letdown flow temperature to prevent CVCS equipment from being exposed to high temperature conditions.

This event is assumed to occur 10 times during the 60-year plant design life.

p. CVCS malfunction that increases RCS inventory

A CVCS malfunction event results in an increase in RCS inventory. The limiting scenario with respect to increases in RCS inventory is the pressurizer level control system malfunction. During this event, the letdown line is isolated and the charging control valve is fully opened.

This event is assumed to occur 10 times during the 60-year plant design life.

q. Inadvertent opening of the pilot-operated safety relief valve (POSRV closes as expected)

The inadvertent opening of the POSRV is an incident that results in a decrease in RCS inventory. Only one POSRV is assumed to open. The RCS pressure stops to decrease as the POSRV recloses.

This event is assumed to occur 10 times during the 60-year plant design life.

r. Failure of small lines carrying coolant outside containment (letdown line break)

The double-ended break of a letdown line outside containment event results in a decrease in RCS inventory. Because of the pipe size, the consequences of a double-ended letdown line break outside containment bound all possible instrument or sample line breaks.

This event is assumed to occur 20 times during the 60-year plant design life.

s. Reactor coolant pump seal failure

An RCP seal failure event is set as a DBE for the design of the RCP bleed-off line.

This event is assumed to occur 10 times during the 60-year plant design life.

t. Loss of seal injection with loss of cooling water

A loss of seal injection with a loss of cooling water event is set as a DBE for the design of the RCP bleed-off line.

This event is assumed to occur five times during the 60-year plant design life.

3.9.1.1.3 Service Level C Conditions

There are no events classified as a Service Level C Condition.

3.9.1.1.4 Service Level D Conditions

Service Level D Conditions consist of the following events:

a. Steam system piping failure

A main steam line break (MSLB) results in an increase in heat removal by the secondary system. A rupture in the main steam line is postulated to cause an uncontrolled blowdown of the steam generators until the main steam isolation valves (MSIVs) close upon the receipt of a main steam isolation signal (MSIS). If the steam line break occurs downstream of the MSIV, the closure of the MSIVs will terminate the primary system cooldown. If the steam line break occurs upstream of the MSIVs, the ruptured steam generator continues to blow down after MSIS, causing a greater cooldown of the primary system.

This event is assumed to occur one time during the 60-year plant design life.

b. Feedwater system line break (FWLB)

The feedwater system pipe break or feedwater line break (FWLB) is an accident that results in a decrease in heat removal from the secondary system. A break in the feedwater system piping is postulated to occur and cause a dependent loss of the main feedwater pumps. If the break occurs upstream of the reverse flow check valves, the thermal-hydraulic response will be similar to that of a loss of

normal feedwater flow event. If the break occurs between the last reverse flow check valve and the steam generator, both steam generators blow down until the main steam isolation valves close following receipt of a main steam isolation signal.

This event is assumed to occur one time during the 60-year plant design life.

c. Reactor coolant pump rotor seizure

A single RCP rotor seizure with the loss of offsite power is an accident that results in a decrease in reactor coolant system flow rate. The RCP rotor seizure is caused by the seizure of either the upper or the lower RCP thrust-journal bearings.

This event is assumed to occur one time during the 60-year plant design life.

d. Reactor coolant pump shaft break

An RCP shaft break with the loss of offsite power is an accident that results in a decrease in reactor coolant system flow rate. A single reactor coolant pump shaft break is postulated to cause a low reactor coolant system flow trip by reactor protection system.

This event is assumed to occur one time during the 60-year plant design life.

e. Rod ejection accident

A CEA ejection is an accident that causes a reactivity or power distribution anomaly. The CEA ejection is postulated to be caused by a circumferential rupture of the CEA drive mechanism housing or nozzle. The rupture is assumed to allow the instantaneous ejection of the rod with the largest reactivity bite. The subsequent rapid reactivity excursion will cause variations in the NSSS process parameters.

This event is assumed to occur one time during the 60-year plant design life.

f. Inadvertent opening of pilot-operated safety relief valve (POSRV fails to close)

The inadvertent opening of POSRV is an accident that results in a decrease in RCS inventory. The RCS pressure will decrease continuously as the POSRV fails to close. Hence, a reactor trip occurs due to the low pressurizer pressure. This event is a small break loss of coolant accident.

This event is assumed to occur one time during the 60-year plant design life.

g. Steam generator tube rupture (SGTR)

The steam generator tube rupture (SGTR) is an accident that causes a decrease in RCS inventory. It is postulated that a double-ended rupture of a single U-tube occurs penetrating the barrier between the reactor coolant system and the main steam system.

This event is assumed to occur one time during the 60-year plant design life.

h. Loss-of-coolant accident (LOCA) resulting from postulated pipe breaks within the RCS pressure boundary

The LOCA is an accident that results in a decrease in RCS inventory.

This event is assumed to occur one time during the 60-year plant design life.

i. Total loss of feedwater flow (TLOFW)

The total loss of feedwater flow event is a beyond design bases event. This event is initiated by a loss of main and auxiliary feedwater flow and results in decrease in RCS heat removal.

This event is assumed to occur one time during the 60-year plant design life.

3.9.1.1.5 Testing Conditions

Testing conditions consist of the following events.

a. RCS hydrostatic test

The hydrostatic test is performed to provide reasonable assurance of the integrity of the RCS pressure boundary, its components, and associated unisolable piping systems at 125 percent of design pressure.

This event is assumed to occur 15 times during the 60-year plant design life.

b. Secondary hydrostatic test

The secondary hydrostatic test is performed to provide reasonable assurance of the integrity of the secondary side of steam generator, including the unisolable portion of the main steam, main feedwater, blowdown, recirculation, and auxiliary feedwater lines, at 125 percent of design pressure.

This event is assumed to occur 15 times during the 60-year plant design life.

c. RCS leak test

Whenever the RCS has been opened, an RCS leak test is conducted at the normal operating pressure. The RCS loop pressure is raised following the limitations curve of the plant hydrostatic test by control of charging and letdown flow.

This event is assumed to occur 200 times during the 60-year plant design life.

d. Secondary leak test

Whenever the secondary system has been opened, a secondary side leak test is conducted at the design pressure.

This event is assumed to occur 200 times during the 60-year plant design life.

e. SIS/SCS preoperational and maintenance test

SIS/SCS preoperational and maintenance test is conducted to provide reasonable assurance of the operability of SIS/SCS components. The preoperational test is a pre-core testing performed to provide reasonable assurance that the components and/or systems perform their intended functions. The maintenance test provides reasonable assurance that the components perform their intended functions after maintenance.

This event is assumed to occur 360 times during the 60-year plant design life.

f. SIS/SCS check valve operability tests

SIS/SCS check valve operability test is conducted to provide reasonable assurance of the operability of SIS/SCS check valves. The test is performed by safety injection pumps (SIPs) in refueling mode or shutdown cooling pumps (SCPs) in cold shutdown mode.

This event is assumed to occur 120 times during the 60-year plant design life.

3.9.1.2 Computer Programs Used in Stress Analyses

3.9.1.2.1 Code Class Systems, Components, and Supports

The following paragraphs provide a summary of the applicable computer programs used in the stress and structural analyses for ASME Code Class systems, components, and supports in the APR1400 design. The summaries include individual descriptions and applicability data. The computer codes used in these analyses have been verified in conformance with design control methods, consistent with the quality assurance program described in Chapter 17.

3.9.1.2.1.1 ABAQUS

The ABAQUS program is a general-purpose nonlinear finite element program with structural and heat transfer capabilities. ABAQUS is used for stress analysis of regions of

vessels, piping, or supports that may deform plastically under prescribed loadings. It is also used for elastic analyses of complex geometries where the graphics capability enables a well-defined solution. The thermal capabilities of ABAQUS are used for complex geometries where simplification of input and graphical output are preferred.

ABAQUS is commercially available and has had sufficient use to justify its applicability and validity. See Reference 5 for information on ABAQUS.

3.9.1.2.1.2 PICEP

The PICEP program calculates the flow through a crack in a pipe. PICEP uses the simplified engineering approach for elastic-plastic fracture analysis for finding the crack opening displacement and area. Fluid calculation options include single and two-phase flow as well as allowance for friction. PICEP, commercial software, was developed by the Electric Power Research Institute (EPRI). See Reference 22 in Subsection 3.6.5 for information on PICEP.

3.9.1.2.1.3 ADLPIPE

ADLPIPE is a linear finite element program for the static and dynamic analysis of piping systems. These systems may include such components as bends, elbows, tees, reducers, socket or butt welds, flexible couplings, and flanges, with the appropriate flexibility factors and stress indices accounted for. Support types may include rigid, spring, constant-force, snubber, anchor, or user-specified types, and may have any desired orientation.

Analyses performed include thermal, dead weight, applied load, frequency and mode shape, and response spectrum. Following the static and dynamic analysis phase, the program performs the ASME Section III Class 1 analysis in any manner specified by the user to create the appropriate loading cases applicable for each of the ASME Code stress equations. See Reference 6 for information on ADLPIPE.

3.9.1.2.1.4 CLEVER

CLEVER determines SG and snubber stroke, and building interface boundaries for the SG snubber lever system. The program verifies the kinematics of the snubber lever linkage

systems based on input motions of the SG lug and detailed snubber lever system geometry. See Reference 7 for information on CLEVER.

3.9.1.2.1.5 HeadPR (Head Penetration Reinforcement Program)

The HeadPR computer program calculates the available reinforcement and the reinforcement that is needed for penetrations in the hemispherical heads. The technique described in ASME Section is used.

The HeadPR computer program is used to perform the preliminary sizing and reinforcement calculations for hemispherical heads in the reactor vessel.

The program was verified by comparisons of program results and hand-calculated solutions of classical problem. See Subsection 3.9.10, Reference 8, for more information.

3.9.1.2.1.6 CEFLASH-4B

The CEFLASH-4B computer program calculates transient conditions resulting from a flow line rupture in a water/steam flow system. The program is used to calculate steam generator internal loadings following a postulated main steam line and main feedwater line break.

The program was verified by comparison of program results and the result of the CEFLASH-4A computer program. See Subsection 3.9.10, Reference 9, for more information.

3.9.1.2.1.7 ANSYS

The ANSYS computer program is a large-scale, general purpose, finite-element analysis program for linear and nonlinear structural and thermal analyses, and additional descriptive information on this code is provided in Subsection 3.9.1.2.2.1.

The program is used to numerous applications for all components in the areas of structural, fatigue, thermal, and eigenvalue analysis. Analysis capabilities include static and dynamic;

elastic, plastic, creep and swelling; small and large deflections; steady-state and transient heat transfer; and fluid flow.

The program has been verified by comparison with known theoretical solutions, experimental results, and by other calculated solutions. See Subsection 3.9.10, Reference 10, for more information.

3.9.1.2.1.8 AFP2D

The AFP2D computer program uses the thermal stresses of two-dimensional axisymmetric structure resulting from the ANSYS program run. The program combines thermal stresses calculated for transient load steps with stresses due to pressure and external mechanical loads, and calculates primary plus secondary stresses, peak stresses and their ranges of stress intensities and fatigue usage factors.

The program was verified by comparisons of program results and hand-calculated solutions of classical problems. See Subsection 3.9.10, Reference 11, for more information.

3.9.1.2.1.9 TSPOST

The TSPOST computer program uses the stresses in the tubesheet of the SG resulting from two-dimensional axisymmetric model using ANSYS program run. The program evaluates the primary stress by various pressure conditions imposed on the primary and the secondary side of SG, and calculates range of primary plus secondary stress intensity and cumulative usage factors, by combining the thermal and pressure stresses.

The program was verified by comparisons of program results and hand-calculated solutions of classical problems. See Subsection 3.9.10, Reference 12, for more information.

3.9.1.2.1.10 AFPOST

The AFPOST computer program uses the thermal stresses of two- and three-dimensional structural resulting from ANSYS program run. The program combines thermal stress calculated for transient load steps with stresses due to pressure and external mechanical

loads, and calculates primary plus secondary stresses, peak stresses and their ranges of stress intensities and fatigue usage factors.

The program was verified by comparisons of program results and hand-calculated solutions of classical problems. See Subsection 3.9.10, Reference 13, for more information.

3.9.1.2.1.11 ATHOS3

The ATHOS3 is a three-dimensional, two-phase flow distribution computer program with both steady-state and transient capability. Homogeneous and axial flow algebraic slip models are available. A typical geometric model includes approximately 1000 cells but more detailed models are possible. Secondary fluid results include the velocity vector components in the three coordinate directions, pressure, temperature/quality, and density in each cell. The corresponding heat flux, tube wall temperature, and primary temperatures and the circulation ratio are also output. Vector and scalar plotting is available. In transient calculations, water level and steam flow are calculated as a function of time.

The program was verified by comparing the results to measured data from small-scale experiments, model SGs, and full-scale SGs (see Subsection 3.9.10, Reference 14 for more information).

3.9.1.2.1.12 AFPOST+e

The AFPOST+e is a fatigue evaluation program that uses environmental fatigue factors (Fen factor) for pressure vessel. This program combines thermal stresses from ANSYS thermal stresses result with other loads depending on several options to give total stresses and stress intensities range. Then, transformed strain rate is calculated in accordance with NRC RG 1.207 (NUREG/CR-6909) using combined total stresses. The program converts sulfur value, oxygen value, and temperature listed in NRC RG 1.207 (NUREG/CR-6909) into transformed sulfur value, transformed oxygen value, and transformed temperature to calculate Fen factor and performs fatigue evaluation with environmental effect considered.

The program was verified by comparison of the results from the program run and hand-calculation for a test problem. See Subsection 3.9.10, Reference 15, for more information.

3.9.1.2.1.13 PTXIG

This program applies the procedures of Appendix G of ASME Section XI and the supplemental procedures in Welding Research Council Bulletin 175 to evaluate pressure vessels against failure. The program calculates the allowable internal pressure as a function of crack size, RT_{NDT}, and thermal conditions that are input by the user. The program was verified by comparison of program results and hand calculations (see Reference 16 for more information).

3.9.1.2.1.14 PIPESTRESS

PIPESTRESS (Reference 17) is a piping analysis program that is applied to the static and dynamic analyses including response spectra and time history analyses.

PIPESTRESS is used for the analysis of ASME Section III, Class 1, 2, and 3 (Reference 18) as well as ASME B31.1 and B31.3 piping systems (References 19 and 20).

3.9.1.2.1.15 REFORC

REFORC determines flow-induced forces in piping system by serving as a post-processor to a thermal hydraulic transient code, RELAP5/MOD3. See Subsection 3.9.10, Reference 21, for more information.

3.9.1.2.2 Reactor Internals, Fuel and CEDMs

The following computer programs are used in the static and dynamic analyses of reactor internals, fuel, and CEDMs.

3.9.1.2.2.1 ANSYS

ANSYS is a general-purpose linear and nonlinear finite element program with structural and heat transfer capabilities, and is described in Reference 22. Finite element analyses of reactor internal structures such as flanges and the lower support structure are performed with ANSYS to determine vertical and lateral stiffnesses. The program is also used to

perform the static and dynamic analyses of the reactor internals to determine its structural stress responses.

The developer, ANSYS, has published an ANSYS verification manual with numerous examples of its usage.

3.9.1.2.2.2 ASHSD

ASHSD is used to obtain the dynamic response of the core support barrel under normal operating conditions and loss-of-coolant accident (LOCA). The program yields the dynamic shell and beam mode response of the structural system.

ASHSD has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted experimental tests or analytical results. The details of these comparisons are provided in References 23 and 24.

3.9.1.2.2.3 CESHOCK

The CESHOCK program is used to obtain the transient response of the reactor internals and fuel assemblies due to pipe break and seismic loads.

The computer program CESHOCK solves the equations of motion for the response of structures that can be represented by lumped-mass and spring systems and are subjected to a variety of arbitrary type loadings. Further description is provided in Reference 25.

CESHOCK has been verified by demonstration that its solutions are substantially identical to those obtained by hand calculations or from accepted analytical results via an independent computer code. The details of these comparisons are available in References 24 and 25.

3.9.1.2.2.4 CEFLASH-4B

The CEFLASH-4B computer code (Subsection 3.9.10, Reference 26) predicts the reactor coolant system pressure and flow distribution during the subcooled and saturated portion of the blowdown period of a LOCA. The equations for conservation of mass, energy, and

momentum, along with a representation of the equation of state, are solved simultaneously in a node and flow path network representation of the primary RCS.

CEFLASH-4B provides transient pressures, flow rates, and densities throughout the primary system following a postulated pipe break in the RCS.

The CEFLASH-4B computer code is a modified version of the CEFLASH-4A code (Subsection 3.9.10, References 27 and 28). The CEFLASH-4A computer code has been approved by the NRC (Subsection 3.9.10, References 29 through 31). The capability of CEFLASH-4B to predict experimental blowdown data is presented in Subsection 3.9.10, Reference 26.

3.9.1.3 Experimental Stress Analyses

When experimental stress analysis is used, it is performed in accordance with Appendix II of ASME Section III, Division I.

3.9.1.4 Consideration for the Evaluation of the Faulted Condition

3.9.1.4.1 Seismic Category I RCS Items

The major components of the RCS are designed to withstand the forces associated with the design basis pipe breaks described in Section 3.6 in combination with the forces associated with the safe shutdown earthquake (SSE), in-containment refueling water storage tank (IRWST) discharge load, and normal operating conditions. For structural evaluation, the design basis pipe breaks are those breaks for which a leak-before-break (LBB) cannot be demonstrated. Since the dynamic effects of breaks in piping systems listed in Subsection 3.6.3 are eliminated by LBB, the pipe break loads analysis procedure considers only those branch line pipe breaks not eliminated by an LBB.

See Subsection 3.9.3 for descriptions of loading combinations.

Analyses are performed to generate component loads and motions due to the forces associated with branch line pipe breaks. The analyses account for the reactor vessel and

supports, major connected piping and components and the reactor internals. The results of the analyses include loads on major component supports and RCS piping loads.

The analyses performed for branch line breaks use the ANSYS (Subsection 3.9.1.2.1.7) code.

The resultant component and support reactions are specified, in combination with the appropriate normal operating and seismic reactions, for design verification by the methods described below and in Subsection 3.9.3.

The system or subsystem analysis used to establish or confirm loads that are specified for the design of components and supports is performed on an elastic basis.

When an elastic system analysis is used to establish the loads that act on components and supports, elastic stress analysis methods are also used in the design calculations to evaluate the effects of the loads on the components and supports. In particular, inelastic methods such as plastic instability and limit analysis methods, as defined in the ASME Section III, are not used in conjunction with an elastic system analysis. The RCS and associated supports, which are analyzed using elastic methods, are shown in Figure 3.9-1.

Inelastic methods of analysis are used in cases where deemed desirable and appropriate to permit significant local inelastic response. In these cases, if any, the system or subsystem analysis performed to establish the loads that act on components and component supports are modified to include the inelastic strain compatibility in the local regions of the components and component supports at which significant local inelastic response is permitted.

Inelastic methods defined in the ASME Section III as plastic instability or limit analysis methods are not used.

3.9.1.4.1.1 Non-Code Items

The components not covered by the ASME Code but related to plant safety include:

a. Reactor internal structures (Class IS)

b. Fuel

c. Control element drive mechanisms (CEDMs)

d. Control element assemblies (CEAs)

Each component is designed in accordance with the relevant criteria to provide reasonable assurance of its operability as related to safety. The fuel assembly and CEA design is described in Section 4.2. The non-code components of the CEDMs are proven by testing as described in Subsection 3.9.4.4.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

Piping vibration, thermal expansion, and dynamic effects are tested during the initial test program (ITP) as delineated in Section 14.2. The ITP of piping systems is applicable to the following systems:

a. ASME Section III Class 1, 2, and 3 piping systems

b. High-energy piping systems inside seismic Category I structures

c. High-energy portions of piping systems whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable level.

d. Seismic Category I portions of moderate-energy piping systems located outside the containment.

The ITP is conducted in accordance with the ASME OM (Reference 32).

The ITP is implemented to demonstrate that these piping systems, restraints, components, and supports have been designed to withstand flow-induced dynamic loading under the steady-state and operational transient conditions anticipated during service, to confirm that proper allowance for thermal contraction and expansion is provided, and to demonstrate

that piping vibrations are within the acceptable level. The supports and restraints necessary for operation during the life of the plant are considered to be parts of the piping system.

The ITP includes a list of systems, flow modes, and selected locations for visual inspections and other measurements, the acceptance criteria, and possible corrective actions if excessive vibration or indications of thermal motion restraint occur.

The proper installation and operation of snubbers is verified through visual inspections, hot and cold position measurements, and observation of thermal movements during the ITP. The list of snubbers on systems that experience sufficient thermal movements from cold to hot position is provided as part of the ITP to measure snubber travel. In addition, the ITP includes the procedure necessary to verify the snubber operability when snubber travel is not measured.

3.9.2.1.1 Steady-State Vibration

The above piping systems in Subsection 3.9.2.1 with the potential to experience significant vibration are monitored for steady-state vibration.

The details relating to this test are described in the test procedure prepared in accordance with ASME OM (Reference 32), Part 3. The piping is monitored for normal operating and test modes along with operating modes expected to result in the most severe vibration. The piping is visually inspected and vibration movements are measured using portable instrumentation at locations where the vibration is determined to be the most severe. The piping, if necessary, is instrumented and monitored remotely.

The measured piping displacements are compared with allowable displacement limits that are based on the allowable stress amplitudes, S_{alt}, calculated in accordance with ASME OM, Part3. S_{alt} is limited as defined below.

a. For ASME Section III Class 1 piping systems

$$S_{alt} = \frac{C_2 K_2}{Z} M \leq \frac{S_{el}}{\alpha}$$

Where:

C_2 = secondary stress index as defined in ASME Section III

K_2 = local stress index as defined in ASME Section III

M = maximum zero to peak dynamic moment loading due to vibration only, or in combination with other loads, as required by the system design specification

S_{el} = 0.8 S_A, where S_A is the alternating stress at 10^6 cycles in psi from ASME Section III, Appendices, Figure I-9.1; or S_A at 10^{11} cycles from the ASME Section III, Appendices, Figure I-9.2.2. The user considers the influence of temperature on the Modulus of Elasticity.

Z = section modulus of the pipe

α = allowable stress reduction factor: 1.3 for materials covered by the ASME Section III, Appendices, Figure I-9.1; or 1.0 for materials covered by the ASME Section III, Appendices, Figure I-9.2.1 or I 9.2.2

b. For ASME Section III Class 2 and 3 piping systems and ASME B31 piping systems

$$S_{alt} = \frac{C_2 K_2}{Z} M \leq \frac{S_{el}}{\alpha}$$

Where:

C_2K_2 = $2i$

i = stress intensification factor, as defined in ASME Section III, Subsections NC and ND or ASME B31

The allowable stress reduction factor provides reasonable assurance that the alternating stress S_{alt} is based on the number of cycles during the design life.

If the measured piping displacements exceed allowable limits, one or more of the following actions are taken so that the vibration can be qualified.

a. Analyses are performed to show that the measured displacements are acceptable.

b. Additional testing is performed to show that the peak stresses due to the vibration are acceptable.

c. The source of the excessive vibrations is eliminated.

d. The pipe supporting arrangement is modified to reduce the vibration within acceptable levels.

3.9.2.1.2 Dynamic Transient Vibration

The dynamic transient vibration differs from steady-state vibration in that it occurs in a relatively short period of time and is usually generated by much larger forces.

In piping systems, the dynamic transient vibrations are most evaluated on the basis of pipe deflections and reactions. The primary cause of the dynamic transient vibrations is a high-pressure or low-pressure pulse traveling through the fluid. The dynamic transient vibrations are usually induced by rapid start or trip of a pump or turbine, and the quick closing or opening of valves such as turbine-stop valves and various types of control valves. The dynamic transients also occur as a result of rapid actuation of safety/relief valve (SRV) opening or as a result of unexpected events, such as condensed water accumulating at a low point in steam piping during a plant outage.

The operational dynamic transient condition having significant impact on the piping system is included in the test. The piping system is verified to operate appropriately by monitoring piping and pipe supports, including snubbers, subjected to the effects of the operational dynamic transient condition during the test.

The piping is instrumented to measure the system response during the dynamic transient events. The measured responses are compared with analytically predicted values from the piping stress reports.

If excessive system vibration is apparent during the dynamic transient events, an evaluation is performed to determine the cause and to identify the corrective action.

Alternatively, an analysis may be performed to demonstrate that the measured dynamic transient vibration does not cause the piping system in question to exceed stress or fatigue acceptance criteria.

3.9.2.1.3 Thermal Expansion

For piping systems expected to experience significant thermal movements, the thermal expansion test is performed to verify that the piping system expands and contracts within the acceptable limits based on analytically predicted movements from the piping stress analyses during the ITP and is performed in accordance with the requirements of ASME OM, Part 7.

Prior to heatup, the locations of potential thermal interferences are identified and appropriate corrective restraints are installed through a pre-heatup walkdown. One complete thermal cycle (i.e., cold to hot position and back to cold position) is monitored.

The piping and components are visually inspected and piping displacements are monitored at predetermined locations. The measurement locations are based on those of snubbers, hangers, and expected large displacements.

3.9.2.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

3.9.2.2.1 Seismic Qualification Testing

The recommended guidance and requirements in NRC RG 1.100 (Reference 33) and IEEE Std. 344-2004 (Reference 34) are used for the development and implementation of methods and procedures for seismic qualification of mechanical and electrical equipment. The seismic qualification testing methods for safety-related mechanical equipment are described in Subsection 3.10.2.2.

3.9.2.2.2 Seismic System Analysis Methods

The seismic system analysis methods (e.g., response spectra analysis, time-history analysis, equivalent static load analysis) are described in Subsections 3.7.2 and 3.7.3. The method

of analysis for piping and supports is described in Section 3.12. Seismic analysis methods for mechanical equipment and supports use the guidelines in IEEE Standard 344-2004 (Reference 35) and Subsection 3.10.2 and 3.10.3. The seismic analysis of mechanical 4equipment is performed by vendors to provide a seismic qualification report that demonstrates the structural integrity in accordance with the requirement of the design specification. The RCS seismic analysis is described in Appendix 3A.

3.9.2.2.3 Determination of Number of Earthquake Cycles

The OBE is chosen as 1/3 of the SSE for the APR1400 (see Section 3.7). When the OBE is less than or equal to 1/3 SSE, design or analysis is not required for the OBE.

With the elimination of the OBE, to account for fatigue in analysis and testing, the guidance for determination of the number of earthquake cycles described in SECY-93-087 (Reference 35) is used to account for fatigue in analysis and testing. For piping analysis, the number of earthquake cycles to be considered is defined in Subsection 3.7.3.1.

Alternatively, an equivalent number of fractional vibratory cycles to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987.

3.9.2.2.4 Basis for Selection of Frequencies

The seismic analysis is accomplished to account for the resonant frequencies and the seismic responses of structures, subsystems, and components in their design. The stiffness of the restraints and supports system is designed to be greater than the zero period acceleration (ZPA). The seismic responses of equipment and subsystems are maintained within the established limits. If the natural frequencies of the equipment and supporting structures are in the same range where resonance can occur, the resonance is considered for the seismic design.

3.9.2.2.5 Three Components of Earthquake Motion

The combination of three directional components of earthquake motion is in accordance with NRC RG 1.92 (Reference 36) as described in Subsections 3.7.2.6 and 3.12.3.2.

3.9.2.2.6 Combination of Modal Responses

The combination of modal responses is applicable when the response spectrum method of analysis is used, because the phase relationship between various modes is not identified and only the maximum responses for each mode are determined. Modal responses are combined by the methods described in Subsection 3.7.2.7.

For piping analysis, the guidance on combining the individual modal results in NRC RG 1.92 is used as described in Subsection 3.12.3.2.

3.9.2.2.7 Analytical Procedures for Piping

All seismic Category I and II piping is analyzed for seismic effects as described in Subsection 3.12.3.

3.9.2.2.8 Multiple-Supported Equipment Components with Distinct Inputs

When the equipment or component is supported at points with different elevations within a building and between buildings, either the envelope of these elevation response spectra or multiple supports excitation is used for the seismic qualification of the equipment.

For analyzing the piping systems supported at multiple locations within a single structure or multiple structures, the used method is described in Subsection 3.12.3.2.

3.9.2.2.9 Use of Constant Vertical Static Factors

A constant static factor is not used for the seismic design of seismic Category I structures, systems, and components specified in Subsections 3.7.2.10 and 3.7.3.6.

3.9.2.2.10 Torsional Effects of Eccentric Masses

All concentrated loads in a piping subsystem, such as valves and valve operators, are modeled as massless members with the mass of each component lumped at its center of gravity. Massless members are modeled by connecting the center of gravity of components to the centerline of piping so that the torsional effects of the eccentric masses are considered.

Torsional effects of eccentric masses are also considered in the analysis of seismic Category I subsystems other than piping.

3.9.2.2.11 Buried Seismic Category I Piping Conduits, and Tunnels

The seismic criteria and methods used to analyze buried seismic Category I piping, conduit, and tunnels are addressed in Subsections 3.7.3.7 and 3.12.3.8.

3.9.2.2.12 Interaction of Other Piping with Seismic Category I Piping

Interaction of other piping with seismic Category I piping is addressed in Subsection 3.12.3.7.

3.9.2.2.13 Analysis Procedure for Damping

The damping values used for seismic analysis are consistent with NRC RG 1.61 (Reference 37) as described in Table 3.7-7.

3.9.2.2.14 Test and Analysis Results

The test and analysis results are documented and available for review. The implementation program that includes milestones and completion dates is further described in Section 3.10.

3.9.2.3 Dynamic Response Analysis for Reactor Internals under Operational Flow Transients and Steady-State Conditions

The flow-induced vibration of the reactor internals components during normal operation can be characterized as a forced response to both deterministic and random pressure fluctuations in the coolant. Methods have been developed to predict the various components of the hydraulic forcing function and the response of the reactor internals to such excitation.

This analytical methodology is summarized in Figure 3.9-2. The method separates the response calculations into two groups in accordance with the physical nature of the loading. Methods for developing the deterministic component of the hydraulic forcing function are described in Subsection 3.9.2.3.1.1, while those relating to the random component are described in Subsection 3.9.2.3.1.2.

The responses of the reactor vessel core support and internal structures including core support barrel assembly, upper guide structure assembly, and lower support structure assembly to the normal operating hydraulic loads are calculated by finite element techniques. The mathematical models used in these response analyses are described in Subsection 3.9.2.3.2. The methods used in calculating the structural responses are described in Subsection 3.9.2.3.3.

3.9.2.3.1 Hydraulic Forcing Function

3.9.2.3.1.1 Deterministic Forcing Function

An analysis based on a hydrodynamic model is used to obtain the relationship between RCP pulsations in the inlet ducts and the deterministic pressure fluctuations on the core support barrel. A detailed description of this model and subsequent solution are given in References 38 and 39. The model represents the annulus of coolant between the core support barrel and the reactor vessel. In deriving the governing hydrodynamic differential equation for the model, the fluid is taken to be compressible and inviscid. Linearized versions of the equations of motion and continuity are used. The excitation on the hydraulic model is harmonic with the frequencies of excitation corresponding to pump rotational speeds and blade passing frequencies.

The dynamic force on the upper guide structure assembly is due to flow-induced forces on the tube bank. The deterministic components of these forces are caused by pressure

pulsations at harmonics of the pump rotor and blade passing frequencies, and vortex shedding due to crossflow over the tubes.

The in-core instrumentation (ICI) nozzles and the skewed beam supports for the ICI support plate of the lower support structure are excited by deterministic and/or random, flow induced forces. The deterministic component of this loading is due to pump-related pressure fluctuations and vortex shedding due to crossflow.

Data from the System 80 pre-operational test (References 40 and 41) is used to determine the magnitude of these pulsations at the pump rotor and blade passing frequencies and their harmonics.

3.9.2.3.1.2 Random Forcing Function

The random hydraulic forcing function is developed by experimental methods. The forcing function is represented in the form of power spectral density together with associated coherence area. The forcing function is modified to reflect the flow rate and density differences based on an analytical expression found in Reference 42.

At normal operating conditions the shroud tubes of upper guide structure assembly are excited by upstream and wake produced turbulent buffeting. The forcing function for this type of loading can be represented as a band limited white noise power spectrum.

The ICI nozzles and ICI support plate support beams of the lower support structure assembly are both subject to turbulent buffeting by the flow skirt jets. The outermost ICI nozzles and beams receive full impact of the jets before the jets decay due to fluid entrainment and the presence of inner tube rows. The force spectrum of these jets is assumed to be represented as wide band white noise. The magnitude of this spectrum is computed based on data from the System 80 pre-operational tests.

3.9.2.3.2 Mathematical Models

A finite element analysis is performed on each of the reactor internals components using mathematical models. These models are designed to provide the most efficient analysis under the most significant loading condition to which each structure is exposed. The core

support barrel (CSB) assembly is modeled as a shell using the ASHSD computer code (Reference 23) as shown in Figure 3.9-3. The structure is fixed at the upper flange to determine the beam modes and frequencies. The shell modes and frequencies are found by considering the upper flange fixed and the lower flange pinned. These analyses include hydrodynamic mass effects. All significant mode shapes and frequencies are used in combination to perform the normal operating deterministic response analysis. A simplified finite element model of the barrel assembly is generated on the ANSYS (Reference 22) for use in the random response analysis.

The inner barrel assembly is modeled as plate and solid elements using ANSYS computer code as shown in Figure 3.9-4. The model is used to determine the static and dynamic characteristics as well as periodic and random response analyses.

The control element assembly (CEA) guide tubes, fuel alignment plate and UGS support plate are modeled as beam and plate elements using the ANSYS code (Figure 3.9-5). The model includes additional details in the regions of selected CEA guide tubes and HJTC tubes to analyze detailed responses of these tubes.

The lower support structure is modeled as plate elements using the ANSYS computer code to determine the modes, natural frequencies, and responses. The in-core instrumentation assembly (Figure 3.9-6) is modeled as plate, beam, mass and spring elements using ANSYS to analyze the dynamic characteristics and responses.

3.9.2.3.3 Response Analysis

3.9.2.3.3.1 Periodic Response

The normal mode method (Reference 43) is used to obtain the structural response of the reactor internals to the periodic forcing functions developed in Subsection 3.9.2.3.1.1. The method is applied to the appropriate finite element models described in Subsection 3.9.2.3.2. Generalized masses based on mode shapes and the mass matrices are calculated for the modes of vibration of each component. Modal force participation factors are based on the mode shapes and the predicted periodic forcing functions are calculated for each mode and forcing function. The generalized coordinate response for each mode is then obtained from the solution of the corresponding set of independent second order single-degree of freedom equations. Using mode shapes, the modal responses of the reactor

internals are obtained by means of the appropriate coordinate transformations. Response to any specific forcing function is obtained through summation of the component modes for that forcing function.

3.9.2.3.3.2 Random Response

The normal mode method (Reference 43) is used to obtain the structural response of the reactor internals subjected to random forcing functions. The random forcing functions are assumed to be of both the band limited and wide band white noise varieties as described in Subsection 3.9.2.3.1.2. Experimental and analytical expressions are used to define the force power spectral density associated with flow related turbulence and jet impact. The appropriate mathematical models described in Subsection 3.9.2.3.2 are used in the ANSYS (Reference 22). This code computes the root mean square (RMS) displacements, loads and stresses in a multi-degree-of-freedom linear elastic structural model subjected to stationary random dynamic loadings, such as those described in Subsection 3.9.2.3.1.2.

A value of 3 × RMS is used for considering peak responses to random loading. These peak values are then combined with results from other analyses and used in design verification analyses. The use of the value 3 × RMS is common design practice based upon the assumptions of Random Gaussian loading of structures made of ductile materials as described in Reference 44.

3.9.2.4 Preoperational Flow Induced Vibration Testing of Reactor Internals

In accordance with NRC RG 1.20 (Reference 45), a comprehensive vibration assessment program (CVAP) is conducted for reactor internals. The APR1400 is classified as non-prototype Category I, as for NRC RG 1.20, with Palo Verde Unit 1, a Westinghouse System 80 Reactor as the valid prototype (Reference 41). According to CVAP Report of System 80 Reactor Internals (Reference 41), evaluation of the comparisons of analytical predictions, test measurements, and visual inspection results leads to the conclusion that the System 80 Prototype Reactor Internals are structurally adequate and acceptable for long-term operation. Palo Verde Unit 1 design and the APR1400 design are substantially the same with regard to arrangement, design, size, and operating conditions. A comparison of the design arrangement is provided in Table 3.9-15. The comparison of nominal dimensions for the reactor internals is shown in Table 3.9-16. The comparison of operating conditions for the reactor internals is shown in Table 3.9-17.

The CVAP for the APR1400 design consists of the analysis program and inspection program. The analysis program is described in Subsection 3.9.2.3.

The inspection program consists of a pre-hot functional test inspection and a post-hot functional test inspection of the reactor internals. The duration of the hot functional test is established to provide reasonable assurance that 10^6 cycles of vibration have occurred before the post-hot functional inspection. A detailed inspection of major load bearing surfaces, contact surfaces, welds, and maximum stress locations identified in the analysis program is performed. All observations made during the pre-hot and post-hot functional test inspections are documented. A comparison of the structures is made to verify that no loss in structural integrity due to flow-induced vibrations has occurred.

The evaluation, consisting of the analysis program and inspection program, confirms the adequacy of the analysis prediction techniques and the structural integrity of the APR1400 design according to the guidance of NRC RG 1.20. The COL applicant is to provide the inspection results for the APR1400 reactor internals classified as non-prototype Category I in accordance with NRC RG 1.20 (COL 3.9(1)).

NRC RG 1.20, Revision 3, recommends that the potential adverse effects from pressure fluctuations and vibrations in piping systems be considered for the steam generator internals in PWRs, which may be adversely affected by the flow-excited acoustic resonances and flow-induced vibrations. The steam dryers and separators in the APR1400 SG are subject to secondary side steam flow. Although there are instances of similar components in BWR experiencing excessive vibration resulting from plant power uprate, none have been reported for PWR SG designs to date. This is further supported by the Operational Performance Information System for Nuclear Power Plant (OPIS) database, which also does not have any incidents related to the flow-induced vibration problems for PWR SG upper internals.

The design of the APR1400 SG upper internals and the flow conditions for which they are subjected are similar to the existing and currently operating SGs in the Republic of Korea and around the world. Based on the operational experience for the SG upper internals, KHNP concludes that these non-safety-related components will not experience excessive vibration. Consequently, no startup testing with measurement is planned for these components.

The vibration of the piping attached to the RCS as well as SG (main steam and main feedwater lines) is measured during initial startup testing. These measurements will be taken at discrete piping locations. Strain gages and accelerometers will be used to measure the vibration at selected locations for both steady-state and transient flow conditions.

The acceptance criteria for the piping will be based upon satisfying the appropriate displacement, acceleration, stress, and fatigue. The acquired data will be used to confirm that unexpected, abnormal vibrations do not occur, and that the vibration responses of the piping are sufficiently small compared to an acceptance criterion based on the design fatigue curves in ASME Section III.

3.9.2.5 Dynamic System Analysis of the Reactor Internals under Faulted Conditions

Dynamic analyses are performed to determine the maximum structural responses of the reactor internals including the core to postulated pipe breaks and seismic loadings and to verify the adequacy of their design.

The results of the analyses are required to meet the stress limits of ASME Section III, Subsection NG, for core support structures and the functional requirements of the reactor internals design specification. More information on the reactor internals design is provided Subsection 3.9.5.3.

3.9.2.5.1 Seismic Analysis of Reactor Internals

The seismic analyses of the reactor internals including the core are performed separately for the horizontal and vertical directions.

In the horizontal direction, because the relative displacements between the core and core shroud and between the core support barrel and reactor vessel snubbers are sufficiently large to close the gaps that exist between these components, a nonlinear horizontal time history analysis is performed. The horizontal nonlinear analysis is divided into two parts. In the first part, the internals and core are analyzed to obtain the response of the internals and the proper dynamic input for the reactor core model. In the second part, the core plate

motion from the first part is applied to a more detailed nonlinear model of the reactor core. The input excitation to the internals model is the response time-history of the reactor vessel at the internals support determined from the RCS analysis. Coupling effects between the internals and reactor vessel are accounted for by including a simplified representation of the internals with the RCS model.

A nonlinear analysis is also performed in the vertical direction. Possible lifting of the core off the core support surface, friction between the fuel rods and spacer grids and fuel assembly holddown force are examples of nonlinear effects included in the vertical analysis.

In these analyses, two horizontal components and one vertical component of the seismic excitation are considered and the maximum responses for the three components are combined by the square root of the sum of the squares (SRSS) method.

Equivalent multi-mass mathematical models are developed to represent the reactor internals and core. The mathematical models of the internals are constructed in terms of lumped masses and linear elastic-spring elements. At appropriate locations within the internals and core, nodes are chosen to lump the weights of the structure. The criterion for choosing the number and location of mass concentration is to provide accurate representation of the significant modes of vibration of each of the internals components. Between the nodes, properties are calculated for moments of inertia, cross-sectional areas, effective shear areas, and lengths. The model definitions employ the procedures established in Westinghouse Topical Report CENPD-178 (Reference 46) and include hydrodynamic coupling effects. Separate horizontal and vertical models of the internals and core are formulated to more efficiently account for structural differences in these directions. In the horizontal nonlinear lumped mass representation of the internals and core, shown in Figure 3.9-16, gap and spring elements are used to represent contact between the fuel and core shroud. Lumped-mass nodes in the core are positioned to coincide with fuel-spacer grid locations. To simulate the nonlinear motion of the fuel, nonlinear spring couplings are used to connect corresponding nodes to the fuel assemblies and core shroud. The core is modeled by subdividing it into fuel assembly groupings and choosing stiffness values to adequately characterize its beam response and contacting under dynamic loading.

The horizontal nonlinear reactor core model consisting of one row of 17 individual fuel assemblies is depicted in Figure 3.9-17. In this model, each fuel assembly is represented

with mass points located at spacer grid locations. To simulate the gaps in the core, nonlinear spring couplings are used to connect corresponding nodes on adjacent fuel assemblies and core shroud. The impact stiffness and impact damping parameters for the gap elements are derived from the impact tests, which are described in Section 4.2. The spacer grid impact representation used for the analysis is capable of representing two types of fuel assembly impact situations. In the first type, only one side of the spacer grid is loaded. This type of impact occurs when the peripheral fuel assembly hits the core shroud, or when two fuel assemblies strike one another. The second type of impact loading occurs typically when the fuel assemblies pile up on one side of the core. In this case, the spacer grids are subjected to a through-grid compressive loading.

The fuel assemblies in the coupled core/internals model and the detailed core model are modeled with beam elements to represent the horizontal stiffness between mass points and rotational springs at each end to simulate the end fixity existing at the top and bottom of the core. The value used for fuel horizontal stiffness and end fixity is based upon a parametric study in which analytic predictions are correlated with fuel assembly static and dynamic test data. Fuel assembly structural damping as a function of vibrational amplitude is derived from fuel assembly forced vibration and pluck tests. The damping values used in the seismic analysis of the reactor internals are in accordance with the values in Table 3.7-7.

The vertical nonlinear model is shown in Figure 3.9-18. The vertical model stiffness values are generally calculated using bar characteristic equations. Nonlinear couplings are included between components to account for structural interactions such as those between the fuel and core support plate, and between the core support barrel and upper guide structure upper flanges. Preloads, which are caused by the combined action of applied external forces, dead weights, and holddown forces, are also included. Friction elements are used to simulate the coupling between the fuel rods and spacer grids.

It has been shown both analytically and experimentally (Reference 47) that immersion of a body in a dense-fluid medium lowers its natural frequency and significantly alters its vibratory response as compared to those in air. The effect is more pronounced when the confining boundaries of the fluid are close to the vibrating body as in reactor internals. The method of accounting for the effects of a surrounding fluid on a vibrating system has been used to describe the system with additional or hydrodynamic mass. The hydrodynamic mass of an immersed system is a function of the dimensions of the real mass

and the space between the real mass and confining boundary. Hydrodynamic mass effects for moving cylinders in a water annulus are discussed in References 47 and 48.

The nonlinear seismic response and impact forces for the internals and fuel are determined using the CESHOCK or ANSYS computer program (see Subsection 3.9.1.2.2.3). The computer programs provide numerical solution to transient dynamic problems by step-by-step integration of the differential equations of motion. The input excitation for the model is the time-history accelerogram of the reactor vessel.

Input to the computer program consists of initial conditions, nodal lumped masses, linear-spring coefficients, mass moments of inertia, nonlinear spring curves, and the acceleration time-histories. The output from the computer program consists of displacements, velocities, translational and angular accelerations, impact forces, axial forces, shears, and moments.

The procedures used to account for damping in the analysis of the reactor internals and core are provided in Subsection 3.7.2.15.

The nonlinear response loads for the internals, including impacting, if any exist, are determined for the vertical and horizontal directions. Vertical loads for the fuel are determined in the coupled model of internals and core and horizontal loads for the fuel are determined in a separate reactor core nonlinear analysis. The results are determined for the SSE.

3.9.2.5.2 Pipe Rupture Analysis of the Reactor Internals

3.9.2.5.2.1 Input Excitation

According to the application of LBB, the postulated pipe breaks for the reactor coolant system, reactor internals and fuel assembly are defined in Subsection 3.6.2. The secondary side breaks, which imparts vibratory motion to the reactor internals does not cause blowdown loads within the reactor vessel. However, the primary side break causes blowdown loads as well as vibratory motion. The blowdown loads consist of transient pressure, flow rate, and density distributions throughout the primary RCS.

The transient pressure, flow rate and density distributions are computed for the subcooled and saturated portions of the blowdown period during the pipe breaks. The computer code that is used is based on a node flow path concept in which control volumes (nodes) are connected in any desired manner by flow areas (flow paths). A complex node-flow path network is used to model the primary coolant system. The modeling procedure has been compared to a large-scale experimental blowdown test with excellent agreement. The laws of conservation of mass, energy, and momentum along with a representation of the equation of state are solved simultaneously. The hydraulic transient of the reactor is coupled to the thermal response of the core by analytically solving the one-dimensional radial heat conduction equation in each core node. Pre-blowdown steady-state conditions in the primary coolant system are established through the use of specified input quantities. The blowdown loads model uses a non-equilibrium critical flow correlation for computing the subcooled and saturated critical fluid discharge through the break.

A break in the primary coolant system results in large local pressure differences across various reactor internal components and an acceleration of the local fluid velocity in various regions. The acceleration of the local fluid velocity can result in higher component drag loads than during steady-state reactor operation.

The total instantaneous load across the core is given by the summation of the pressure forces acting in the direction of the pressure gradient and the drag forces acting parallel to the flow. The loads are obtained using a control volume approach using an integrated fluid momentum equation. The drag forces are represented by the fluid shear term in this equation and consist of both frictional and form drag.

3.9.2.5.2.1.1 Reactor Internals Asymmetric Blowdown Loads

During an inlet break, dynamic lateral loads are developed on the core support barrel by the time varying radial pressure disturbances. These are highly asymmetric in the circumferential direction and are caused by the expansion of wave fronts and flow redistributions acting on the surface of the barrel. In order to obtain these applied lateral loads, the time-dependent Fourier coefficients that define the circumferential pressure distributions are first obtained. A linear variation in pressure is assumed to act between each elevation. The sine and cosine pressure distributions produce a response in the barrel, which is analogous to a beam-bending mode. When integrated over the surface area of the

core support barrel, these pressures produce resultant lateral loads that are applied to the model.

3.9.2.5.2.1.2 Control Element Tube Loads

During normal operation, the reactor coolant flows axially through the core into the upper guide structure. Within the upper guide structure, the coolant flow changes direction so that it exits through the hot leg nozzles.

During the pipe breaks, the transverse flow of the coolant across the control element guide tubes gives rise to loads that induce deflections in these tubes. The transverse drag forces are determined from flow model experiments that are geometrically and dynamically similar to the full scale upper guide structure design for the System 80.

3.9.2.5.2.1.3 Vertical Hydraulic Loads

The vertical hydraulic loads on the reactor internals due to postulated pipe breaks are considered. The loads are obtained using a control volume approach using an integrated fluid momentum equation. This is based on control volumes that are consistent with both fluid volumes and the lumped mass nodes of the vertical analysis model.

The resulting reactor internals blowdown loads are used in subsequent analyses to determine the structural response of the internals and fuel.

3.9.2.5.2.2 Analysis Model

The horizontal model of the reactor internals used for the pipe break model is similar to the seismic model except for additional detail in the CSB and UGS upper flange region which allows for relative displacements and rotations of these components and the reactor vessel ledge. Friction, gap, and hysteresis elements are used to model the complex interactions of these components. The horizontal model including this region is shown in Figure 3.9-19. The same model with seismic analysis (Figure 3.9-18) is used for pipe break analysis in the vertical direction.

3.9.2.5.2.3 Analysis Method

The nonlinear pipe break response and impact forces for the internals and fuel are determined using the CESHOCK or ANSYS computer program. The computer programs provide the numerical solution to transient dynamic problems by step-by-step integration of the differential equations of motion. The input excitation for the model is the time-history accelerogram of the reactor vessel.

Input to the computer program consists of initial conditions, nodal lumped masses, linear-spring coefficients, mass moments of inertia, nonlinear spring curves, and the acceleration time-histories. The output from the computer program consists of displacements, velocities, translational and angular accelerations, impact forces, axial forces, shears, and moments.

A cold leg branch line break causes a pressure transient on the core support barrel (CSB) that varies circumferentially as well as longitudinally. The ASHSD finite element computer code is used to analyze the shell response of the CSB to the pressure transient from a cold leg branch line break. The CSB is modeled as a series of shell elements joined at their nodal point circles, as shown in Figure 3.9-3. The length of the elements in each model is selected to be a fraction of the shell attenuation length.

A damped equation of motion is formulated for each degree of freedom of the system. Four degrees of freedom, radial displacement, circumferential displacement, vertical displacement, and meridional rotation are considered in the analysis. The differential equations of motion are solved numerically using a step-by-step integration procedure.

The circumferential variation of the pressure time-history is considered by representing the pressure as a Fourier expansion. The pressure at each node in the model is determined by linear interpolation. Thus a complete spatial time load distribution compatible with the ASHSD computer program is obtained. Each load harmonic is considered separately by ASHSD. The results for each harmonic are then added to obtain the nodal displacements, resultant shell forces and shell stresses as a function of time.

3.9.2.5.3 In-containment Refueling Water Storage Tank Discharge Analysis

The reactor internals and core are analyzed for the IRWST discharge loads.

The analytical models and the procedures of the reactor internals and core and the detailed core for IRWST discharge loads are the same as those used in the seismic analyses.

The input excitation to the reactor internals and core model is the acceleration time histories of reactor vessel flange and snubbers determined from the RCS IRWST analysis.

The maximum stresses resulting from IRWST, pipe break, and SSE are combined using the SRSS method to obtain the total stress intensities.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

The results of the dynamic analysis of the reactor internals are compared with the results of preoperational tests of the prototype reactor (Reference 41) since the APR1400 reactor internals are classified as non-prototype Category I as described in Subsection 3.9.2.4.

The dynamic analysis models for the faulted condition are developed using the dynamic characteristics measured in the test of the prototype.

3.9.2.7 Dynamic System Analysis of the CEDM

The pressure-retaining components of the control element drive mechanism (CEDM) are designed to the appropriate stress criteria of ASME Section III for all loadings specified. The structural integrity of the CEDM for the seismic loadings is verified by combination of test and analysis. Methods of dynamic analysis using response spectrum analysis or time history analysis are supported with experimentally obtained information.

3.9.2.7.1 Input Excitation Data

For the dynamic analyses, response spectra or time history definition of the excitation at the base of the CEDM nozzle is obtained from the seismic analysis of the RCS. The

excitation is applied simultaneously in three mutually perpendicular directions (two horizontal and one vertical).

3.9.2.7.2 Analysis

A dynamic analysis of the mathematical structural model is performed using one or more of the computer programs described in Subsection 3.9.1.2.

3.9.2.7.3 Functional Test

A functional test using a minimum drop weight was performed to verify that drop characteristics meet the input design requirements. Results from this test are compared to the calculated CEDM deflections under seismic loading for the individual site. Verification of the proper function is thus established based on both analytical and test results.

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

This section describes the structural integrity of pressure-retaining components, component supports, and core support structures that are designed and constructed in accordance with the rules of the ASME Section III, Division 1, and GDC 1, 2, 4, 14, and 15.

The APR1400 design meets the SRP 3.9.3 criteria as described in the following respects:

a. 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1, as they relate to structures and components being designed, fabricated, erected, and tested to quality standards commensurate with the safety-related functions to be performed. These requirements provide reasonable assurance that safety-related components and structures meet service loading conditions, stress limits, and quality requirements of ASME Code permitted in 10 CFR 50.55a.

b. GDC 2 and 10 CFR 50, Appendix S, as they relate to structures and components important to safety being designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. The effects of expected

natural phenomena on the normal and accident conditions are considered in the loading combinations for structures and components important to safety.

c. GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including lost-of-coolant accidents. The safety-related structures and components are protected against dynamic effects including LOCA by considering appropriate loading combinations as described in this section.

d. GDC 14 as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The RCPB components are designed in accordance with the ASME Code requirements that follow with GDC 14.

e. GDC 15 as it relates to the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to provide reasonable assurance that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. These requirements provide reasonable assurance that the RCS and associated auxiliary system components meet ASME Code requirements.

f. 10 CFR 52 as it requires that components, component supports, and core support structures be designed and built in accordance with the certified design. Reasonable assurance that the design for the components, component supports, and core support structures is certified is provided by adhering to the applicable ASME Code requirements.

g. 10 CFR 52.47(b)(1), which requires that a DC application contains the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, then a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. Raesonable assurance that the requirements for the components,

component supports, and core support structures are met is provided by adhering to the the applicable ASME Code requirements and ITAACs.

h. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee will perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility is constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act and NRC's regulations. The proposed inspections, tests, and analyses are described in Section 14.3 and those applicable to emergency planning in Section 13.3.

In accordance with ASME Code, design specifications and design reports are to be provided for ASME Section III Class 1, 2, and 3 components and piping. The design specification defines the jurisdictional boundary for the NF portion of the piping support. The supports are designed in accordance with *[ASME Section III, Subsection NF]** and meet its requirements.

Welding activities are performed in accordance with the requirements of ASME Section III. Component supports are fabricated in accordance with the requirements of *[ASME Section III, Subsection NF.]**

When dissimilar metal welding joints are used for the fabrication of ASME Code Class components and piping, the weld materials and processes are selected in consideration of the joint degradation caused by the stress corrosion cracking in service environments.

Loading conditions, stress limits, design transients, and methods of analysis for ASME Code Class 1 reactor coolant loop piping and associated components and component supports are described in Subsection 3.9.3.1.

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

The loading combinations specified for the design ASME Section III Class 1 components, supports, and piping are categorized as Design, Level A, Level B, Level C, and Level D conditions. The following specific loading combinations are specified for design:

Design Loadings

Mechanical loads are combined as defined in (a) through (b) below.

a. Design pressure plus dead weight plus IRWST discharge

b. Installation or handling loads (only for RV)

Level A Service Loadings

The following loading combination is considered as Level A service loading.

Normal operation loads in combination with specified system operating transient loads resulting from the normal events.

In addition, handling loads alone at cold shutdown conditions are used to design only RV.

Level B Service Loadings

Normal operation (including dead weight) and IRWST discharge loads in conjunction with the upset transients are considered Level B service loadings.

The seismic cycles mentioned in Subsection 3.7.3 are applied in a fatigue analysis, with considering the effects of reactor coolant environment in accordance with NRC RG 1.207.

Level C Service Loadings

Level C service loadings are the concurrent loadings associated with the Level C (emergency) condition.

Level D Service Loadings

Normal operation (including dead weight), SSE, IRWST discharge, branch line pipe breaks, other accident loads, and faulted transients are combined as defined below.

a. Normal Operation + $[SSE^2 + (BLPB + IRWST)^2]^{1/2}$ in conjunction with the faulted transients

b. Normal Operation + $[SSE^2 + (POSRV + IRWST)^2]^{1/2}$ in conjunction with faulted transients (only for the pressurizer)

The SSE and pipe rupture loadings plus IRWST discharge loads are combined by the SRSS method in accordance with the guidelines of NUREG-0484, Rev. 1, 1980, or by a more conservative method.

Test Loadings

Test loadings are the concurrent test pressure in conjunction with hydrostatic dead weight.

The specific design transients specified for design are described in Subsection 3.9.1.1. The loading combinations and the stress limits associated with them, as defined in the ASME code, also apply to the internals parts which are essential to the component in performing its safety function.

ASME Code Class 1, 2, and 3 piping and components of fluid systems are designed and constructed in accordance with ASME Section III. Hydrostatic testing is performed per ASME Section III.

Design pressure, temperature, and other loading conditions that provide the bases for design of fluid systems are presented in the sections that describe the systems.

Stress analysis and fatigue evaluations are performed to determine structural adequacy of pressure components under the operating conditions of normal, upset, emergency, or faulted, as applicable.

Significant structural discontinuities in parts such as nozzles and flanges are considered. In addition to the design calculation required by ASME Section III, stress analysis is also performed by methods outlined in the code appendices or by other methods by reference to analogous codes or other published literature.

Thermal stratification and striping are described not to create adverse effect on the associated piping in Subsection 3.12.5.10.

The loading combinations for the design of ASME Code Class 2 and 3 components and supports, CEDM, and reactor internals are described in Subsections 3.9.3.1.3, 3.9.4, and 3.9.5, respectively.

The COL applicant is to provide a summary of the maximum total stress, deformation, and cumulative usage factor values for each of the component operating conditions for ASME Code Class 1 components. For those values that differ from the allowable limits by less than 10 percent, the COL applicant is to provide the contribution of each of the loading categories (e.g., seismic, dead weight, pressure, and thermal) to the total stress for each maximum stress value identified in this range.

The COL applicant is to also provide a summary of the maximum total stress and deformation values for each of the component operating conditions for Class 2 and 3 components required to shut down the reactor or mitigate consequences of a postulated piping failure without offsite power (with identification of those values that differ from the allowable limits by less than 10 percent) (COL 3.9(2)).

3.9.3.1.1 ASME Code Class 1 Components and Supports

Design transients for ASME Code Class 1 components and component supports are described in Subsection 3.9.1.1. Loading combinations, for ASME Code Class 1 components are described in Table 3.9-2. Stress limits for ASME Code Class 1 components, supports, and piping are described in Table 3.9-3. The operating pressures of Code Class 1 active valves are limited to the pressures taken from the pressure-temperature rating of the ASME Section III for the maximum temperature of the given condition.

3.9.3.1.2 Reactor Internals

Design transients for reactor internals are described in Subsection 3.9.1.1. Loading combinations and stress limits are presented in Subsection 3.9.5.

3.9.3.1.3 ASME Code Class 2 and 3 Components and Supports

ASME Section III, Subsections NC and ND, provide design requirements for ASME Class 2 and 3 components. Loading combinations applicable to these Class 2 and 3 components and supports are described in Table 3.9-2. The stress criteria for the components are presented in Tables 3.9-5 through 3.9-9.

The Class 2 and 3 components that are subject to thermal or dynamic cyclic loads are evaluated for their fatigue sustainability using the ASME Section III NC-3219.2, as per NUREG-0800, SRP 3.9.3. A fatigue analysis is also performed in accordance with NC-3200 for the components that do not meet the NC-3219.2 criteria.

Stresses in valve bodies and pump casings are limited to the elastic limit of a material when the pump or valve is subjected to a combination of normal operating loads, seismic, and other applicable loads.

The functionality and operability of Class 2 and 3 safety-related active components are confirmed under combined service loadings at a stress level bounded by the specified service limit. The operability of active valves and pumps is described in Subsection 3.9.3.3.2.

In addition, the integrity of the pressure boundary for these components is properly maintained by verifying that calculated stresses are lower than their corresponding stress limits.

3.9.3.1.4 Preparation of Design Specifications

Design specifications for ASME Section III components and supports are prepared using guidelines given in Reference 33.

3.9.3.2 Design and Installation of Pressure-Relief Devices

The pressure relieving valves are designed in accordance with the requirements of ASME Section III (Reference 18) Appendix O. Where more than one valve is installed on the same pipe run, the sequence of valve openings to be assumed in analyzing for the stress at any piping location is estimated to induce the maximum instantaneous valve of stress at that location. The applicable stress limits are satisfied for all components of the pipe run and connecting systems, and the pressure relief valve station, including supports. After the dynamic structural system analysis, a dynamic load factor affects the reaction forces and moments and a dynamic load factor of 2.0 is used in lieu of a dynamic analysis to determine the dynamic load factor.

3.9.3.2.1 Pressure Relief Devices Connected to the Pressurizer

The pressurizer pilot-operated safety relief valves (POSRVs) are designed to provide overpressure protection for the RCS. The POSRVs connected to the pressurizer are the only ASME Section III (Reference 18), Class 1 pressure relief valves in the APR1400.

Four pressurizer POSRVs are connected to the top of the pressurizer by separate inlet lines. There are two main discharge lines to the in-containment refueling water storage tank (IRWST). The steam from two POSRVs is discharged through one common discharge line. Each pressurizer POSRV provides the overpressure protection function with a main valve and two spring-loaded pilot valves in the assembly. The pressurizer POSRVs pass sufficient pressurizer steam to limit the RCS pressure to 110 percent of design pressure following a loss of load with delayed reactor trip, which is assumed to be initiated by the safety grade signal from the reactor protection system (RPS). A delayed reactor trip is assumed to occur on a high-pressurizer pressure signal.

The pressurizer POSRVs and their pilot operators are qualified to operate in saturated steam, water, and steam and water mixtures in hot or cold conditions.

Details on the design of the POSRVs are provided in Subsections 5.2.2 and 5.4.13.

The opening of the POSRV introduces thermal-hydraulic transients in the discharge lines. The POSRVs discharge over-pressurized reactor coolant into piping that carries the flow

through a system of turns, straight segments, and area transitions or branches before expulsion at a remote location. Flow acceleration creates reactions forces.

For each main discharge line (two POSRVs), an analytical model is developed. The analytical model consists of the pressurizer, the valves, and the connecting piping from the pressurizer nozzles to the IRWST spargers. The pressurizer is modeled as a reservoir that is filled with saturated steam at constant pressure. For each straight pipe segment, a transient load is calculated. The analyses of the piping and support consider these reaction forces to be induced by the opening of the POSRVs.

3.9.3.2.2 Pressure Relief Devices for Class 2 Systems and Components

Pressure-relieving devices for ASME Section III (Reference 18), Class 2 systems include the main steam safety valves (MSSVs) on the steam line and the low temperature overpressure protection (LTOP) relief valves on the containment isolation portion of the normal shutdown cooling system (SCS).

The MSSVs are direct acting, spring loaded, carbon steel valves. The valves are mounted on each of the main steam lines upstream of the main steam isolation valves (MSIVs), outside the containment. The design and analysis requirements for the MSSVs and discharge piping for the steam line are described in Subsection 10.3.2.

A relief valve on each of the SCS suction lines provides LTOP for RCS in a failure that initiates the pressure transient while in shutdown cooling and also can prevent overpressurization of SCS. These relief valves are addressed in Subsection 5.4.7.

3.9.3.2.3 Pressure Relief Device Discharge System Design and Analysis

ASME Section III, Appendix O describes two types of discharge systems for pressure relief devices: open discharge systems and closed discharge systems. An open discharge system discharges fluid directly to the atmosphere or to a vent pipe that is open to the atmosphere. A closed discharge system is hard-piped to a distant location or closed tank. ASME Section III, Appendix O also describes the layout considerations and limits for both types of systems, as well as design equations and considerations for analysis of these systems. The APR1400 design complies with these requirements.

3.9.3.3 Pump and Valve Operability Assurance

3.9.3.3.1 Operability Assurance Program

Active pumps and valves are defined as pumps and valves and those components that perform a mechanical motion in order to shut down the plant, maintain the plant in a safe shutdown condition, or mitigate the consequences of a postulated event. The functional design and qualification of safety-related pumps and valves are performed in accordance with ASME QME-1-2007 (Reference 49) as endorsed by NRC RG 1.100 (Reference 33). The operability (performance of this mechanical motion) of active components during and after exposure to design bases events is confirmed by:

a. Designing each component to be capable of performing all safety functions during and following design bases events.

   The design specification includes applicable loading combinations, and conservative design limits for active components. The specification requires that the manufacturer demonstrate operability by analysis, by test, or by a combination of analysis and test.

b. Analysis and/or test demonstrating the operability of each design under the most severe postulated loadings.

   Methods of operability demonstration programs are detailed in Subsections 3.9.3.3.2 and 3.9.3.3.3.

c. Inspection of each component to provide reasonable assurance of critical parameter compliance with specifications and drawings.

   This inspection confirms that specified materials and processes are used, that wall thicknesses meet code requirements, and that fits and finishes meet the manufacturer's requirements based on design clearance requirements.

d. Shop testing of each component to verify as-built conditions, as defined in Subsections 3.9.3.3.2 and 3.9.3.3.3.

e. Startup and periodic in-service testing in accordance with ASME OM Code (Reference 33) to demonstrate that the active pumps and valves are in operating condition throughout the life of the plant.

The combined license (COL) applicant is to identify the site-specific active pumps (COL 3.9(3)).

3.9.3.3.2 Pump Operability

ASME Class 2 and 3 safety-related active pumps are listed in Table 3.9-17. The following criteria are employed in a qualification program to ensure operability of the pumps required to function during and following design bases events.

a. Analysis, test, or a combination of test and analysis are used in accordance with ASME QME-1-2007 (Reference 49) as endorsed by NRC RG 1.100 (Reference 34) to confirm the adequacy of the pumps to function over the expected range of service conditions specified, including design bases event and post-design bases event conditions, as well as inservice testing (IST) conditions.

b. The loads imposed by the attached piping along with the sustained dynamic and seismic loads are taken into account. The design specification includes applicable loading combinations, and design stress limits for the pumps. In order to assure operability under combined loadings, the stresses resulting from the applied test loads envelope the specified service stress limit for which the pump's operability is intended. Design stress limits applied in evaluating loading combinations are described in Subsection 3.9.3.1.

3.9.3.3.3 Valve Operability

Safety-related active valves are listed in Table 3.9-4. ASME Class 1, 2, and 3 valves are designed/analyzed according to the requirements of ASME Section III, sub-articles NB/NC/ND-3500.

The following criteria are employed in a qualification program to ensure operability of the valves required to function during and following design bases events.

a. Analysis, test, or a combination of test and analysis are used in accordance with ASME QME-1-2007 (Reference 49) as endorsed by NRC RG 1.100 (Reference 34) to confirm the adequacy of the valves to function over the expected range of service conditions specified, including design bases event and post-design bases event conditions, as well as inservice testing (IST) conditions.

b. The loads imposed by the attached piping along with the sustained dynamic and seismic loads are taken into account. The design specification includes applicable loading combinations, and design stress limits for the valves. In order to assure operability under combined loadings, the stresses resulting from the applied test loads envelope the specified service stress limit for which the valve's operability is intended. Design stress limits applied in evaluating loading combinations are described in Subsection 3.9.3.1.

The safety-related valves are subjected to a series of tests prior to service and during the plant life. Prior to installation, the following tests are performed:

a. Shell hydrostatic test to ASME Section III requirements

b. Backseat and main seat leakage tests

c. Disc hydrostatic test

d. Functional tests to verify that valve opens and closes as required when subjected to the design differential pressure and flow.

Cold hydro qualification tests, hot functional qualification tests, periodic in-service inspections, and periodic in-service operational tests are performed in situ to verify and assure the functional ability of the valves. These tests provide reasonable assurance of the reliability of the valve for the design life of the plant.

3.9.3.3.4 Non-NSSS Active ASME Code Class 2 and 3 Pumps and Class 1, 2, and 3 Valves

3.9.3.3.4.1 Pumps

ASME Class 2 and 3 safety-related active pumps are listed in Table 3.9-18. The following criteria are employed in a qualification program to provide reasonable assurance of operability of the pumps required to function during and following design bases events.

a. Analyses, tests, or a combination of analyses and tests are used in accordance with ASME QME-1-2007 (Reference 49) as endorsed by NRC RG 1.100 (Reference 33) to confirm that pumps function adequately over the expected range of service conditions, including design bases events and post-design bases event conditions, as well as inservice inspections and test conditions.

b. The loads imposed by the attached piping along with the sustained dynamic and seismic loads are taken into account. The design specification includes applicable loading combinations, and design stress limits for the pumps. To provide reasonable assurance of operability under combined loadings, the stresses resulting from the applied test loads envelop the specified service stress limit for which pump operability is intended. Design stress limits applied in evaluating loading combinations are described in Subsection 3.9.3.1.3.

3.9.3.3.4.2 Valves

Reasonable assurance of the operability of active valves is provided to perform a safety-related function during and after the specified plant design basis events. The active valves are seismically and functionally qualified in accordance with NRC RG 1.100 (Reference 33), and IEEE Std. 344-2004 (Reference 34), and are described in Subsection 3.9.6. Subsection 3.9.6 also provides a description of the functional design and qualification provisions and IST programs for safety-related valves.

To provide reasonable assurance of the operability of the valves for the plant design life, they are analyzed and/or tested and inspected during construction including the factory tests.

The valves are designed using either stress analysis or conformance to the standard design rules for minimum wall thickness requirements in accordance with the applicable ASME Section III design requirement.

The maximum stress limits used in the analyses are those required by the applicable ASME Section III for the valve that is analyzed. For active valves with extended topworks (e.g., the operator), an analysis of the extended topworks is performed by applying equivalent static seismic loads applied at the center of gravity of the extended topworks.

Prior to installation, functionality and structural integrity tests of the valve are performed as follows:

a. Shell hydrostatic and valve closure tests in accordance with the ASME Section III NB/NC/ND-3530 requirements

b. Functional and operational tests to verify that the valve opens and closes as required when subjected to the design differential pressure and flow

c. Operability qualification of motor operators for seismic and environmental conditions, if any, in accordance with IEEE Std. 323-2003, 344-2004, and 382-2006 (References 51, 34, and 53)

After installation, the following tests are performed to verify and provide reasonable assurance of the functionality of the valve (References 55, 56, 57, and 58). These tests enhance the reliability of the valve for the design life of the plant.

a. Cold hydrostatic tests

b. Hot functional tests

c. Periodic inservice inspections

d. Periodic inservice operations tests

In addition to the valve qualifications noted above, a representative sample of the valves according to type, load level, and size is tested for operability during a simulated plant design basis event of an SSE. The seismic qualification of the valves is performed for the SSE preceded by one or more earthquakes. The number of preceding earthquakes is calculated based on Appendix D of IEEE Std.344-2004 (Reference 34).

Selection of damping values for valves to be qualified is determined in accordance with NRC RG 1.61 (Reference 37) and IEEE Std. 344-2004.

The valve is mounted to represent the typical valve installation or the specified valve installation. The valve includes operators and all appurtenances that are normally attached to the valve in the plant service. Section 3.10 provides the details of seismic qualification.

The operability of the active valves with extended topworks during a Level D service (SSE) condition is verified by satisfying the following criteria:

a. The operability evaluation of the valve is performed by analysis and/or test when subjected to equivalent static load resulting from the accelerations applied at the center of gravity during Level D service conditions.

b. For evaluation by test, the valve is subjected to a static deflection. The valve is operable within the specified operating time limits while an equivalent static load is applied to the extended structure center of gravity in the direction of the weakest axis of the yoke under design pressure.

c. For evaluation by analysis, stresses at critical locations are evaluated and compared to the allowable stress to determine structural integrity, and deflections at critical locations are compared to evaluate operability.

d. Electrical motor operators and other electrical appurtenances necessary for operation are qualified in accordance with IEEE Std. 382-2006 and IEEE Std. 344-2004.

The active valves without extended topworks, such as check valves and other compact valves, are simple in the valve design, and typically there are no masses causing significant

distortions in the affection of the valve operation. Therefore, if these valves are designed so that if structural integrity is maintained, reasonable assurance of valve operability is considered to be provided.

In addition to the above design considerations, the valves are tested and inspected for in-shop hydrostatic test, in-shop seat leakage test, and periodic valve testing and inspection as installed.

Using the methods as described above, the active valves in the piping system are qualified for the valve operability during and after an SSE. In addition, these methods conservatively simulate the seismic event and verify that the active valves will perform their safety-related function when necessary.

3.9.3.4 Component Supports

Jurisdictional boundaries between ASME Section III Class 1, 2, and 3 component supports and the building structure are established in accordance with *[ASME Section III, NF.]**

ASME Section III Class 1, 2, and 3 component supports are designed and constructed in accordance with ASME Section III and ASME Code Case(s).

Seismic Category I component supports are designed to meet the requirements of *[ASME Section III, NF.]** Welding fabrication and installation, nondestructive examination (NDE), and acceptance standards are in accordance with *[ASME Section III, NF.]**

Component support building structures are designed to meet the criteria in Appendix 3.8A.

Component supports that are loaded during normal operation, seismic and following a pipe break (branch line breaks not eliminated by a leak-before-break) are specified for design for loading combinations of Subsection 3.9.3.1. Design stress limits applied in evaluating loading combinations for Level A, B, C or D plant conditions described in Subsection 3.9.3.1 are in accordance with ASME Section III. Loads in compression members are limited to 2/3 of the critical buckling load.

Concrete expansion anchors meet the requirements of *[ACI-349]** and IE Bulletin 79-02 with the provisions identified in Subsection 3.8.4.5.

See Subsection 3.12.6.1 for a discussion of concrete expansion anchors.

Where required, snubber supports are used as shock arrestors for safety-related systems and components. Snubbers are used as structural supports during a dynamic event such as an earthquake or a pipe break but during normal operation act as passive devices that accommodate normal expansions and contractions of the systems without resistance. For the APR1400, snubbers are minimized to the extent practical through the use of design optimization.

Reasonable assurance of snubber operability is provided by incorporating analytical, design, installation, in-service, and verification criteria. The elements used to provide reasonable assurance of snubber operability for the APR1400 include:

a. Consideration of load cycles and travel that each snubber experiences during normal plant operating conditions.

b. Verification that the thermal growth rates of the system do not exceed the required lock-up velocity of the snubber.

c. Accurate characterization of snubber mechanical properties in the structural analysis of the snubber-supported system.

d. For engineered, large bore snubbers, issuance of a design specification to the snubber supplier describing the required structural and mechanical performance of the snubber and verification that the specified design and fabrication requirements are met.

e. Verification that snubbers are properly installed and operable prior to plant operation through visual inspection and measurement of thermal movements of snubber-supported systems during start-up tests.

f. A snubber in-service inspection and testing program, which includes periodic maintenance and visual inspection, inspection following a faulted event, a functional testing program, and repair or replacement of snubbers failing inspection or test acceptance criteria. The inservice testing program for snubbers is described in Subsection 3.9.6.4.

Site-specific information includes a list of all safety-related components that use snubbers in accordance with SRP 3.9.3.

Energy absorbing or non-linear piping restraints may be used on the APR1400. If used, a description of the methodology used to analyze and design the piping systems incorporating these elements is provided with site-specific information.

3.9.4 Control Element Drive Mechanisms

The CEDM is designed, constructed, and tested to meet the requirements of GDC 1, 2, 4, 14, 26, 27, and 29, as well as CFR 50.55a.

3.9.4.1 Descriptive Information of Control Element Drive Mechanism

The control element drive mechanism (CEDM) is a magnetic jack type driving apparatus used to vertically position the control element assemblies (CEAs) as an independent reactivity control system. Each CEDM is capable of withdrawing, inserting, holding, or tripping the CEA from any point within its 3.9 m (153 in) stroke in response to operation signals.

The CEDM is designed to function during and after all plant transients. The CEA drop time for 90 percent insertion is 4.0 seconds maximum. The drop time is defined as the interval between the time the power is removed from the CEDM coils and the time the CEA has reached 90 percent of its fully inserted position. Drop motion begins within 0.5 second after the removal of power. The CEDM is designed to function normally during and after being subjected to seismic loads. The vibratory motion of the SSE is included in the fatigue evaluation in accordance with Subsection 3.9.2.2.3. The CEDM allows for tripping of the CEA during and after an SSE.

The design and construction of the CEDM pressure housing fulfills the requirements of ASME Section III, Subsection NB. The pressure housings are capable of withstanding, throughout the design life, all normal operating loads, including the steady-state and transient operating conditions specified for the vessel. Mechanical excitations are also defined and included as a normal operating load. Design condition of the CEDM pressure housings is 17.2 MPa (2,500 psia) at 343.3 °C (650 °F), and normal operating condition is 15.5 MPa (2,250 psia) at 323.9 °C (615 °F). The loading combinations and stress limit categories are presented in Table 3.9-11 and are consistent with those defined in the ASME Section III.

The design duty requirement for the CEDM is a total cumulative CEA travel of 30,480 m (100,000 ft) operation without loss of function.

The test programs performed in support of the CEDM design are described in Subsection 3.9.4.4.

3.9.4.1.1 Control Element Drive Mechanism Design Description

The CEDMs are mounted on nozzles on the top of the reactor vessel closure head. A CEDM consists of upper pressure housing, motor housing, motor assembly, coil stack assembly, two reed switch position transmitter (RSPT) assemblies, and an extension shaft assembly (ESA). The CEDM is shown in Figure 3.9-7. The drive power is supplied by the coil stack assembly, which is positioned around the motor housing. Two RSPT assemblies are supported by the upper shroud which encloses the upper pressure housing assembly.

The lifting operation consists of a series of magnetically operated step movements. Two sets of mechanical latches are used to engage an ESA. The magnetic force is obtained from the coil stack assembly mounted on the outside of the motor housing.

The CEDM control system actuates the stepping cycle and moves the CEA by a withdrawal or insertion stepping sequence. CEDM-hold is obtained by energizing a latch coil at a reduced current, while all other coils are de-energized. The CEAs are tripped upon interruption of electrical power to all coils. Each CEDM is connected to the CEAs by an ESA.

The axial position of a CEA in the core is indicated by three independent readout systems. One system counts the CEDM steps electronically, and the other two consist of magnetically actuated reed switches located at regular intervals along the upper pressure housing.

3.9.4.1.1.1 Control Element Drive Mechanism Pressure Housing

The CEDM pressure housing consists of the motor housing assembly and the upper pressure housing assembly. The motor housing assembly is attached to the reactor vessel closure head nozzle by means of a threaded joint and seal welding. The upper pressure housing is threaded into the top of the motor housing assembly and seal welded. The upper pressure housing encloses the ESA and contains a vent.

3.9.4.1.1.2 Motor Assembly

The motor assembly is an integral unit that fits into the motor housing and provides the linear motion to the CEA. The motor assembly consists of a latch guide tube, upper latches, and lower latches. Clearances in the motor assembly enable the CEDM to avoid stuck rod condition, which is verified by the tests described in Subsection 3.9.4.4.

Both upper latches and lower latches are used to perform the stepping of the CEA. The upper latch also performs the holding function when CEA motion is not required. Engagement of the ESA occurs when the appropriate set of magnetic coils is energized. Total CEA motion per cycle is 19.1 mm (3/4 in).

3.9.4.1.1.3 Coil Stack Assembly

The coil stack assembly consists of four large DC magnetic coils mounted on the outside of the motor housing assembly. The coils supply magnetic force to actuate magnets for engaging and driving the ESA. Power for the magnetic coils is supplied from two separate supplies.

A conduit assembly containing the lead wires for the coil stack assembly is located at the side of the upper shroud.

3.9.4.1.1.4 RSPT Assembly

Two RSPT assemblies provide separate means for transmitting CEA position indication. Reed switches and voltage divider networks are used to provide two independent output voltages proportional to the CEA position. The RSPT assemblies are positioned so as to use the permanent magnet in the top of the ESA. The permanent magnet actuates the reed switches as it is passed by them.

3.9.4.1.1.5 ESA

The ESAs are used to link the CEDMs to the CEAs. The ESA has a permanent magnet assembly at the top for actuating reed switches in the RSPT assemblies. The center section of the extension shaft assembly is called the drive shaft, and the lower end of it is a coupling device for connection to the CEA.

The drive shaft is threaded and pinned to the extension shaft. The drive shaft has circumferential notches in 19.1 mm (3/4 in) increments along the shaft to provide the means of engagement to the motor assembly.

3.9.4.1.2 Description of the CEDM Motor Operation

Withdrawal or insertion of the CEA is accomplished by programmed electric current to the magnetic coils. There are three programmed conditions for each magnetic coil; high current for initial gap closure, low current for maintaining the gap closed, and zero current to allow opening of the gap.

3.9.4.1.2.1 Operating Sequence for the Double Stepping Mechanism

The initial condition is the hold mode. In this condition, the upper latch coil is energized with low current.

a. Withdrawal

1) The upper lift coil is energized, causing the 11.1 mm (7/16 in) upper lift gap to close lifting the CEA.

2) Low current is supplied to hold the CEA in the withdrawn position.

3) The lower latch coil is energized, causing the lower latches to engage the drive shaft with 0.8 mm (1/32 in) clearance.

4) The upper lift coil is de-energized, allowing the upper latches to drop 19.1 mm (7/16 in) and the drive shaft to lower 0.8 mm (1/32 in) placing the load on the lower latches.

5) The upper latch coil is de-energized, disengaging the upper latches.

6) The lower lift coil is energized, lifting the drive shaft 9.5 mm (3/8 in).

7) The upper latch coil is energized, engaging the upper latches in the drive shaft with 0.8 mm (1/32 in) clearance.

8) The lower lift coil is de-energized, allowing the lower latches to drop 9.5 mm (3/8 in) and causing the drive shaft to drop 0.8 mm (1/32 in), applying the load on the upper latches.

9) The lower latch coil is de-energized, disengaging the lower latches from the drive shaft.

b. Insertion

1) The lower latch coil is energized, causing the lower latches to engage the drive shaft with 8.7 mm (11/32 in) clearance.

2) The lower lift coil is energized, lifting the lower latches 9.5 mm (3/8 in) and lifting the drive shaft 0.8 mm (1/32 in) thus applying the load to the lower latches.

3) The upper latch coil is de-energized, causing the upper latches to disengage the drive shaft.

4) The upper lift coil is energized, moving the de-energized upper latch assembly up 11.1 mm (7/16 in).

5) The upper latch coil is energized, engaging the latches with 8.7 mm (11/32 in) clearance.

6) The lower lift coil is de-energized, allowing the lower latch to drop 9.5 mm (3/8 in). The drive shaft moves down 8.7 mm (11/32 in), stopping on the upper latch assembly, which is energized and in its up position.

7) The lower latch coil is de-energized, disengaging the lower latches.

8) The upper lift coil is de-energized, lowering the upper latch assembly with the drive shaft 11.1 mm (7/16 in).

3.9.4.2 Applicable CEDM Design Specifications

The quality assurance requirements of ASME Section III, Subsection NCA, and ASME NQA are satisfied for design, fabrication, and test of the CEDM. Classification of the CEDM components is provided in Table 3.2-1.

The components forming the pressure boundary are the motor housing assembly, upper pressure housing assembly, vent stem, and housing nut. The pressure boundary components are designed, constructed, and tested in accordance with ASME Section III, Subsection NB, to provide reasonable assurance of extremely low probability of leakage or gross rupture. The material of the CEDM is described in Subsection 4.5.1.

The adequacy of the design of the non-pressure boundary components has been verified by the life tests as described in Subsection 3.9.4.4.

The RSPT assembly is designed to comply with IEEE Std. 323-2003 (Reference 51) and IEEE Std. 344-2004 (Reference 34). The electrical components are external to the pressure boundary and are non-pressurized.

3.9.4.3 Design Loads, Stress Limits and Allowable Deformations

The CEDM stress analyses consider the following loads:

a. Reactor coolant pressure and temperature

b. Reactor operating transient conditions

c. Normal operating loads

   1) Dead weight

   2) Impulse load due to stepping of the CEDM

   3) Mechanical base excitation loads

   4) Loads produced by the thermal expansion of the reactor vessel closure head

d. IRWST loads

e. BLPB loads

f. Seismic loads

The design and fabrication of the CEDM pressure boundary components fulfill the requirements of ASME Section III, Subsection NB. The pressure housings are capable of withstanding all the steady-state and transient operating conditions specified in Table 3.9-11 for a 60-year life. The design report for the ASME Code Class 1 components is to be prepared in accordance with ASME Section III.

Deformation of the CEDM under seismic conditions is evaluated to verify scramability as presented in Subsection 3.9.2.7.3.

The adequacy of the design of the CEDM pressure boundary and non-pressure boundary components has been verified by the life cycle tests as described in Subsection 3.9.4.4.

3.9.4.4 CEDM Operability Assurance Program

The APR1400 CEDM is essentially identical to the System 80 CEDM, which is presently operating at the Palo Verde Nuclear Generating Station, except for some minor changes of non-nuclear safety related components. The following describes the tests performed during development of System 80 CEDM, which provides design verification for the APR1400 CEDM.

For the life cycle test, the CEDM was installed on a test facility that was operated at a nominal temperature of 315.6 °C (600 °F) and a gauge pressure of 15.5 MPa (2,250 psig). The CEDM was operated for total travel length of 47,854 m (157,000 ft) with no abnormality, which is about 1.5 times the design duty requirement.

During the CEA scram test, 300 full-height drops were completed. All release times were less than 0.3 second, and CEA drop times to 90 percent of full insertion were less than 4.0 seconds, which meets the design criterion.

Operating experience also provides design verification of the APR1400 CEDM. The APR1400 CEDM is essentially identical to the CEDM of Palo Verde, YGN 3&4, YGN 5&6, UCN 3&4, UCN 5&6, SKN 1&2, and SWN 1, which are all in operation. The experience has demonstrated that the CEDM operates without malfunction.

First production test programs were completed on the CEDM to verify operability. During the course of this program, more than 1,219 m (4,000 ft) of travel was accumulated and 30 full-height gravity drops were made without mechanism malfunction or measurable wear on operating parts. The program included the following:

a. Operation at 76.2 cm/min (30 in/min) traveling 159 kg (350 lb) of weight at ambient temperature and a gauge pressure of 0.7 MPa (100 psig) for 15.2 m (50 ft)

b. Fifteen full-height drops at simulated reactor operating conditions with 159 kg (350 lb) of weight during the first 61 m (200 ft) travel at 76.2 cm/min (30 in/min)

c. Fifteen full-height drops at simulated reactor operating conditions with 159 kg (350 lb) of weight after traveling 1,162 m (3,812 ft)

d. Operation at simulated reactor operating conditions traveling 159 kg (350 lb) of weight for over 1,219 m (4,000 ft)

e. Operation at 76.2 cm/min (30 in/min) traveling 159 kg (350 lb) at ambient temperature and a gauge pressure of 0.7 MPa (100 psig) for 15.5 m (51 ft)

The mechanism operated without malfunction throughout the test program and, upon final inspection, no measurable wear was found.

Production testing is performed prior to shipment to confirm the capability of the CEDM to meet operation requirements. All production CEDM motors are tested at both ambient pressure and fluid temperatures and at simulated reactor operating conditions for a minimum of 122 m (400 ft) of travel and six full-height gravity drop tests at simulated reactor operating conditions.

No malfunction such as mis-stepping is allowed during the tests, and the total drop time is required to be less than 4.0 seconds with a release time of less than 0.5 second.

After installation of the CEDMs, initial startup testing is performed to verify the insertion, withdrawal, and drop time. The initial startup test program is described in Section 14.2.

3.9.5 Reactor Pressure Vessel Internals

Reactor pressure vessel internals described as reactor internals in this subsection refer to the core support structures and internal structures. Core support structures are those structures or parts of structures which are designed to provide direct support or restraint of the core within the reactor vessel. Internal structures are all structures within the pressure vessel other than core support structures, fuels, control element assemblies, and instrumentations.

3.9.5.1 Design Arrangements

The components of the reactor internals are divided into two major parts consisting of the core support barrel assembly and the upper guide structure assembly. The flow skirt, although functioning as an integral part of the coolant flow path, is separate from the reactor internals and is affixed to the bottom head of the reactor vessel. The arrangement

of these components is shown in Figure 3.9-8. The flow paths of the main coolant flow and bypass flow within the reactor vessel are described in Subsection 4.4.2.6.1.

The component classification for core support structures and internal structures as reactor internals is summarized below:

a. Core support structures:

1) Core support barrel

2) Lower support structure

3) Upper guide structure barrel assembly

b. Internal structures:

1) Core shroud

2) Alignment keys

3) Hold-down ring

4) Core support barrel snubbers

5) In-core instrumentation nozzle assembly

6) Core support barrel outlet nozzles

7) Inner barrel assembly

8) Guide lugs

9) Heated junction thermocouple tube assembly

The design arrangements including functional requirements for each component of the reactor internals are described in detail below.

3.9.5.1.1 Core Support Barrel Assembly

The major structural member of the reactor internals is the core support barrel assembly. The core support barrel assembly consists of the core support barrel, the lower support structure and in-core instrumentation nozzle assembly, and the core shroud. The material for the assembly is austenitic stainless steel.

The core support barrel assembly is supported at its upper end by the upper flange of the core support barrel, which rests on a ledge in the reactor vessel. Alignment is accomplished by means of four equally spaced keys in the flange, which fit into the keyways in the reactor vessel ledge and closure head. The lower flange of the core support barrel supports, secures, and positions the lower support structure and is attached to the lower support structure by means of a welded flexural connection. The lower support structure provides support for the core by means of support beams that transmit the load to the core support barrel lower flange. The locating pins in the beams provide orientation for the lower ends of the fuel assemblies. The core shroud provides a flow path for the coolant and limits the amount of coolant bypass flow. Support and positioning for the fuel assemblies are provided by the lower support structure. The lower end of the core support barrel is restricted from excessive lateral and torsional movement by six snubbers that interface with the reactor vessel wall. The core support barrel assembly is shown in Figure 3.9-9.

3.9.5.1.1.1 Core Support Barrel

The core support barrel is a right circular cylinder including a heavy external ring flange at the top end and an internal ring flange at the lower end. The core support barrel is supported from a ledge on the reactor vessel. The core support barrel supports the lower support structure upon which the fuel assemblies rest. Shrunk-fit into the flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head, and upper guide structure assembly flange are slotted in locations corresponding to the alignment key locations to provide alignment between these components in the reactor vessel flange region.

The upper section of the core support barrel contains two outlet nozzles that interface with internal projections on the reactor vessel outlet nozzles to minimize leakage of coolant from inlet to outlet. Since the weight of the core support barrel is supported at its upper end, it is possible that coolant flow could induce vibrations in the structure. Therefore, amplitude limiting devices, or snubbers, are installed on the outside of the core support barrel near the bottom end. The snubbers consist of six equally spaced lugs around the circumference of the core support barrel and act as a tongue-and-groove assembly with the mating lugs on the reactor vessel. Minimizing the clearance between the tongue-and-groove assembly limits the amplitude of vibration. During assembly, as the reactor internals are lowered into the reactor vessel, the reactor vessel lugs engage the core support barrel lugs in an axial direction. Radial and axial expansions of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted. The reactor vessel lugs have bolted and captured nickel-based alloy X-750 shims. The core support barrel lug mating surfaces are hardfaced with Stellite to minimize wear. The reactor vessel shims are machined during initial installation to provide minimum clearance. The snubber assembly is shown in Figure 3.9-10.

3.9.5.1.1.2 Lower Support Structure and ICI Nozzle Assembly

The lower support structure and in-core instrumentation (ICI) nozzle assembly position and support the fuel assemblies, core shroud, and ICI nozzles. The structure is a welded assembly consisting of a short cylinder, support beams, a bottom plate, ICI nozzles, and ICI nozzle support plate. The lower support structure is made up of a short cylindrical section enclosing an assemblage of grid beams arranged in egg-crate fashion. The outer ends of these beams are welded to the cylinder. Fuel assembly locating pins are attached to the top of the beams. The bottoms of the main support beams in one direction are welded to an array of plates which contain flow holes to provide proper flow distribution. These plates also provide support for the ICI nozzles, support columns, and ICI nozzle support plate. The cylinder guides the main coolant flow and limits the core shroud bypass flow by means of holes located near the base of the cylinder. The ICI nozzle support plate provides lateral support for the ICI nozzles. This plate is provided with flow holes for the requisite flow distribution. The lower support structure and ICI nozzle assembly are shown in Figure 3.9-11.

3.9.5.1.1.3 Core Shroud

The core shroud provides an envelope for the core and limits the amount of coolant bypass flow. The core shroud consists of a welded vertical assembly of plates designed to channel the coolant through the core. Circumferential rings and top and bottom end plates provide lateral support. The rings are attached to the vertical plates by means of full length welded ribs and horizontal braces. A small gap is provided between the core shroud outer perimeter and the core support barrel in order to provide upward coolant flow in the annulus, thereby minimizing thermal stresses in the core shroud. The core shroud is shown in Figure 3.9-12. Four hard-faced guide lugs, spaced 90 degrees apart, protrude vertically from the top of the core shroud and engage in corresponding hard-faced slots in the upper guide structure fuel alignment plate to provide reasonable assurance of proper alignment between the upper guide structure assembly and core shroud/lower support structure.

3.9.5.1.2 Upper Guide Structure Assembly

The upper guide structure (UGS) assembly aligns and laterally supports the upper end of the fuel assemblies, maintains the control element spacing, holds down the fuel assemblies during operation, prevents fuel assemblies from being lifted out of position during a severe accident condition and protects the control elements from the effects of coolant cross flow in the upper plenum. The UGS assembly is handled as one unit during installation and refueling.

The UGS assembly consists of the UGS barrel assembly and the inner barrel assembly (IBA) (Figure 3.9-13). The UGS barrel assembly consists of UGS support barrel, fuel alignment plate, UGS support plate and control element guide tubes. The UGS support barrel consists of a right circular cylinder welded to a ring flange at the upper end and to a circular plate (UGS support plate) at the lower end. The flange, which is the supporting member for the entire UGS assembly, seats on its upper side against the reactor vessel head during operation. The lower side of the flange is supported by the hold-down ring, which rests on the core support barrel upper flange. The UGS flange and the hold-down ring engage the core support barrel alignment keys by means of four accurately machined and located keyways equally spaced at 90-degree intervals. This system of keys and slots provides an accurate means of aligning the core with the closure head and thereby with the control element drive mechanisms. The fuel alignment plate is positioned below the UGS

support plate by cylindrical control element guide tubes. These tubes are attached to the UGS support plate and the fuel alignment plate by rolling the tubes into the holes in the plates and welding. The fuel alignment plate is designed to align the lower ends of the control element guide tubes which in turn locate the upper ends of the fuel assemblies. The fuel alignment plate also has four equally spaced slots on its outer edge that engage with Stellite hard-faced lugs protruding from the core shroud to provide alignment. The control element guide tubes bear the upward force on the fuel assembly hold-down devices. This force is transmitted from the fuel alignment plate through the control element guide tubes to the UGS barrel support plate.

The IBA limits crossflow and provides separation of the CEA. The IBA consists of top plate welded to a right circular barrel open at the bottom and containing an assemblage of large vertical tubes connected by vertical plates in a grid pattern welded to the inside of the barrel. The IBA is held in position by continuous weld between the barrel flange and the top surface of the UGS barrel upper flange.

Guides for the CEA extension shafts are provided by the top plate of the IBA. The tubes and connecting plates within IBA are furnished with multiple holes to permit hydraulic communication.

The hold-down ring provides axial force on the flanges of the UGS assembly and the core support barrel assembly in order to prevent movement of the structures under hydraulic forces. The hold-down ring is designed to accommodate the differential thermal expansion between the reactor vessel and the reactor internals in the vessel ledge region.

3.9.5.1.3 Flow Skirt

The nickel-based alloy flow skirt is a right circular cylinder, perforated with flow holes, and reinforced with two stiffening rings. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The flow skirt is supported by nine equally spaced machined sections that are welded to the bottom head of the reactor vessel.

3.9.5.1.4 In-Core Instrumentation Support System

The in-core neutron flux monitoring system includes self-powered, in-core detector assemblies, supporting structures and guide paths and an amplifier system to process detector signals. The self-powered in-core detector assemblies and the amplifier system are described in Section 7.7. The instrumentation supporting structures and guide paths are described in this section and shown in Figure 3.9-14.

The support system begins outside the reactor vessel, penetrates the bottom of the vessel boundary and terminates in the upper end of the fuel assembly. Each in-core instrument is guided over the full length by the external guidance conduit, ICI guide tube nozzles of the reactor vessel, the lower support structure, and the guidepost of the fuel assembly. Figure 3.9-14 shows the in-core instrument support structure. The in-core instrumentation support system routes the instruments so that detectors are located in selected fuel assemblies throughout the core. An equal instrument has the same length for all locations. The guide tube routing outside the reactor vessel is a simple 180-degree bend to the seal table. The pressure boundaries for the individual instruments are at the out-of-reactor seal table, where the external electrical connections to the in-core instruments are made. Each instrument has an integral seal plug which forms a seal at the instrument seal table and through which the signal cables pass. Static O-ring seals are used to seal against operating pressure.

3.9.5.2 Loading Conditions

The following loading conditions are considered in the design of the reactor internals:

a. Normal operating temperature differences

b. Normal operating pressure differences

c. Flow loads

d. Weights, reactions, and superimposed loads

e. Vibration loads

f. Shock loads (including SSEs)

g. Anticipated transient loadings

h. Handling loads (not combined with other loads above)

i. Appropriate design-basis pipe break (DBPB), secondary side break, and LOCA loads

j. IRWST discharge loads

3.9.5.2.1 Design Loadings

The following loading combination is considered as design loadings.

Normal operation loads in combination with IRWST discharge loads. Normal operation loads are defined as the following sustained loads resulting from the normal events:

a. Pressure difference

b. Temperature

c. Mechanical loads

    1) Weight

    2) Loads from flow impingement or flow of reactor coolant

    3) Superimposed or reaction loads

3.9.5.2.2 Level A Service Loadings

The following loading combination is considered as Level A service loadings.

Normal operation loads in combination with specified system operating transient loads resulting from normal events.

3.9.5.2.3 Level B Service Loadings

The following separate loading combination is considered as Level B service loadings.

a. Normal operation loads in combination with IRWST discharge loads and system operating transient loads from the upset events. The IRWST discharge loads are defined as the loads due to postulated discharge to in-containment refueling water storage tank.

b. Normal operation loads in combination with the system operating transient loads from the upset event (the loss of external load with turbine control system failure). Note that the loss of external load of the upset event, which is evaluated as if it occurs once during the plant lifetime, is the emergency event. This event is evaluated with this combination of loadings for conservatism.

3.9.5.2.4 Level C Service Loadings

Level C service loadings are derived from the combination of normal operation loads and the DBPB loads.

The DBPB is defined as a postulated pipe break that results in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system.

3.9.5.2.5 Level D Service Loadings

The following loading combination is considered as Level D service loadings.

a. Normal operation loads

b. Either the main steam/feed water pipe break (MS/FWPB) or LOCA loads (including asymmetric blowdown loads), whichever are greater

c. SSE loads

d. IRWST discharge loads

LOCA is defined as the loss of reactor coolant at a rate in excess of the reactor coolant normal makeup rate, from breaks in the reactor coolant pressure boundary inside primary containment up to, and including, a break equivalent in size to the largest primary branch line not eliminated by leak-before-break (LBB) criteria.

3.9.5.3 Design Bases for Reactor Internals

The RCS transient design basis for reactor internals is addressed in Subsection 3.9.1.1.

The potential adverse flow effects of flow-induced vibration (FIV) and acoustic resonances on reactor internals are addressed in Subsection 3.9.2.3. The CVAP for reactor internals is addressed in Subsection 3.9.2.4. The dynamic system analysis of reactor internals under faulted conditions is addressed in Subsection 3.9.2.5.

The reactor internals are designed to meet interface cold gaps between reactor internals and the reactor vessel and between the main parts of the reactor internals.

The reactor internals are designed, fabricated, erected, and tested to comply with the requirements of 10 CFR 50.55a, 10 CFR 52.47(b)(1), 10 CFR 52.80(a), and 10 CFR 50, Appendix A (GDC 1, 2, 4, and 10).

The design code, code cases, and acceptance criteria applicable to the design, analysis, fabrication, and non-destructive examination are provided in the design specification to meet the structural and functional integrity of the reactor internals in accordance with ASME Section III. The design specification and design report for core support structures are prepared in accordance with the NCA requirements of ASME Section III.

The stress limits to which the reactor internals are designed are listed in Table 3.9-12.

The loading categories and stress limits are defined in the applicable section of ASME Section III. The design and construction of core support structures and internal structures are in accordance with ASME Section III as described in Table 3.9-12.

To properly perform their functions, the reactor internals are designed to meet the following deformation limits:

a. Under Level A, Level B, and Level C service loadings, the core is held in place, and deflections are limited so that the CEAs can be inserted under their own weight as the only driving force.

b. Under Level D service loadings that require CEA insertability, deflections are limited so that the core is held in place, adequate core cooling is preserved, and all CEAs can be inserted. Those deflections that would influence CEA movement are limited to less than 80 percent of the deflections required to prevent CEA insertion.

   The allowable deformation limits are established as 80 percent of the loss-of-function deflection limits.

   The significant component deflection limits are designed as follows:

   1) Fuel lower end fitting interface with the lower support structure is deflection limited to avoid disengagement.

   2) Fuel upper end fitting interface with the upper guide structure relative displacement precludes disengagement.

   3) The CEA guide tube lateral deflection allows CEA insertion.

In the design of critical reactor internals that are subject to fatigue, stress analysis is performed using the design fatigue curve of Figure I-9.2 of ASME Section III. A cumulative usage factor of less than one is used as the limiting criterion.

As indicated in the preceding sections, the stress and fatigue limits for reactor internals are obtained from ASME Code. Allowable deformation limits are established as 80 percent of the loss-of-function deflection limits. These limits provide adequate safety factors providing reasonable assurance that as long as calculated stresses, cumulative usage factors, or deformations do not exceed these limits, the design is conservative.

The COL applicant is to provide a summary of the maximum calculated total stress, deformation, and cumulative usage factor for each service limit of core support structures in accordance with ASME Section III (COL 3.9(4)).

3.9.6 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

3.9.6.1 Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints

Plant-specific information includes the design and qualification requirements and acceptance criteria for ASME OM Code (Reference 32) and applicable addenda. The functional design and qualification of safety-related pumps, valves, and dynamic restraints are performed in accordance with ASME QME-1 (Reference 49) as endorsed by NRC RG 1.100 (Reference 33).

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

For the safety-related valves, plant-specific information includes the following design and qualification requirements and acceptance criteria for these requirements. By testing each size, type, and model, the force requirements to operate the power-operated valves (POV) is determined to provide reasonable assurance of the adequacy of the force that the operator can deliver under design conditions. For the safety-related power operated valves, each size, type, and model are tested 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 pressure, spring force, minimum voltage, and minimum and maximum stroke time requirements. This testing of each size, type, and model includes test data from the manufacturer, field test data for plant-specific dedication, empirical data supported by test, or tests of similar valves that support the qualification of the required valve where similarity is justified by technical data. This preoperational testing demonstrates that the results of the testing under in-situ or installed conditions can be used to provide reasonable assurance of the capability of the valves to operate under design conditions. Test data are used to provide reasonable assurance that the structural capability limits of the individual parts of the valves are not exceeded under design conditions.

Test data are used to provide reasonable assurance of the 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 (vertical versus horizontal) as recommended by the manufacturer. Valve design features, material, and surface finish provide reasonable assurance that the non-intrusive diagnostic testing methods available in the industry or as specified can be accommodated. Flow through the valve is determinable from installed instrumentation, and valve disk positions are determinable without disassembly, such as by the use of non-intrusive diagnostic methods. Valve internal parts are designed with self-aligning features for correct installation.

3.9.6.2 Inservice Testing Program for Pumps

Inservice testing (IST) for safety-related pumps is developed in accordance with the requirements of ASME OM Code, ISTA and ISTB. Pumps subject to IST in accordance with the ASME Code are listed in Table 3.9-13. This table includes the safety class, test parameters, and the frequency at which the testing is performed.

This program includes baseline pre-service testing to support the periodic in-service testing of the safety-related pumps. Depending on the test results, the plan will provide commitment to disassemble and inspect the safety-related pumps.

For each size, type, and model of pump, testing that encompasses design conditions is performed to demonstrate acceptable flow rate and corresponding pump head, bearing

vibration levels, and pump internals wear rates for the operating time specified for each system mode of pump operation. From these tests, baseline hydraulic and vibration data for evaluating the acceptability of the pump after installation are also developed. Test data are used to provide reasonable assurance that the pump specified for each application is not susceptible to inadequate minimum flow rate and inadequate thrust bearing capacity. With respect to minimum pump flow operation, the sizing of each minimum recirculation flow path is evaluated to provide reasonable assurance that its use under all analyzed conditions does not result in degradation of the pump. The flow rate through minimum recirculation flow paths can be measured periodically to verify that flow is in accordance with the design specification.

The 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 net positive suction head available (NPSHA) is greater than or equal to the net positive suction head required (NPSHR) during all modes of pump operation. These pumps can be disassembled for evaluation when ASME OM Code, ISTB testing results in a deviation which falls within the required action range. The code provides criteria limits for the test parameters identified in Table 3.9-13. The detailed IST program establishes 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. Factors to be considered in the frequency and extent of disassembly include, but are not limited to the following:

a. Historical performance of the pump to identify pumps that are prone to degradation/wear.

b. Analysis of trends of pump test parameters and service conditions.

c. Analysis of pump components that are subject to aging and require a maintenance replacement approach (e.g., O-rings).

d. Results of non-intrusive pump testing. The non-intrusive technologies used may obviate the need for inspection/disassembly of safety-related pumps, provided the technologies demonstrate an equivalent ability to detect pump degradation as inspection/disassembly would.

The testing requirements and acceptance criteria are identified in ISTB-5000.

The program may be revised throughout the plant life to minimize disassembly based on disassembly experience.

The COL applicant is to provide type of testing and frequency of site-specific pumps subject to IST in accordance with the ASME Code (COL 3.9(5)).

3.9.6.3 Inservice Testing Program for Valves

The IST of safety-related valves is addressed in plant-specific information.

IST for safety-related valves is developed in accordance with the requirements of ASME OM Code ISTA and ISTC. Table 3.9-13 lists the valves to be included in the IST program as well as the valve type, valve identification number, code class, valve category, valve functions, required tests, and test frequencies. Safety-related valves include the vales that are necessary to provide reasonable assurance of the following:

a. Integrity of the RCPB

b. Capability to achieve safe shutdown of the reactor and keep it in a safe shutdown condition

c. Cpability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures in excess of 10 CFR 50.34 guidelines

Table 3.9-13 also provides explanatory notes/justifications for any code defined testing exceptions. Plant-specific information includes safety-related valve IST details, including test schedules and frequencies, in the inspection and testing program. This program includes baseline pre-service testing to support the periodic in-service testing of the safety-related valves. Depending on the test results, the plan will provide commitment to disassemble and inspect the safety-related valves. The primary elements of this plan, including the requirements of GL 89-10 for motor-operated valves (MOVs), are promulgated in Subsection 3.9.6.3.1.

Inservice inspection (ISI) is described in Subsection 5.2.5 and Section 6.6.

The specific testing requirements and acceptance criteria are identified in ISTC-5000.

The COL applicant is to provide the type of testing and frequency of site-specific valves subject to IST in accordance with the ASME Code (COL 3.9(6)).

3.9.6.3.1 Inservice Testing Program for Motor-Operated Valves

IST of ASME Section III Classes 1, 2, and 3, and safety-related MOVs is performed in accordance with ASME OM Code with applicable addenda and GL 89-10 (Reference 57), as required by 10 CFR 50.55a(f).

The IST of MOVs relies on diagnostic techniques that are consistent with the state-of-the-art and that permit an assessment of the performance of the valve under actual loading. Periodic testing per GL 89-10 Paragraphs D and J is conducted under adequate differential pressure and flow conditions that allow a justifiable demonstration of continuing MOV capability for design basis conditions. The detailed IST program includes the optimal frequency of this periodic verification. The frequency and test conditions are sufficient to demonstrate continuing design basis and required operating capability. The code provides criteria limits for the test parameters identified in Table 3.9-13 for the ASME Code IST.

Each MOV is tested in the open and closed states under static and maximum achievable preoperational conditions using diagnostic equipment that measures torque and thrust, and motor parameters. The MOV is tested under various differential pressure and flow conditions up to maximum achievable conditions to determine torque and thrust requirements at design conditions. The parameters and acceptance criteria, which demonstrate fulfillment of the functional performance requirements, are as follows:

a. As required by the safety function, the valve is fully open or fully closed with diagnostic indication of hard-seat contact.

b. The control switch settings provide adequate margin to achieve design requirements including consideration of diagnostic equipment inaccuracies, control switch repeatability, load sensitive behavior, and margin for degradation.

c. The motor output capability at degraded voltage is equal to or exceeds the control switch setting including consideration of diagnostic equipment inaccuracies, control switch repeatability, load sensitive behavior, and margin for degradation.

d. The maximum torque and thrust achieved by the MOV, including diagnostic equipment inaccuracies and control switch repeatability, do not exceed the allowable structural capability limits for the individual parts of the MOV.

e. The remote position indication testing verifies that proper disk position is indicated in the control room.

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

The detailed IST program establishes 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 extent of each disassembly. Factors to be considered in the frequency and extent of disassembly include, but are not limited to the following:

a. Historical performance of the safety-related valves to identify valves which are prone to degradation/wear.

b. Analysis of trends of valve test parameters and service condition.

c. Analysis of valve components that are subject to aging and require a maintenance replacement approach (e.g., O-rings).

d. Results of non-intrusive valve testing. Use of non-intrusive technologies may obviate the need for inspection/disassembly of safety-related valves altogether, provided the technologies demonstrate an equivalent ability to detect valve degradation as inspection/disassembly would.

The program may be revised throughout plant life to minimize disassembly based on past disassembly experience.

3.9.6.3.2 Inservice Testing Program for Power-operated Valves Other than Motor-operated Valves

All safety-related piping systems incorporate provisions for testing to demonstrate the operability of the POVs under design conditions. Periodic testing is conducted under adequate differential pressure and flow conditions per the guidance of Regulatory Issue Summary 2000-03, which incorporates the lessons learned from MOV analyses and tests in response to GL 89-10 and the Joint Owners Group (JOG) air-operated valve program (Reference 59). Periodic testing allows a justifiable demonstration of continuing POV capability for design basis conditions. The detailed IST program includes the optimal frequency of this periodic verification. The frequency and test conditions are sufficient to demonstrate continuing design basis and required operating capability. The in-service testing of POVs incorporates the use of advanced non-intrusive techniques to periodically assess degradation and the performance characteristics of the POVs. Solenoid-operated valves (SOVs) are tested using Class 1E electrical power supply voltage and current to verify SOVs are capable of performing their safety functions at design-basis accident conditions. SOV tests include confirmation of the energized position and fail position when de-energized. The ASME OM Code, Subsection ISTC tests are performed, and valves that fail to exhibit the required performance can be disassembled for evaluation.

The ASME OM Code, Subsection ISTC, provides criteria limits for the test parameters identified in Table 3.9-13.

The detailed IST program establishes 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. Factors to be considered in the frequency and extent of disassembly include but are not limited to the following:

a. Historical performance of the safety-related valves to identify valves that are prone to degradation/wear.

b. Analysis of trends of valve test parameters and service conditions.

c. Analysis of valve components that are subject to aging and require a maintenance replacement approach (e.g., O-rings).

d. Results of non-intrusive valve testing. Use of non-intrusive technologies may obviate the need for inspection/disassembly of safety-related valves altogether, provided the technologies demonstrate an equivalent ability to detect valve degradation as inspection/ disassembly would.

The program may be revised throughout plant life to minimize disassembly based on past disassembly experience.

3.9.6.3.3 Inservice Testing Program for Check Valves

All safety-related piping systems incorporate provisions for testing to demonstrate the operability of check valves under design conditions. The IST of check valves incorporates the use of advanced non-intrusive techniques to periodically assess degradation and the performance characteristics of the check valves. The effects of rapid pump starts and stops, as expected for system operating conditions, are considered in the testing. Conditions of any reverse flow that may occur during expected system operation are also considered in the testing. Non-intrusive technique includes acoustic, ultrasonic, magnetic, and x-ray technologies, which are used to measure valve operating parameters (e.g., fluid flow, disk position, disk movement, disk impact forces). The technique is also used for monitoring an upstream pressure or tank level, and for performing a leak test, a system hydrostatic or pressure test, or radiography.

The ASME OM Code, ISTC, tests are performed, and check valves that fail to exhibit the required performance can be disassembled for evaluation. The ASME OM Code, ISTC, provides criteria limits for the test parameters identified in Table 3.9-13.

The detailed IST program includes the frequency and 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. Factors to be considered in the disassembly frequency and extent of disassembly include but are not limited to the following:

a. Historical performance of the safety-related check valves to identify valves that are prone to degradation/wear.

b. Analysis of trends of valve test parameters and service conditions.

c. Analysis of valve components that are subject to aging and require a maintenance replacement approach.

d. Results of non-intrusive valve testing. Use of non-intrusive technologies may obviate the need for inspection/disassembly of safety-related check valves altogether, provided the technologies demonstrate an equivalent ability to detect check valve degradation as inspection/disassembly would.

The program may be revised throughout plant life to minimize disassembly based on past disassembly experience.

3.9.6.3.4 Pressure Isolation Valve Leak Testing

The leak-tight integrity is verified for each valve relied on to provide a leak-tight function. These valves include:

a. Pressure-isolation valves (PIVs) that provide isolation of a pressure differential from one part of a system to another part or between systems. PIVs associated with the RCS are defined in GL 89-04, Attachment 1, Section 4a. The RCS PIVs are listed in Table 3.9-14 and are tested in accordance with Table 3.9-13 and Technical Specifications surveillance requirement 3.4.13.1.

b. Temperature-isolation valves (TIVs) whose leakage may cause unacceptable thermal stress, fatigue, or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps. Safety-related valves performing this duty are listed in Table 3.9-13, along with a description of specific leakage test requirements.

3.9.6.3.5 Containment Isolation Valve Leak Testing

The leak-tight integrity is verified for each valve relied on to provide a leak-tight function. These valves include containment-isolation valves (CIVs) that provide isolation capability for the piping systems penetrating containment.

CIVs are leak tested in accordance with 10 CFR 50, Appendix J. CIVs are listed along with their required testing in Table 6.2.4-1. Those CIVs for which a Type-C leakage rate test is specified in Table 6.2.4-1 are also tested in accordance with ASME OM Code, Subsection ISTC. These CIVs are designated in Table 3.9-13 by the valve function CIC. Those CIVs for which a Type-C leakage rate test is not specified in Table 6.2.4-1 are designated in Table 3.9-13 by the valve function CIN. The CIN valve function designation indicates that these valves are listed in Table 6.2.4-1, but are not leakage rate tested in Tables 6.2.4-1 and 3.9-13.

3.9.6.3.6 Inservice Testing Program for Safety and Relief Valves

Pressure-relief devices are tested in accordance with ASME OM Code, including Appendix I to the OM Code, for IST.

Stroke tests are performed for dual-function safety and relief valves. Power-operated relief valves subject to the IST program are tested in accordance with Subsection ISTC-5100 for Category B valves, and Subsection ISTC-5240 for Category C valves. The test equipment, including gages, transducers, load cells, and calibration standards, used to determine valve set-pressure is acceptable if the overall combined accuracy does not exceed ±1 percent of the indicated (measured) set pressure.

A list of safety and relief valves included in the IST program is provided in Table 3.9-13.

3.9.6.3.7 Inservice Testing Program for Manually Operated Valves

Safety-related active manually operated valves are identified in the IST Program Plan, and exercised periodically in accordance with frequency and requirements specified in the ASME OM Code.

Manual valves are exercised at least every 2 years. Exercise of a manual valve includes a complete cycle from open to fully closed.

A list of manual valves included in the IST program is provided in Table 3.9-13.

3.9.6.3.8 Inservice Testing Program for Explosively Activated Valves

The in-service testing program for explosively activated valves is not applicable to the APR1400 design.

3.9.6.4 Inservice Testing Program for Dynamic Restraints

Safety-related systems inside and outside of containment may experience dynamic effects under various accident conditions, including seismic events and DBAs. Snubbers are attached to these systems to reduce these dynamic effects in areas where rigid supports are unacceptable. The snubber is selected to satisfy the system design requirements. The snubber design and operating information form the basis for snubber examination and testing requirements.

As described in Subsection 3.12.6.6, dynamic restraints within piping systems is to be minimized as-much-as-possible due to the maintenance and testing requirements for these components. However, dynamic restraints in the form of snubber supports are used where free thermal movements are required and restraining movements caused by dynamic loadings are also required. Snubber operability inspections and tests including scope and frequency requirements are specified and controlled in the components support inspection and testing program plan. The ASME OM Code, 2004 Edition with 2005 and 2006 Addenda, provides ISI methods and requirements for examinations and tests of snubbers at nuclear power plants.

Preservice and in-service examinations are performed using the VT-3 visual examination method described in IWA-2213 of the ASME Section XI, 2007 edition with 2008 addenda. Snubbers are visually examined to identify impaired function caused by physical damage, leakage, corrosion, or degradation from environmental exposure or operating conditions. External features that may affect operability are also examined.

Preservice functional testing is performed on snubbers prior to initial plant operation. This testing may be performed at the manufacturer’s facility. Inservice functional testing is performed over the test plan intervals specified in ASME OM Code Subsection ISTD. Snubbers are tested in their installed location or removed and bench tested. Snubbers are tested in their as-found condition and the test parameters are selected so that the snubbers are tested to the fullest extent practicable.

The APR1400 snubber design incorporates accessibility provisions for maintenance, inspection, and testing of components. The correct installation and operation of snubbers is confirmed as part of the ITP described in Section 14.2. This program includes visual inspections, hot and cold position measurements, and documenting thermally induced component movement that occurs during plant startup.

The COL applicant is to provide a table listing all safety-related components that use snubbers in their support systems including the following information (COL 3.9(7):

a. Identification of the systems and components that use snubbers

b. The number of snubbers used in each system and on the components in that system

c. Identification of the type(s) of snubber (hydraulic or mechanical)

d. Specification whether the snubber was constructed in accordance with the *[ASME Section III, Subsection NF]**

e. A statement of whether the snubber is used as a shock, vibration, or dual-purpose snubber

f. If a snubber is identified as either a dual-purpose or vibration arrester type, indication of whether the snubber and/or component was evaluated for fatigue strength

3.9.6.5 Relief Requests and Alternative Authorizations to ASME OM Code

In case implementing the requirements of ASME OM Code is impractical, the relief request will be made on a case-by-case basis. Information provided will describe the specific area of relief requested, explain why compliance with ASME OM Code is impractical, and describe any alternative test pursuant to 10 CFR 50.55a.

3.9.7 [Reserved]

3.9.8 [Reserved]

3.9.9 Combined License Information

COL 3.9(1) The COL applicant is to provide the inspection results for the APR1400 reactor internals classified as non-prototype Category I in accordance with NRC RG 1.20.

COL 3.9(2) The COL applicant is to provide a summary of the maximum total stress, deformation, and cumulative usage factor values for each of the component operating conditions for ASME Code Class 1 components. For those values that differ from the allowable limits by less than 10 percent, the COL applicant is to provide the contribution of each of the loading categories (e.g., seismic, dead weight, pressure, thermal) to the total stress for each maximum stress value identified in this range.

The COL applicant is to also provide a summary of the maximum total stress and deformation values for each of the component operating conditions for Class 2 and 3 components required to shut down the reactor or mitigate consequences of a postulated piping failure without offsite power (with identification of those values that differ from the allowable limits by less than 10 percent).

COL 3.9(3) The COL applicant is to identify the site-specific active pumps.

COL 3.9(4) The COL applicant is to provide a summary of the maximum calculated total stress, deformation, and cumulative usage factor for each service limit of core support structures in accordance with ASME Section III.

COL 3.9(5) The COL applicant is to confirm the type of testing and frequency of site-specific pumps subject to IST in accordance with the ASME Code.

COL 3.9(6) The COL applicant is to confirm the type of testing and frequency of site-specific valves subject to IST in accordance with the ASME Code.

COL 3.9(7) The COL applicant is to provide a table listing all safety-related components that use snubbers in their support systems.

3.9.10 References

1. "ATWS: A Reappraisal, Part 3: Frequency of Anticipated Transients," EPRI-NP-2230, January 1982.

2. "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessment," NUREG/CR-3862, May 1985.

3. "Rates of Initiating Event at U.S. Nuclear Power Plants: 1987 - 1995," INEEL (Sponsored by U.S. NRC), NUREG/CR-5750 (INEEL/EXT-98-00401), February 1999.

4. "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," NUREG/CR-6928 (INL/EXT-06-11119) February 2007.

5. Computer code ABAQUS Version 6.10-1; Installed on DELL Workstation with Windows Server 2008; Verification Document No. 00000-SM-VV-038, Jan. 2011.

6. Computer code, ADLPIPE Version 3F10.1; Computers with Windows XP O/S; Verification Document No. 00000-SM-VV-015, Rev. 04, April 2011.

7. Computer code, CLEVER Version 1.0; Computers with Windows XP O/S; Verification Document No. 00000-SM-VV-037, Rev. 01, Oct. 2012.

8. Computer code, HeadPR Version 1; Computers with Windows XP, Windows 2000 O/S; Verification Document No. ND-G-CV-033, Rev. 0, July 2005.

9. Doherty, P. K., Software Verification and Validation Report of CEFLASH-4B, Version f4b.1.1, VV-FF-0178-1, January 1995.

10. Computer code, ANSYS Version 10.0; Installed on HP Integrity Superdome 16 way of Hewlett Packard Co.; Verification Document No. DAVM100, Rev. 0, July 2006.

11. Computer code, AFP2D Version 3; Installed on HP Integrity Superdome 16 way of Hewlett Packard Co.; Verification Document No. ND-G-CV-019, Rev. 7, Oct. 2008.

12. Computer Program, TSPOST Version 0; Installed on HP Integrity Superdome 16 Way of Hewlett Packard Co., Verification Document No. ND-G-CV-018, Rev. 2, Oct. 2008.

13. Computer code, AFPOST Version 2; Installed on HP Integrity Superdome 16 way of Hewlett Packard Co.; Verification Document No. ND-G-CV-027, Rev. 6, Feb, 2008.

14. Computer Program ATHOS3 Mod-01; Installed on HP Integrity Superdome 16 Way of Hewlett Packard Co., Verification Document No. ND-G-CV-017, Rev. 2, Dec, 2008.

15. Computer code, AFPOST+e Version 1; Installed on HP Integrity Superdome 16 way of Hewlett Packard Co.; Verification Document No. ND-G-CV-037, Rev. 1, June 2013.

16. Computer code, PTXIG Version 1.0; Computers with Microsoft.Net 2.0 O/S; Verification Document No. 00000-RM-VV-002, Rev. 02, Sep, 2012.

17. DST Computer Services SA, "a nuclear and non-nuclear piping analysis program," PIPESTRESS Version 3.7.0, Geneva, Switzerland, 2012.

18. American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code," Section III, Division 1, 2007 Edition with 2008 Addenda.

19. American Society of Mechanical Engineers, "Code for Pressure Piping, Power Piping," ASME B31.1, 2012 Edition.

20. American Society of Mechanical Engineers, "Code for Pressure Piping, Process Piping," ASME B31.3, 2012 Edition.

21. REFORC-DEC User Manual, REF 03.7.483-1.0, Rev. 1, D.J. Pichurski, S&L, 21 January 1994.

22. Computer code ANSYS Version 12.0, ANSYS, Inc., 2011.

23. "Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading," Ghosh, S. and Wilson, E., EERC 69-10, University of California, Berkeley, September 1969.

24. "Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions with Application of Analysis to C-E 800 MWe Class Reactors," Combustion Engineering, Inc., CENPD-42, August 1972 (Proprietary).

25. Gabrielson, V. K., "SHOCK, A Computer Code to Solve the Dynamic Response of Lumped-Mass Systems," SCL-DR-69-98, November 1969.

26. "Method for the Analysis of Blowdown Induced Forces in a Reactor Vessel," Combustion Engineering, Inc., CENPD-252-P-A, July 1979 (Proprietary).

27. "CEFLASH-4A: A Fortran-IV-Digital Computer Program for Reactor Blowdown Analysis," Combustion Engineering, Inc., CENPD-133P, August 1974 (Proprietary).

28. "CEFLASH-4A: A Fortran-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," Combustion Engineering, Inc., CENPD-133P, Supplement 2, February 1975 (Proprietary).

29. Scherer, A. E., Licensing Manager, (C-E), Letter to D. F. Ross, Assistant Director of Reactor Safety Division of Systems Safety, LD-76-026, March 1976 (Proprietary).

30. Parr, O.D., Chief Light Water Reactor Project Branch 1-3, Division of Reactor Licensing (NRC), Letter to F. M. Stern, Vice President of Projects (C-E), June 1975.

31. Kniel, K., Chief Light Water Reactors Branch No. 2, Letter to A. E. Scherer, Licensing Manager (C-E), August 1976 (Staff Evaluation of CENPD-213).

32. American Society of Mechanical Engineers, "Code for Operation and Maintenance of Nuclear Power Plants," ASME OM Code, New York, NY, USA, 2004 Edition with 2006 Addenda.

33. NRC RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Rev. 3, U.S. Nuclear Regulatory Commission, September 2009.

34. IEEE Std. 344-2004 (Reaffirmed 2009), "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers (IEEE), June 2005.

35. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” SECY-93-087, United States Regulatory Commission, April 2, 1993.

36. NRC RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” Rev. 1, U.S. Nuclear Regulatory Commission, March 2007.

37. NRC RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” Rev. 1, U.S. Nuclear Regulatory Commission, March 2007.

38. “Theory of Pump Induced Pulsating Coolant Pressure in PWRs,” Penzes, L.E., 2nd Int. Conf. on Structural Mechanics in Reactor Technology, Vol. II, Part E-F.

39. Horvay, G., Bowers, G., “Forced Vibration of a Shell Inside a Narrow Water Annulus,” Nuclear Engr. Design V34, 1975.

40. “Final Report on the Performance Evaluation of the Palo Verde Control Element Assembly Shroud,” Combustion Engineering, Inc., CEN-267-(V)-P Rev. 1-P, 1984.

41. “A Comprehensive Vibration Assessment Program for Palo Verde Nuclear Generating Station Unit 1 (System 80 Prototype),” Combustion Engineering, Inc., CEN-263(V)-P, Rev. 1-P, January 1985 (Proprietary).

42. M.K. Au-Yang, “Flow-Induced Vibration of Power and Process Plant Components,” Professional Engineering Publishing Limited, 2001.

43. Hurty, W. C., Rubinstein, M. F., “Dynamics of Structures,” Prentice-Hall, 1964.

44. “Random Vibrations, Elementary Theory, Structural Dynamics and Design, Signal Analysis and Testing,” University of Arizona Seminar, October 29 to November 2, 1990.

45. NRC RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” Rev. 3, U.S. Nuclear Regulatory Commission, March 2007.

46. “Structural Analysis of Fuel Assemblies for Seismic and Loss-of-Coolant Accident Loading,” Combustion Engineering, Inc., CENPD-178, Revision 1, August 1981.

47. Firtz, R.J., "The Effect of Liquids on the Dynamic Motions of Immersed Solids," Journal of Engineering for Industry, ASME Paper No. 71-Vibr-100.

48. McDonald, C.K., "Seismic Analysis of Vertical Pumps Enclosed in Liquid Filled Containers," ASME Paper No. 75-PVP-56.

49. American Society of Mechanical Engineers, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," ASME QME-1, 2007 Edition.

50. IEEE Std. 112-2004, "IEEE Standard Test Procedure for Polyphase Induction Motors and Generators," Institute of Electrical and Electronics Engineers (IEEE), November 2004.

51. Institute of Electrical and Electronics Engineers Power Engineering Society, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," IEEE-323-2003, New York, NY, USA, January 2004.

52. NRC RG 1.89, "Environmental Qualification of Certain Electric Equipment important to Safety for Nuclear Power Plants," Rev. 1, U.S. Nuclear Regulatory Commission, June 1984.

53. Institute of Electrical and Electronics Engineers Power Engineering Society, "IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants," IEEE-382-2006, New York, NY, USA, March 2007.

54. U.S. Nuclear Regulatory Commission, "Qualification Tests of Electric Valve Operators Installed Inside the Containment ofNuclear Power Plants," NRC RG 1.73, Washington, DC, January, 1974.

55. U.S. Nuclear Regulatory Commission, "Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions," Regulatory Issue Summary 2000-03, Washington, DC, USA, March 15, 2000.

56. U.S. Nuclear Regulatory Commission, "Guidelines for Inservice Testing at Nuclear Power Plants," NUREG-1482, Rev. 2, Washington, DC, USA, August 2011.

57. U.S. Nuclear Regulatory Commission, "Safety-Related Motor-Operated Valves, Testing and Surveillance," GL 89-10, Washington, DC, USA, June 1989.

58. U.S. Nuclear Regulatory Commission, "Periodic Verification of Design Basis Capability of Safety-Related Motor-Operated Valves," GL 96-05, Washington, DC, USA, September, 1996.

59. Joint Owners Group Air Operated Valve Program, Revision 1, December 13, 2000.

Table 3.9-1 (1 of 3)

Transients Used in Stress Analysis of Code Class 1 and CS Components

Event Title	Occurrences for Design Purposes	
	60 Years	40 Years
Level A Service Conditions		
Steady-state operation with normal NSSS parameter variations (5-100% power)	1,500,000	1,000,000
Daily load follow operation	22,000	15,000
Turbine power step changes of ±10% power (15-100% power)	3,200	2,000
Turbine power step changes of ±1% power (5-15% power)	1,600	1,000
Turbine load rejection of up to 50% power (50-100% power)	60	40
Turbine generator runback to house load	60	40
Reactor trip	150	100
Turbine trip	150	100
Turbine power ramp changes of ±5%/min (15-100% power)	3,200	2,000
Turbine power ramp changes of ±1%/min (5-15% power)	1,600	1,000
Loss of main feedwater pumps without reactor trip	60	40
NSSS operations with manual control of the CEAs, turbine bypass valves, pressurizer spray/heaters, pressurizer level control and feedwater flow (0-5% power)	1,600	1,000
Opening or closure of the economizer feedwater control valve	500	500
NSSS operations with the NSSS control systems in the manual mode (5-100% power)	3,200	2,000
Manual operation of the auxiliary spray system	250	170
High capacity steam generator blowdown	3,200	2,000
Shift from normal to maximum CVCS flow rate	3,200	2,000
Low-low VCT level and charging pump diversion to the boric acid storage tank	60	40
Spurious actuation of the pressurizer spray	60	40
Spurious actuation of the pressurizer heaters	60	40
Inadvertent closure of one economizer or downcomer feedwater control valve	60	40
Inadvertent opening of one economizer or downcomer feedwater control valve	60	40
Inadvertent isolation of one main FW heater	60	40
Startup and coastdown of a reactor coolant pump at HSB	2,000	1,340
Startup and shutdown of the shutdown cooling system at HSD	250	170
Spurious startup of a safety injection pump during shutdown condition	60	40
Spurious actuation of the pressurizer heaters at HSB	60	40
Plant heatup and cooldown	250	170

Table 3.9-1 (2 of 3)

Event Title	Occurrences for Design Purposes	
	60 Year	40 Year
Level B Service Conditions		
Decrease in feedwater temperature	20	20
Increase in feedwater flow rate	20	20
Increase in steam flow rate	20	20
Inadvertent opening of a main steam safety valve	10	10
Loss of external load	20	20
Loss of condenser vacuum	20	20
Loss of non-emergency AC power to the station auxiliaries	20	20
Main steam isolation valve closure	20	20
Loss of normal feedwater flow	20	20
Loss of forced reactor coolant flow	20	20
Natural circulation cooldown (HSB to HSD)	10	10
Uncontrolled CEA withdrawal at low power	5	5
Uncontrolled CEA withdrawal at power	5	5
Control rod misoperation, RPCS inadvertent operation or operator error	50	35
Loss of component cooling water to the letdown heat exchanger	10	10
CVCS mal-function that increases RCS inventory	10	10
Inadvertent opening of pilot operated safety relief valve (POSRV closed as expected)	10	10
Failure of small lines carrying coolant outside containment (letdown line break)	20	20
Reactor coolant pump seal failure	10	10
Loss of seal injection concurrent with loss of cooling water	5	5
Level C Service Conditions		
None		

Table 3.9-1 (3 of 3)

Event Title	Occurrences for Design Purposes	
	60 Year	40 Year
Level D Service Conditions		
Steam system piping failure	1	1
Feedwater system line break (FWLB)	1	1
Reactor coolant pump rotor seizure	1	1
Reactor coolant pump shaft break	1	1
Rod ejection accident	1	1
Inadvertent opening of pilot operated safety relief valve (POSRV fails to close)	1	1
Steam generator tube rupture (SGTR)	1	1
Loss of coolant accidents resulting from postulated pipe breaks within the RCS pressure boundary (LOCA)	1	1
Total loss of feedwater flow	1	1
Test Conditions		
RCS hydrostatic test	15	10
Secondary hydrostatic test	15	10
RCS leak test	200	200
Secondary leak test	200	200
SIS/SCS pre-operational and maintenance test	360	240
SIS check valve operability test	120	80

Table 3.9-2

Loading Combinations for ASME Code Class 1, 2, and 3 Components(1) and Component Supports

Condition	Design Loading(2) Combination
Design	PD + DW + IRWST
Level A (Normal)(3)	PO + DW
Level B (Upset)(3)	PO + DW + IRWST
Level C (Emergency)	PO + DW + DE
Level D (Faulted)	PO + DW + SRSS (SSE + (DF + IRWST))

(1) For piping, see Tables 3.9-10 and 3.9-11.

(2) Legend:

PD = design pressure

PO = operating pressure

DW = dead weight

SSE = safe shutdown earthquake

DE = dynamic system loadings associated with the emergency condition

DF = dynamic system loadings associated with pipe breaks (not eliminated by a leak-before-break analysis)

IRWST = In-containment refueling water storage tank discharge loads

(3) As required by the ASME Section III, other loads, such as thermal transient, and thermal gradient, require consideration in addition to the primary stress producing loads listed. SSE is considered in equipment fatigue evaluations in accordance with Subsection 3.7.3.1.

Table 3.9-3

Stress Limits for ASME Code Class 1 Components, Piping, and Component Supports

	Component and Piping Stress Limits(1)	Component Support Stress Limits(2)
Design	NB-3221, NB-3231, and NB-3652	Table NF-3131(a) - 1
Level A (Normal)	NB-3222, NB-3232, and NB-3653	Table NF-3131(a) - 1
Level B (Upset)	NB-3223, NB-3233, and NB-3654	Table NF-3131(a) - 1
Level C (Emergency)	NB-3224, NB-3234, and NB-3655	Table NF-3131(a) - 1
Level D (Faulted)(3)	NB-3225, NB-3235, and NB-3656	Table NF-3131(a) - 1

(1) Stress limits listed are used as required by the ASME Section III, and applicable addenda for all components except active components. Active components are designed to the stress limits of NB-3221 and NB-3231 for design conditions and the stress limits of NB-3222 and NB-3232 for all other conditions for active components.

(2) Stress limits used are as required by the ASME Section III and modified by NRC RGs 1.124 and 1.130.

(3) For faulted condition loadings, bolts in the load path connecting two members of an NF support for Class 1 components are designed in accordance with Appendix F of the ASME Section III for friction type connections with tensile stresses limited to the lesser of 0.7 Su or Sy.

Table 3.9-4 (1 of 20)

Seismic Category I Active Valves

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
SI-100	IRWST recirculation isolation (operate)	Check	2	None
SI-101	IRWST recirculation isolation (operate)	Check	2	None
SI-113	SIS (operate)	Check	2	None
SI-123	SIS (operate)	Check	2	None
SI-133	SIS (operate)	Check	2	None
SI-143	SIS (operate)	Check	2	None
SI-157	CSS (operate)	Check	2	None
SI-158	CSS (operate)	Check	2	None
SI-159	SC pump suction check	Check	2	None
SI-160	SC pump suction check	Check	2	None
SI-168	SCS (operate)	Check	2	None
SI-178	SCS (operate)	Check	2	None
SI-179	SCS (operate)	Relief	2	None
SI-189	SCS (operate)	Relief	2	None
SI-215	SI tank (operate)	Check	1	None
SI-217	SI system (operate)	Check	1	None
SI-225	SI tank (operate)	Check	1	None
SI-227	SI system (operate)	Check	1	None
SI-235	SI tank (operate)	Check	1	None
SI-237	SI system (operate)	Check	1	None
SI-245	SI tank (operate)	Check	1	None
SI-247	SI system (operate)	Check	1	None
SI-300	CS/SCS IRWST recirculation isolation	Gate	2	Motor
SI-301	CS/SCS IRWST recirculation isolation	Gate	2	Motor
SI-302	SI IRWST recirculation isolation	Gate	2	Motor
SI-303	SI IRWST recirculation isolation	Globe	2	Motor
SI-304	IRWST isolation	Gate	2	Motor
SI-305	IRWST isolation	Gate	2	Motor
SI-308	IRWST isolation	Gate	2	Motor
SI-309	IRWST isolation	Gate	2	Motor
SI-310	SCS 1 flow control (operate)	Globe	2	Motor

Table 3.9-4 (2 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
SI-311	SCS 2 flow control (operate)	Globe	2	Motor
SI-312	SDCHX bypass (operate)	Globe	2	Motor
SI-313	SDCHX bypass (operate)	Globe	2	Motor
SI-314	SCS 1 IRWST recirculation line flow control	Globe	2	Motor
SI-315	SCS 2 IRWST recirculation line flow control	Globe	2	Motor
SI-321	Hot leg injection (operate)	Globe	2	Motor
SI-322	Hot leg injection leakage return (close)	Globe	1	Pneumatic
SI-331	Hot leg injection (operate)	Globe	2	Motor
SI-332	Hot leg injection leakage return (close)	Globe	1	Pneumatic
SI-340	SCS/CSS pump suction cross connection (close)	Gate	2	Motor
SI-341	SCS/CSS pump discharge cross(close) connection (operate)	Gate	2	Motor
SI-342	SCS/CSS pump suction cross connection (close)	Gate	2	Motor
SI-343	SCS/CSS pump discharge cross connection (close)	Gate	2	Motor
SI-344	SC pump suction isolation (close)	Gate	2	Motor
SI-346	SC pump suction isolation (close)	Gate	2	Motor
SI-347	CS pump suction isolation (operate)	Gate	2	Motor
SI-348	CS pump suction isolation (operate)	Gate	2	Motor
SI-391	Reactor cavity isolation (operate)	Gate	2	Motor
SI-393	Reactor cavity isolation (operate)	Gate	2	Motor
SI-395	IRWST return line isolation (operate)	Gate	2	Motor
IW-0001	HVT flooding isolation (close)	Gate	2	Motor
IW-0002	HVT flooding isolation (close)	Gate	2	Motor
IW-0003	Reactor cavity flooding isolation (close)	Gate	2	Motor
IW-0004	Reactor cavity flooding isolation (close)	Gate	2	Motor
IW-0005	BAMP suction isolation (close)	Gate	2	Motor
IW-0006	BAMP suction isolation (close)	Gate	2	Motor
IW-0010	IRWST level instrument isolation (close)	Globe	2	Solenoid

Table 3.9-4 (3 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
IW-0011	IRWST level instrument isolation (close)	Globe	2	Solenoid
IW-0012	HVT level instrument isolation (close)	Globe	2	Solenoid
IW-0013	HVT level instrument isolation (close)	Globe	2	Solenoid
IW-0014	HVT level instrument isolation (close)	Globe	2	Solenoid
IW-0015	HVT level instrument isolation (close)	Globe	2	Solenoid
IW-0016	HVT level instrument isolation (close)	Globe	2	Solenoid
IW-0017	HVT level instrument isolation (close)	Globe	2	Solenoid
IW-0018	Reactor cavity level instrument isolation (close)	Globe	2	Solenoid
IW-0019	Reactor cavity level instrument isolation (close)	Globe	2	Solenoid
IW-0020	Reactor cavity level instrument isolation (close)	Globe	2	Solenoid
IW-0021	Reactor cavity level instrument isolation (close)	Globe	2	Solenoid
IW-0022	IRWST level instrument isolation (close)	Globe	2	Solenoid
IW-0023	IRWST level instrument isolation (close)	Globe	2	Solenoid
IW-0024	IRWST level instrument isolation (close)	Globe	2	Solenoid
IW-0025	IRWST level instrument isolation (close)	Globe	2	Solenoid
IW-0026	IRWST level instrument isolation (close)	Globe	2	Solenoid
IW-0027	IRWST level instrument isolation (close)	Globe	2	Solenoid
IW-0028	HVT level instrument isolation (close)	Globe	2	Solenoid
IW-0029	HVT level instrument isolation (close)	Globe	2	Solenoid
IW-0030	HVT level instrument isolation (close)	Globe	2	Solenoid
IW-0031	HVT level instrument isolation (close)	Globe	2	Solenoid
IW-0032	Reactor cavity level instrument isolation (close)	Globe	2	Solenoid
IW-0033	Reactor cavity level instrument isolation (close)	Globe	2	Solenoid
IW-0034	Reactor cavity level instrument isolation (close)	Globe	2	Solenoid
IW-0035	Reactor cavity level instrument isolation (close)	Globe	2	Solenoid
IW-1003	BAMP suction line relief (operate)	Relief	2	None
SI-404	Safety injection system (operate)	Check	2	None
SI-405	Safety injection system (operate)	Check	2	None
SI-424	SI pump minimum flow IRWST return (operate)	Check	2	None
SI-426	SI pump minimum flow IRWST return (operate)	Check	2	None

Table 3.9-4 (4 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
SI-434	Safety injection system (operate)	Check	2	None
SI-446	Safety injection system (operate)	Check	2	None
SI-448	SI pump minimum flow IRWST return (operate)	Check	2	None
SI-451	SI pump minimum flow IRWST return (operate)	Check	2	None
SI-522	Hot leg injection (operate)	Check	1	None
SI-523	Hot leg injection (operate)	Check	1	None
SI-532	Hot leg injection (operate)	Check	1	None
SI-533	Hot leg injection (operate)	Check	1	None
SI-540	SIS (operate)	Check	1	None
SI-541	SIS (operate)	Check	1	None
SI-542	SIS (operate)	Check	1	None
SI-543	SIS (operate)	Check	1	None
SI-568	SCS (operate)	Check	2	None
SI-569	SCS (operate)	Check	2	None
SI-600	SCS isolation (operate)	Globe	2	Motor
SI-601	SCS isolation (operate)	Globe	2	Motor
SI-602	SIS throttle (operate)	Globe	2	Motor
SI-603	SIS throttle (operate)	Globe	2	Motor
SI-604	Hot Leg Injection (Operate)	Gate	2	Motor
SI-605	SI tank vent (operate)	Globe	2	Solenoid
SI-606	SI tank vent (operate)	Globe	2	Solenoid
SI-607	SI tank vent (operate)	Globe	2	Solenoid
SI-608	SI tank vent (operate)	Globe	2	Solenoid
SI-609	Hot leg injection (operate)	Gate	2	Motor
SI-611	SI tank fill/drain (operate)	Globe	2	Pneumatic
SI-613	SI tank vent (operate)	Globe	2	Solenoid
SI-614	SI tank isolation (operate)	Gate	1	Motor
SI-616	SI system (operate)	Globe	2	Motor
SI-618	Leakage return to IRWST/RDT (close)	Globe	1	Pneumatic
SI-621	SI tank fill/drain (close)	Globe	2	Pneumatic
SI-623	SI tank vent (operate)	Globe	2	Solenoid

Table 3.9-4 (5 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
SI-624	SI tank isolation (operate)	Gate	1	Motor
SI-626	SI isolation (operate)	Globe	2	Motor
SI-628	Leakage return to IRWST/RDT (close)	Globe	1	Pneumatic
SI-631	SI tank fill/drain (close)	Globe	2	Pneumatic
SI-633	SI tank vent (operate)	Globe	2	Solenoid
SI-634	SI tank isolation (operate)	Gate	1	Motor
SI-636	SI system (operate)	Globe	2	Motor
SI-638	Leakage return to IRWST/RDT (close)	Globe	1	Pneumatic
SI-641	SI tank fill/drain (close)	Globe	2	Pneumatic
SI-643	SI tank vent (operate)	Globe	2	Solenoid
SI-644	SI tank isolation (operate)	Gate	1	Motor
SI-646	SI isolation (operate)	Globe	2	Motor
SI-648	Leakage return to IRWST/RDT (close)	Globe	1	Pneumatic
SI-651	Shutdown cooling suction (operate)	Gate	1	Motor
SI-652	Shutdown cooling suction (operate)	Gate	1	Motor
SI-653	Shutdown cooling suction (operate)	Gate	1	Motor
SI-654	Shutdown cooling suction (operate)	Gate	1	Motor
SI-655	Shutdown cooling suction (operate)	Gate	2	Motor
SI-656	Shutdown cooling suction (operate)	Gate	2	Motor
SI-682	SIT fill/drain (close)	Globe	2	Pneumatic
SI-688	SCS 1 IRWST recirculation isolation (operate)	Gate	2	Motor
SI-690	SCS warmup line flow control (operate)	Globe	2	Motor
SI-691	SCS warmup line flow control (operate)	Globe	2	Motor
SI-693	SCS 2 IRWST recirculation isolation (operate)	Gate	2	Motor
SI-801	External emergency injection line check	Check	2	None
SI-803	External emergency injection line isolation	Gate	2	Manual
CV-189	IRWST makeup line check (close)	Check	2	None
CV-255	Seal injection containment isolation	Globe	2	Motor
CV-362	Shutdown purification line isolation	Gate	2	Manual (handwheel)
CV-363	Shutdown purification line check (close)	Check	2	None
CV-494	RSSH to reactor drain header check (close)	Check	2	None

Table 3.9-4 (6 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
CV-505	RCP controlled bleed-off containment isolation (close)	Globe	2	Pneumatic
CV-506	RCP controlled bleed-off containment isolation (close)	Globe	2	Pneumatic
CV-509	IRWST makeup line containment isolation (close)	Gate	2	Motor
CV-515	Letdown isolation (close)	Globe	1	Pneumatic
CV-516	Letdown isolation (close)	Globe	1	Pneumatic
CV-522	Letdown containment isolation (close)	Globe	2	Pneumatic
CV-523	Letdown containment isolation (close)	Globe	2	Pneumatic
CV-524	Charging containment isolation (close)	Globe	2	Motor
CV-560	RDT effluent containment isolation (close)	Globe	2	Pneumatic
CV-561	RDT effluent containment isolation (close)	Globe	2	Pneumatic
CV-576	Charging flow restricting (close)	Globe	2	Motor
CV-577	Charging flow restricting (close)	Globe	2	Motor
CV-580	RSSH to RDH isolation (close)	Gate	2	Pneumatic
CV-747	Charging line check (close)	Check	2	None
CV-835	Seal injection containment isolation (close)	Check	2	None
RC-120	ReaC-tor C-oolant System	to be determined by vendor	1	Motor
RC-121	ReaC-tor C-oolant System	to be determined by vendor	1	Motor
RC-122	ReaC-tor C-oolant System	to be determined by vendor	1	Motor
RC-123	ReaC-tor C-oolant System	to be determined by vendor	1	Motor
RC-124	ReaC-tor C-oolant System	to be determined by vendor	1	Motor
RC-125	ReaC-tor C-oolant System	to be determined by vendor	1	Motor
RC-126	Reactor Coolant System	to be determined by vendor	1	Motor
RC-127	Reactor Coolant System	to be determined by vendor	1	Motor
RC-130	Reactor Coolant System	to be determined by vendor	1	Motor

Table 3.9-4 (7 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
RC-131	Reactor Coolant System	to be determined by vendor	1	Motor
RC-132	Reactor Coolant System	to be determined by vendor	1	Motor
RC-133	Reactor Coolant System	to be determined by vendor	1	Motor
RC-134	Reactor Coolant System	to be determined by vendor	1	Motor
RC-135	Reactor Coolant System	to be determined by vendor	1	Motor
RC-136	Reactor Coolant System	to be determined by vendor	1	Motor
RC-137	Reactor Coolant System	to be determined by vendor	1	Motor
RC-200	Reactor Coolant System	POSRV	1	Pilot
RC-201	Reactor Coolant System	POSRV	1	Pilot
RC-202	Reactor Coolant System	POSRV	1	Pilot
RC-203	Reactor Coolant System	POSRV	1	Pilot
RC-300	Reactor Coolant System	Spring-loaded	1	None
RC-301	Reactor Coolant System	Spring-loaded	1	None
RC-302	Reactor Coolant System	Spring-loaded	1	None
RC-303	Reactor Coolant System	Spring-loaded	1	None
RC-304	Reactor Coolant System	Spring-loaded	1	None
RC-305	Reactor Coolant System	Spring-loaded	1	None
RC-306	Reactor Coolant System	Spring-loaded	1	None
RC-307	Reactor Coolant System	Spring-loaded	1	None
VQ-0011	Reactor containment purge system (close)	Butterfly	2	Electro-hydraulic
VQ-0013	Reactor containment purge system (close)	Butterfly	2	Electro-hydraulic
VQ-0014	Reactor containment purge system (close)	Butterfly	2	Motor
VQ-0031	Reactor containment purge system (close)	Butterfly	2	Pneumatic
VQ-0032	Reactor containment purge system (close)	Butterfly	2	Pneumatic
VQ-0033	Reactor containment purge system (close)	Butterfly	2	Pneumatic
VQ-0034	Reactor containment purge system (close)	Butterfly	2	Pneumatic

Table 3.9-4 (8 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
WI-0012	Plant chilled water system	Gate	2	Pneumatic
WI-0013	Plant chilled water system	Gate	2	Pneumatic
WI-0014	Plant chilled water system	Relief	2	None
WI-0015	Plant chilled water system	Gate	2	Motor
WI-1043	Plant chilled water system	Check	2	None
WO-0906A	Essential chilled water system	Threeway	3	Pneumatic
WO-0906B	Essential chilled water system	Threeway	3	Pneumatic
WO-1001A	Essential chilled water system	Relief	3	None
WO-1001B	Essential chilled water system	Relief	3	None
WO-1003A	Essential chilled water system	Check	3	None
WO-1003B	Essential chilled water system	Check	3	None
WO-1010A	Essential chilled water system	Check	3	None
WO-1010B	Essential chilled water system	Check	3	None
WO-1011A	Essential chilled water system	Check	3	None
WO-1011B	Essential chilled water system	Check	3	None
WO-1014A	Essential chilled water system	Check	3	None
WO-1014B	Essential chilled water system	Check	3	None
WO-1022A	Essential chilled water system	Check	3	None
WO-1022B	Essential chilled water system	Check	3	None
WO-1031A	Essential chilled water system	Check	3	None
WO-1031B	Essential chilled water system	Check	3	None
WO-1032A	Essential chilled water system	Check	3	None
WO-1032B	Essential chilled water system	Check	3	None
DO-1005A	Diesel fuel oil transfer system (operate)	Check	3	None
DO-1007A	Diesel fuel oil transfer system (operate)	Check	3	None
DO-1005B	Diesel fuel oil transfer system (operate)	Check	3	None
DO-1007B	Diesel fuel oil transfer system (operate)	Check	3	None

Table 3.9-4 (9 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
MS-0011	Main steam system isolation (close)	Gate	2	Gas hydraulic
MS-0012	Main steam system isolation (close)	Gate	2	Gas hydraulic
MS-0013	Main steam system isolation (close)	Gate	2	Gas hydraulic
MS-0014	Main steam system isolation (close)	Gate	2	Gas hydraulic
MS-0101	Main steam atmospheric dump (close)	Angle	2	Electro-hydraulic
MS-0102	Main steam atmospheric dump (close)	Angle	2	Electro-hydraulic
MS-0103	Main steam atmospheric dump (close)	Angle	2	Electro-hydraulic
MS-0104	Main steam atmospheric dump (close)	Angle	2	Electro-hydraulic
MS-1301	Main steam safety (operate)	Safety	2	None
MS-1302	Main steam safety (operate)	Safety	2	None
MS-1303	Main steam safety (operate)	Safety	2	None
MS-1304	Main steam safety (operate)	Safety	2	None
MS-1305	Main steam safety (operate)	Safety	2	None
MS-1306	Main steam safety (operate)	Safety	2	None
MS-1307	Main steam safety (operate)	Safety	2	None
MS-1308	Main steam safety (operate)	Safety	2	None
MS-1309	Main steam safety (operate)	Safety	2	None
MS-1310	Main steam safety (operate)	Safety	2	None
MS-1311	Main steam safety (operate)	Safety	2	None
MS-1312	Main steam safety (operate)	Safety	2	None
MS-1313	Main steam safety (operate)	Safety	2	None
MS-1314	Main steam safety (operate)	Safety	2	None
MS-1315	Main steam safety (operate)	Safety	2	None
MS-1316	Main steam safety (operate)	Safety	2	None
MS-1317	Main steam safety (operate)	Safety	2	None
MS-1318	Main steam safety (operate)	Safety	2	None
MS-1319	Main steam safety (operate)	Safety	2	None
MS-1320	Main steam safety (operate)	Safety	2	None

Table 3.9-4 (10 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
MS-0015	MSIV bypass (close)	Gate	2	Hydraulic
MS-0016	MSIV bypass (close)	Gate	2	Hydraulic
MS-0017	MSIV bypass (close)	Gate	2	Hydraulic
MS-0018	MSIV bypass (close)	Gate	2	Hydraulic
MS-0109	AFW pump turbine steam supply (open)	Globe	2	Solenoid
MS-0110	AFW pump turbine steam supply (open)	Globe	2	Solenoid
MS-0111	AFW pump turbine warmup (open)	Globe	2	Solenoid
MS-0112	AFW pump turbine warmup (open)	Globe	2	Solenoid
MS-105	MSADV isolation valve (operate)	Gate	2	Motor
MS-106	MSADV isolation valve (operate)	Gate	2	Motor
MS-107	MSADV isolation valve (operate)	Gate	2	Motor
MS-108	MSADV isolation valve (operate)	Gate	2	Motor
MS-0090	Main steam drip leg isolation valve	Globe	2	Electro-hydraulic
MS-0091	Main steam drip leg isolation valve	Globe	2	Electro-hydraulic
MS-0092	Main steam drip leg isolation valve	Globe	2	Electro-hydraulic
MS-0093	Main steam drip leg isolation valve	Globe	2	Electro-hydraulic
FW-0121	SG 1 economizer FW isolation (close)	Gate	2	Electro-hydraulic
FW-0122	SG 1 economizer FW isolation (close)	Gate	2	Electro-hydraulic
FW-0123	SG 2 economizer FW isolation (close)	Gate	2	Electro-hydraulic
FW-0124	SG 2 economizer FW isolation (close)	Gate	2	Electro-hydraulic
FW-0131	SG 1 economizer FW isolation (close)	Gate	2	Electro-hydraulic
FW-0132	SG 1 economizer FW isolation (close)	Gate	2	Electro-hydraulic
FW-0133	SG 2 economizer FW isolation (close)	Gate	2	Electro-hydraulic
FW-0134	SG 2 economizer FW isolation (close)	Gate	2	Electro-hydraulic

Table 3.9-4 (11 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
FW-1039	SG 1 downcomer FW line (operate)	Check	2	None
FW-1040	SG 1 downcomer FW line (operate)	Check	2	None
FW-1046	SG 2 downcomer FW line (operate)	Check	2	None
FW-1047	SG 2 downcomer FW line (operate)	Check	2	None
FW-1035	SG 1 economizer FW line (operate)	Check	2	None
FW-1037	SG 1 economizer FW line (operate)	Check	2	None
FW-1042	SG 2 economizer FW line (operate)	Check	2	None
FW-1043	SG 2 economizer FW line (operate)	Check	2	None
FW-1044	SG 2 economizer FW line (operate)	Check	2	None
FW-0138	FW chemical injection valve	Globe	2	Pneumatic
FW-0139	FW chemical injection valve	Globe	2	Pneumatic
FW-1050	FW chemical injection check valve	Check	2	None
FW-1051	FW chemical injection check valve	Check	2	None
NT-0004	Nitrogen system containment isolation (close)	Globe	2	Pneumatic
NT-1016	Nitrogen system containment isolation (operate)	Check	2	None
IA-0020	Instrument air system containment isolation (close)	Globe	2	Pneumatic
IA-1601	Instrument air system containment isolation (operate)	Check	2	None
AF-1012A	AFW recirculation (operate)	Check	3	None
AF-1014A	AFW recirculation (operate)	Check	3	None
AF-1012B	AFW recirculation (operate)	Check	3	None
AF-1014B	AF recirculation (operate)	Check	3	None
AF-1003A	AF discharge (operate)	Check	3	None
AF-1004A	AF discharge (operate)	Check	3	None
AF-1003B	AF discharge (operate)	Check	3	None

Table 3.9-4 (12 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
AF-0035	AF modulation (open)	Globe	3	Solenoid
AF-0037	AF modulation (open)	Globe	3	Solenoid
AF-0038	AF modulation (open)	Globe	3	Solenoid
AF-0043	AF isolation (closed)	Gate	2	Motor
AF-0044	AF isolation (closed)	Gate	2	Motor
AF-0045	AF isolation (closed)	Gate	2	Motor
AF-0046	AF isolation (closed)	Gate	2	Motor
AF-1007A	AF isolation (operate)	Check	2	None
AF-1008A	AF isolation (operate)	Check	2	None
AF-1007B	AF isolation (operate)	Check	2	None
AF-1008B	AF isolation (operate)	Check	2	None
[[AF-1022A]]	AF chemical injection (operate)	Check	3	None
[[AF-1022B]]	AF chemical injection (operate)	Check	3	None
AF-1024A	AF chemical injection (operate)	Check	3	None
AF-1024B	AF chemical injection (operate)	Check	3	None
AT-0009	AF pump turbine steam isolation (open)	Globe	3	Pneumatic
AT-0010	AF pump turbine steam isolation (open)	Globe	3	Pneumatic
AT-3001	AFWPT electrical trip solenoid valve	Gate	3	Solenoid
AT-3002	AFWPT electrical trip solenoid valve	Gate	3	Solenoid
AT-0007	Steam supply line drip leg control (open)	Globe	3	Pneumatic
AT-0008	Steam supply line drip leg control (open)	Globe	3	Pneumatic
AT-3015	AFWPT throttle (open)	Globe	3	Pneumatic
AT-3016	AFWPT throttle (open)	Globe	3	Pneumatic
AT-1020A	AFWPT main steam supply (operate)	Check	3	None
AT-1020B	AFWPT main steam supply (operate)	Check	3	None
AT-1022A	AFWPT auxiliary steam supply (operate)	Check	3	None
AT-1022B	AFWPT auxiliary steam supply (operate)	Check	3	None

Table 3.9-4 (13 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
WM-1752	Demineralized water makeup system containment isolation (operate)	Check	2	None
DG-3023A	Safety relief valve	Safety	3	None
DG-3023B	Safety relief valve	Safety	3	None
DG-3031A	Safety relief valve	Safety	3	None
DG-3031B	Safety relief valve	Safety	3	None
DG-3035A	Globe valve	Safety	3	None
DG-3035B	Globe valve	Safety	3	None
DG-3037A	Globe valve	Safety	3	None
DG-3037B	Globe valve	Safety	3	None
DG-3059A	Gate valve	Gate	3	None
DG-3059B	Gate valve	Gate	3	None
DG-3098A	Globe valve	Globe	3	None
DG-3098B	Globe valve	Globe	3	None
DG-3099A	Globe valve	Globe	3	None
DG-3099B	Globe valve	Globe	3	None
DG-3114A	Three-way valve	Threeway	3	Self-controlled
DG-3114B	Three-way valve	Threeway	3	Self-controlled
DG-3217A	Three-way valve	Threeway	3	Self-controlled
DG-3217B	Three-way valve	Threeway	3	Self-controlled
DG-3250A	Three-way valve	Threeway	3	Self-controlled
DG-3250B	Three-way valve	Threeway	3	Self-controlled
DG-4022A	Check valve	Check	3	None
DG-4022B	Check valve	Check	3	None
DG-4030A	Check valve	Check	3	None
DG-4030B	Check valve	Check	3	None
DG-4034A	Check valve	Check	3	None
DG-4034B	Check valve	Check	3	None

Table 3.9-4 (14 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
DG-4038A	Check valve	Check	3	None
DG-4038B	Check valve	Check	3	None
DG-4109A	Check valve	Check	3	None
DG-4109B	Check valve	Check	3	None
DG-4111A	Check valve	Check	3	None
DG-4111B	Check valve	Check	3	None
DG-4140A	Check valve	Check	3	None
DG-4140B	Check valve	Check	3	None
DG-4214A	Check valve	Check	3	None
DG-4214B	Check valve	Check	3	None
DG-4222A	Check valve	Check	3	None
DG-4222B	Check valve	Check	3	None
DG-4242A	Check valve	Check	3	None
DG-4242B	Check valve	Check	3	None
DG-4319A	Check valve	Check	3	None
DG-4319B	Check valve	Check	3	None
DG-4321A	Check valve	Check	3	None
DG-4321B	Check valve	Check	3	None
CA-0013	Condenser exhaust gas containment isolation valve	Gate	2	Motor
CA-1023	Condenser exhaust gas containment isolation check valve	Check	2	None
SW-0035	ESW scrn wash PP Trn A disch	Butterfly	3	Motor
SW-0036	ESW scrn wash PP Trn B disch	Butterfly	3	Motor
SW-1301A	ESW scrn wash PP Trn A disch check valve	Check	3	None
SW-1301B	ESW scrn wash PP Trn A disch check valve	Check	3	None
SW-1401A	ESW scrn wash PP Trn A disch check valve	Check	3	None
SW-1401B	ESW scrn wash PP trn A disch check valve	Check	3	None
FP-0030	Reactor containment building fire water standpipe system containment isolation valve	Globe	2	Pneumatic

Table 3.9-4 (15 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
FP-1440	Reactor containment building fire water standpipe system containment isolation check valve	Check	2	None
CS-0001	Spray header isolation (operate)	Gate	2	Motor
CS-0002	Spray header isolation (operate)	Gate	2	Motor
CS-0003	Spray header isolation (operate)	Gate	2	Motor
CS-0004	Spray header isolation (operate)	Gate	2	Motor
CS-0005	IRWST return line isolation (operate)	Globe	2	Motor
CS-0006	IRWST return line isolation (operate)	Globe	2	Motor
CS-0007	IRWST return line isolation (operate)	Gate	2	Motor
CS-0008	IRWST return line isolation (operate)	Gate	2	Motor
CS-1001	Spray line (operate)	Check	2	None
CS-1002	Spray line (operate)	Check	2	None
CS-1005	Spray line relief (operate)	Relief	2	None
CS-1006	Spray line relief (operate)	Relief	2	None
CS-1007	Spray header isolation (operate)	Check	2	None
CS-1008	Spray header isolation (operate)	Check	2	None
CS-1014	ECSBS line (operate)	Check	2	None
SD-0005	SG blowdown isolation (close)	Gate	2	Pneumatic
SD-0006	SG blowdown isolation (close)	Gate	2	Pneumatic
SD-0007	SG blowdown isolation (close)	Gate	2	Motor
SD-0008	SG blowdown isolation (close)	Gate	2	Motor
RG-0410	Reactor coolant gas vent system (close)	Globe	1	Solenoid
RG-0411	Reactor coolant gas vent system (close)	Globe	1	Solenoid
RG-0412	Reactor coolant gas vent system (close)	Globe	1	Solenoid
RG-0413	Reactor coolant gas vent system (close)	Globe	1	Solenoid
RG-0414	Reactor coolant gas vent system (close)	Globe	1	Solenoid
RG-0415	Reactor coolant gas vent system (close)	Globe	1	Solenoid
RG-0416	Reactor coolant gas vent system (close)	Globe	1	Solenoid
RG-0417	Reactor coolant gas vent system (close)	Globe	1	Solenoid
RG-0419	Reactor coolant gas vent system (close)	Globe	2	Solenoid
RG-0420	Reactor coolant gas vent system (close)	Globe	2	Solenoid

Table 3.9-4 (16 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
PX-0001	Hot leg sample line isolation (close)	Globe	2	Solenoid
PX-0002	Hot leg sample line isolation (close)	Globe	2	Solenoid
PX-0003	PZR surge line sample isolation (close)	Globe	2	Solenoid
PX-0004	PZR surge line sample isolation (close)	Globe	2	Solenoid
PX-0005	PZR steam space sample isolation (close)	Globe	2	Solenoid
PX-0006	PZR steam space sample isolation (close)	Globe	2	Solenoid
PX-0020	SI tank sample isolation (close)	Globe	2	Solenoid
PX-0021	SI tank sample isolation (close)	Globe	2	Solenoid
PX-0022	SI tank sample isolation (close)	Globe	2	Solenoid
PX-0023	SI tank sample isolation (close)	Globe	2	Solenoid
PX-0024	SI tank sample isolation (close)	Globe	2	Solenoid
PX-0025	SI tank sample isolation (close)	Globe	2	Solenoid
PX-0026	CS pump 1 miniflow (close)	Globe	2	Solenoid
PX-0027	CS pump 2 miniflow (close)	Globe	2	Solenoid
PX-0041	Containment air sample line isolation (close)	Gate	2	Motor
PX-0042	Containment air sample line isolation (close)	Gate	2	Motor
PX-0043	Containment air sample return line isolation (close)	Gate	2	Motor
PX-1020	Containment air sample return line isolation (operate)	Check	2	None
PX-0039	Pass sample isolation (close)	Globe	2	Solenoid
PX-0053	Return to HVT Isolation (close)	Globe	2	Solenoid
PX-1023	Return to HVT Isolation (operate)	Check	2	None
CC-0011	CCW system (operate)	Globe	3	Motor
CC-0012	CCW system (operate)	Globe	3	Motor
CC-0021	CCW system (open)	Butterfly	3	Motor
CC-0022	CCW system (open)	Butterfly	3	Motor
CC-0023	CCW system (open)	Butterfly	3	Motor
CC-0024	CCW system (open)	Butterfly	3	Motor
CC-0027	CCW system (close)	Butterfly	3	Motor
CC-0028	CCW system (close)	Butterfly	3	Motor
CC-0037	CCW system (close)	Butterfly	3	Motor
CC-0038	CCW system (close)	Butterfly	3	Motor

Table 3.9-4 (17 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
CC-0149	CCW system (close)	Butterfly	3	Motor
CC-0191	CCW system (close)	Butterfly	3	Motor
CC-0192	CCW system (close)	Butterfly	3	Motor
CC-0025	CCW system (open)	Butterfly	3	Motor
CC-0026	CCW system (open)	Butterfly	3	Motor
CC-0031	CCW system (open)	Butterfly	3	Motor
CC-0032	CCW system (open)	Butterfly	3	Motor
CC-0033	CCW system (open)	Butterfly	3	Motor
CC-0034	CCW system (open)	Butterfly	3	Motor
CC-0035	CCW system (open)	Butterfly	3	Motor
CC-0036	CCW system (open)	Butterfly	3	Motor
CC-0097	CCW system (open)	Butterfly	3	Motor
CC-0098	CCW system (open)	Butterfly	3	Motor
CC-0131	CCW system (operate)	Butterfly	3	Motor
CC-0132	CCW system (operate)	Butterfly	3	Motor
CC-0143	CCW system (close)	Butterfly	3	Motor
CC-0144	CCW system (close)	Butterfly	3	Motor
CC-0145	CCW system (close)	Butterfly	3	Motor
CC-0146	CCW system (close)	Butterfly	3	Motor
CC-0147	CCW system (close)	Butterfly	3	Motor
CC-0148	CCW system (close)	Butterfly	3	Motor
CC-0150	CCW system (close)	Butterfly	3	Motor
CC-0181	CCW system (open)	Butterfly	3	Motor
CC-0182	CCW system (open)	Butterfly	3	Motor
CC-0231	Containment isolation (operate)	Butterfly	2	Motor
CC-0249	Containment isolation (operate)	Butterfly	2	Motor
CC-0250	Containment isolation (operate)	Butterfly	2	Motor
CC-0296	Containment isolation (close)	Butterfly	2	Motor
CC-0297	Containment isolation (close)	Butterfly	2	Motor
CC-0301	Containment isolation (close)	Butterfly	2	Motor
CC-0302	Containment isolation (close)	Butterfly	2	Motor
CC-0351	CCW system (close)	Butterfly	3	Motor
CC-0352	CCW system (close)	Butterfly	3	Motor

Table 3.9-4 (18 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
CC-0383	CCW system (operate)	Butterfly	3	Motor
CC-0384	CCW system (operate)	Butterfly	3	Motor
CC-0389	CCW system (Open)	Butterfly	3	Motor
CC-0390	CCW system (Open)	Butterfly	3	Motor
CC-0901	CCW flow control (operate)	3-way	3	Pneumatic
CC-0902	CCW flow control (operate)	3-way	3	Pneumatic
CC-0905	CCW flow control (operate)	3-way	3	Pneumatic
CC-0906	CCW flow control (operate)	3-way	3	Pneumatic
CC-1001	CCW system (operate)	Check	3	None
CC-1002	CCW system (operate)	Check	3	None
CC-1003	CCW system (operate)	Check	3	None
CC-1004	CCW system (operate)	Check	3	None
CC-1099	Containment isolation (operate)	Check	2	None
CC-1100	Containment isolation (operate)	Check	2	None
CC-1107	CCW system (operate)	Relief	3	None
CC-1108	CCW system (operate)	Relief	3	None
CC-1109	CCW system (operate)	Check	3	None
CC-1110	CCW system (operate)	Check	3	None
CC-1111	CCW system (operate)	Relief	3	None
CC-1112	CCW system (operate)	Relief	3	None
CC-1303	CCW system (operate)	Check	3	None
CC-1304	CCW system (operate)	Check	3	None
CC-1309	CCW system (operate)	Check	3	None
CC-1310	CCW system (operate)	Check	3	None
CC-1317	CCW system (operate)	Check	3	None
CC-1318	CCW system (operate)	Check	3	None
CC-1319	CCW system (operate)	Check	3	None
CC-1320	CCW system (operate)	Check	3	None
CC-1685	Containment isolation (operate)	Check	2	None
CC-1686	Containment isolation (operate)	Check	2	None
CC-1325	CCW system (operate)	Check	3	None
CC-1326	CCW system (operate)	Check	3	None
SX-1003	ESW system (operate)	Check	3	None

Table 3.9-4 (19 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
SX-1004	ESW system (operate)	Check	3	None
FC-1005	SFP cooling system (operate)	Check	3	None
FC-1006	SFP cooling system (operate)	Check	3	None
FC-1145	Containment isolation (close)	Check	2	None
DE-0005	Containment isolation (close)	Globe	2	Motor
DE-0006	Containment isolation (close)	Globe	2	Pneumatic
PS-0031	SG 1 blowdown hot leg isolation (close)	Gate	2	Pneumatic
PS-0032	SG 2 blowdown hot leg isolation (close)	Gate	2	Pneumatic
PS-0033	SG 1 downcomer isolation (close)	Gate	2	Pneumatic
PS-0034	SG 2 downcomer isolation (close)	Gate	2	Pneumatic
PS-0035	SG 1 blowdown cold leg isolation (close)	Gate	2	Pneumatic
PS-0036	SG 2 blowdown cold leg isolation (close)	Gate	2	Pneumatic
PS-0257	SG 1 blowdown hot leg isolation (close)	Gate	2	Pneumatic
PS-0258	SG 2 blowdown hot leg isolation (close)	Gate	2	Pneumatic
PR-0431	Containment radiation monitoring system (closed)	Gate	2	Motor
PR-0432	Containment radiation monitoring system (closed)	Gate	2	Motor
PR-0434	Containment radiation monitoring system (closed)	Gate	2	Motor
PR-1433	Containment radiation monitoring system (closed)	Check	2	None
CM-001	Containment hydrogen sampling (closed)	Globe	2	Solenoid
CM-002	Containment hydrogen sampling (closed)	Globe	2	Solenoid
SX-1003	ESW system (operate)	Check	3	None
SX-1004	ESW system (operate)	Check	3	None
FC-1005	SFP cooling system (operate)	Check	3	None
FC-1006	SFP cooling system (operate)	Check	3	None
FC-1145	SFP cooling system (operate)	Check	3	None
DE-0005	Containment isolation (close)	Globe	2	Motor
DE-0006	Containment isolation (close)	Globe	2	Pneumatic
PS-0031	SG 1 blowdown hot leg isolation (close)	Gate	2	Pneumatic
PS-0032	SG 2 blowdown hot leg isolation (close)	Gate	2	Pneumatic

Table 3.9-4 (20 of 20)

Valve No.	System Name (Safety Function) (1) (2) (3)	Valve Type	ASME Section III Class	Actuator Type
PS-0033	SG 1 downcomer isolation (close)	Gate	2	Pneumatic
PS-0034	SG 2 downcomer isolation (close)	Gate	2	Pneumatic
PS-0035	SG 1 blowdown cold leg isolation (close)	Gate	2	Pneumatic
PS-0036	SG 2 blowdown cold leg isolation (close)	Gate	2	Pneumatic
PS-0257	SG 1 blowdown hot leg isolation (close)	Gate	2	Pneumatic
PS-0258	SG 2 blowdown hot leg isolation (close)	Gate	2	Pneumatic
CM-003	Containment hydrogen sampling (closed)	Globe	2	Solenoid
CM-004	Containment hydrogen sampling (closed)	Globe	2	Solenoid
CM-009	Containment hydrogen sampling (closed)	Globe	2	Solenoid
CM-010	Containment hydrogen sampling (closed)	Globe	2	Solenoid
CM-011	Containment refueling water storage tank hydrogen sampling (closed)	Globe	2	Solenoid
CM-0012	Containment refueling water storage tank hydrogen sampling (closed)	Globe	2	Solenoid
CM-0013	Containment refueling water storage tank hydrogen sampling (closed)	Globe	2	Solenoid
CM-0014	Containment refueling water storage tank hydrogen sampling (closed)	Globe	2	Solenoid
CM-0017	Containment pressure monitoring system (operate)	Globe	2	Solenoid
CM-0018	Containment pressure monitoring system (operate)	Globe	2	Solenoid
CM-0019	Containment pressure monitoring system (operate)	Globe	2	Solenoid
CM-0020	Containment pressure monitoring system (operate)	Globe	2	Solenoid
CM-0021	Containment pressure monitoring system (operate)	Globe	2	Solenoid
CM-0022	Containment pressure monitoring system (operate)	Globe	2	Solenoid
CM-0023	Containment hydrogen sampling (closed)	Globe	2	Solenoid
CM-0024	Containment hydrogen sampling (closed)	Globe	2	Solenoid
CM-1013	Containment hydrogen sampling (closed)	Check	2	None
CM-1014	Containment hydrogen sampling (closed)	Check	2	None

(1) “Operate” is defined as valve being capable of both opening and closing.

(2) “Close” is defined as valve being capable of moving to or maintaining a closed position.

(3) “Open” is defined as valve being capable of moving to or maintaining an open position.

Table 3.9-5

Stress Criteria for Safety-Related
ASME Section III Class 2 and Class 3 Vessels

Service Level	Stress Limits[1]
Design and Level A	$\sigma_m \leq 1.0\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.5\ S$
Level B	$\sigma_m \leq 1.1\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.65\ S$
Level C	$\sigma_m \leq 1.5\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 1.80\ S$
Level D	$\sigma_m \leq 2.0\ S$ $(\sigma_m \text{ or } \sigma_L) + \sigma_b \leq 2.4\ S$

(1) Stress limits are taken from ASME Section III, NC/ND, Table 3321-1.

Table 3.9-6

Stress Criteria for ASME Section III Class 2 and Class 3 Inactive Pumps

Plant Condition	Service Limits[1]	Loads	Stress Limits[2]	P_{max}[3]	Subsections[5]
Design	Design	Sustained loads: pressure, weight, other mechanical loads	$\sigma_m \leq 1.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5$ S	—	ASME Section III NC/ND-3400
Normal	Level A	Sustained loads: pressure, weight, other mechanical loads	$\sigma_m \leq 1.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5$ S	1.0	ASME Section III NC/ND-3400
Upset	Level B	Occupational loads: pressure, weight, thermal effects, dynamic fluid loads,[4] wind[6]	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	1.1	ASME Section III NC/ND-3400
Emergency	Level C	Occupational loads: pressure, weight, thermal effects, dynamic fluid loads,[4] tornado[6]	$\sigma_m \leq 1.5$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.8$ S	1.2	ASME Section III NC/ND-3400
Faulted	Level D	Occupational loads: pressure, weight, thermal effects, dynamic fluid loads,[4] SSE inertia, pipe break	$\sigma_m \leq 2.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 2.4$ S	1.5	ASME Section III NC/ND-3400

(1) Service limits are taken from ASME Section III, NCA-2142.4.

(2) Stress limits are taken from ASME Section III, Subsections NC and ND, Table NC/ND-3416-1.

(3) The maximum pressure does not exceed the tabulated factors listed under Pmax times the design pressure.

(4) Dynamic fluid loads (DFL) are occasional loads such as safety and relief valve thrust, steam hammer, water hammer, or other loads associated with plant upset or faulted condition as applicable. Dynamic loads are combined by the SRSS method.

(5) SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, Paragraph 9, “Elimination of Operating Basis Earthquake,” Nuclear Regulatory Commission, July 21, 1993.

(6) Wind and tornado loads are not combined with earthquake loading.

Table 3.9-7

Stress Criteria for ASME Section III Class 2 and Class 3 Active Pumps

Plant Condition	Service Limits[1]	Loads	Stress Limits[2]	P_{max}[3]	Subsections[5]
Design	Design	Sustained loads: pressure, weight, other mechanical loads	$\sigma_m \leq 1.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5$ S	—	ASME Section III NC/ND-3400
Normal	Level A	Sustained loads: pressure, weight, other mechanical loads	$\sigma_m \leq 1.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5$ S	1.0	ASME Section III NC/ND-3400
Upset	Level B	Occupational loads: pressure, weight, thermal effects, dynamic fluid loads,[4] wind[6]	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	1.1	ASME Section III NC/ND-3400
Emergency	Level B	Occupational loads: pressure, weight, thermal effects, dynamic fluid loads,[4] tornado[6]	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	1.1	ASME Section III NC/ND-3400
Faulted	Level B	Occupational loads: pressure, weight, thermal effects, dynamic fluid loads,[4] SSE inertia, pipe break	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	1.1	ASME Section III NC/ND-3400

(1) Service limits are taken from ASME Section III NCA-2142.4.

(2) Stress limits are taken from ASME Section III, Subsections NC and ND, Table NC/ND-3416-1. However, the stress limits for service level C and D are more restrictive than the ASME Section III limits to provide reasonable assurance of pump operability.

(3) The maximum pressure does not exceed the tabulated factors listed under Pmax times the design pressure.

(4) Dynamic fluid loads (DFL) are occasional loads such as safety and relief valve thrust, steam hammer, water hammer, or other loads associated with plant upset or faulted condition as applicable. Dynamic loads are combined by the SRSS method.

(5) SECY-93-087, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, Paragraph 9, “Elimination of Operating Basis Earthquake,” July 21, 1993.

(6) Wind and tornado loads are not combined with earthquake loading.

Table 3.9-8

Stress Criteria for Safety-Related ASME Section III
Class 2 and Class 3 Non-active Valves

Service Level	Stress Limits(1)–(4)(6)	P_{max}(5)
Design and Level A	$\sigma_m \leq 1.0$ S (σ_m or σ_L) + $\sigma_b \leq 1.5$ S	1.0
Level B	$\sigma_m \leq 1.1$ S (σ_m or σ_L) + $\sigma_b \leq 1.65$ S	1.1
Level C	$\sigma_m \leq 1.5$ S (σ_m or σ_L) + $\sigma_b \leq 1.8$ S	1.2
Level D	$\sigma_m \leq 2.0$ S (σ_m or σ_L) + $\sigma_b \leq 2.4$ S	1.5

(1) Valve nozzle (piping load) stress analysis is not required when both of the following conditions are satisfied:

a) The section modulus and areas of every plane, normal to the flow, through the region defined as the valve body crotch are at least 110 percent of those for the piping connected (or joined) to the valve body inlet and outlet nozzles.

b) Code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (1) above is multiplied by the ratio of Spipe/Svalve. If unable to comply with this requirement, the design by analysis procedure of the ASME Section III, NB-3545.2 is an acceptable alternate method.

(2) Casting quality factor of 1.0 is used.

(3) These stress limits are applicable to the pressure retaining boundary, and include the effects of loads transmitted by the extended structures, when applicable.

(4) Design requirements listed in this table are not applicable to valve stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet.

(5) The maximum pressure resulting from Service Levels B, C, or D does not exceed the tabulated factors listed under Pmax times the design pressure or the rated pressure at the applicable operating condition temperature. If the pressure rating limits are met at the operating conditions, the stress limits in this table are considered to be satisfied.

(6) Stress limits are taken from ASME Section III, Table NC/ND-3521-1.

Table 3.9-9

Stress Criteria for ASME Section III Class 2 and Class 3 Active Valves

Service Loading Conditions (Plant Condition)	Service Limits[1]	Stress Limits[2]	P_{max}[3]
Design	Design	$\sigma_m \leq 1.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5$ S	—
Level A (normal)	Level A	$\sigma_m \leq 1.0$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.5$ S	1.0
Level B (upset)	Level B	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	1.1
Level C (emergency)	Level B	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	1.1
Level D (faulted)	Level B	$\sigma_m \leq 1.1$ S $(\sigma_m$ or $\sigma_L) + \sigma_b \leq 1.65$ S	1.1

(1) Service Limits (Level A, B, C, and D) are defined in ASME Section III, NCA-2142.4.

(2) Stress limits are in accordance with ASME Section III, Table NC/ND-3521-1. For service loading conditions level C and level D, the stress limits specified for active valves are more restrictive than ASME Section III limits to assure the operability of valves.

(3) The maximum pressure does not exceed the tabulated factors listed under P_{max} times the design pressure or the rated pressure at the applicable operating condition temperature.

(4) Valve nozzle (piping load) stress analysis is not required when both of the following conditions are satisfied:

a) The section modulus and areas of every plane, normal to the flow, through the region defined as the valve body crotch are at least 110 percent of those for the piping connected (or joined) to the valve body inlet and outlet nozzles.

b) Code allowable stress, S, for valve body material is equal to or greater than the code allowable stress, S, of connected piping material. If the valve body material allowable stress is less than that of the connected piping, the valve section modulus and area as calculated in (1) above is multiplied by the ratio of S_{pipe}/S_{valve}. If unable to comply with this requirement, the design by analysis procedure of ASME Section III, NB-3545.2 is an acceptable alternate method.

(5) Casting quality factor of 1.0 is used.

(6) Design requirements listed in this table are not applicable to valve stems, seat rings, or other parts of valves which are contained within the confines of the body and bonnet.

Table 3.9-10

Loading Conditions and Load Combinations Requirements
for ASME Section III Class 1, 2, and 3 Piping Supports

Service Level	Loading Combination
Level A	Weight Thermal$^{(1)}$ Friction
Level B	Weight Thermal$^{(1)}$ Dynamic fluid loads$^{(2)}$ Wind
Level C	Weight Thermal$^{(1)}$ Dynamic fluid loads$^{(2)}$ Tornado
Level D	Weight Thermal$^{(1)}$ Dynamic fluid loads$^{(2)}$ SSE inertia SSE seismic movements Pipe break loads

(1) Thermal conditions (including ambient temperature) to be combined to provide maximum load combinations.

(2) Dynamic Fluid Loads due to safety/relief valve thrust, steam hammer, and water hammer.

Table 3.9-11

Stress Limits for CEDM Pressure Housings

Service Level	Stress Categories and Limits of Stress Intensities[1]
1. Design: design pressure, normal operating loads[2], IRWST loads	NB-3221 and Figure NB-3221-1 including notes
2. Level A: normal operating loads, normal operating transients	NB-3222 and Figure NB-3222-1 including notes
3. Level B: normal operating loads, upset transients, IRWST loads, fatigue loads due to SSE[3]	NB-3223 and Figure NB-3222-1 including notes
4. Level D[4]: operating pressure, normal operating loads, IRWST loads, BLPB loads, SSE loads	Appendix F Article F-1000 Rules for evaluation of service conditions loading with level D service limits
5. Testing: testing plant transients	NB-3226

(1) References listed are taken from ASME Section III.

(2) 'Normal operating loads' is defined in Subsection 3.9.4.3

(3) Fatigue loads due to SSE are applied in accordance with Subsection 3.9.2.2.3.

(4) SSE loads is combined with BLPB and IRWST by the SRSS method in accordance with the guidelines of NUREG-0484.

Table 3.9-12

Stress Limits for Reactor Internals Design and Service Loads

Stress Limit	Description
Design Limits	The reactor internals are designed to meet the design limits defined in ASME Section III, NG-3221, for design loadings. The reactor internals are safety Class 3, seismic Category I, and Quality Class 1 in accordance with ANSI/ANS-51.1-1983. Core support structures are constructed in accordance with ASME Section III, NG-1100. The reactor internals other than core support structures meet the guidelines of ASME Section III, NG-3000 and be constructed so as not to adversely affect the integrity of the core support structures. Under Level D service loadings, the maximum stress intensity is obtained from principal stresses resulting from an SRSS combination of IRWST, BLPB and SSE plus normal operating dynamic and static loading in accordance with NUREG-0484, Rev. 1. For other than Level D service loading conditions maximum stress intensity are derived from an SRSS combination of dynamic loads in accordance with NUREG-0484, Rev. 1, or a more conservative summation of stress intensities.
Level A Service Limits	The reactor internals are designed to meet the Level A service limits defined in ASME Section III, NG-3222, for Level A service loadings.
Level B Service Limits	The reactor internals are designed to meet the Level B service limits defined in ASME Section III, NG-3223, for Level B service loadings.
Level C Service Limits	The reactor internals are designed to meet the Level C service limits defined in ASME Section III, NG-3224, for Level C service loadings.
Level D Service Limits	The reactor internals are designed to meet the Level D service limits defined in ASME Section III, NG-3225, for elastic system analysis of Appendix F of ASME Section III using Level D service loadings. Maximum stress intensity is obtained from principal stresses resulting from an SRSS combination of IRWST, BLPB, and SSE loadings plus normal operation loads in accordance with NUREG-0484, Rev. 1.

Table 3.9-13 (1 of 90)

Inservice Testing of Safety-Related Pumps and Valves

Pump	Safety Class	Test Parameter [h]	Test Freq	Test Config.[i]	Figure No.
CCW pump 1A	3	DP, SPs, SPo, Q, V	3 mo	16	9.2.2-1
CCW pump 1B	3	DP, SPs, SPo, Q, V	3 mo	16	9.2.2-1
CCW pump 2A	3	DP, SPs, SPo, Q, V	3 mo	16	9.2.2-1
CCW pump 2B	3	DP, SPs, SPo, Q, V	3 mo	16	9.2.2-1
CCW makeup pump 3A	3	DP, SPs, SPo, Q, V	3 mo	22	9.2.2-1
CCW makeup pump 3B	3	DP, SPs, SPo, Q, V	3 mo	22	9.2.2-1
SI pump 1	2	DP, SPs, SPo, Q, V (40)	3 mo	18	6.3.2-1
SI pump 2	2	DP, SPs, SPo, Q, V (40)	3 mo	18	6.3.2-1
SI pump 3	2	DP, SPs, SPo, Q, V (40)	3 mo	18	6.3.2-1
SI pump 4	2	DP, SPs, SPo, Q, V (40)	3 mo	18	6.3.2-1
SC pump 1	2	DP, SPs, SPo, Q, V	3 mo	19	6.3.2-1
SC pump 2	2	DP, SPs, SPo, Q, V	3 mo	19	6.3.2-1
CS pump 3	2	DP, SPs, SPo, Q, V	3 mo	19	6.2.2-1
CS pump 4	2	DP, SPs, SPo, Q, V	3 mo	19	6.2.2-1
ESW pump 1A	3	DP, SPs, SPo, Q, V	3 mo	17	9.2.1-1
ESW pump 1B	3	DP, SPs, SPo, Q, V	3 mo	17	9.2.1-1
ESW pump 2A	3	DP, SPs, SPo, Q, V	3 mo	17	9.2.1-1
ESW pump 2B	3	DP, SPs, SPo, Q, V	3 mo	17	9.2.1-1
SFP cooling pump 1	3	DP, SPs, SPo, Q, V	3 mo	20	9.1.3-1
SFP cooling pump 2	3	DP, SPs, SPo, Q, V	3 mo	30	9.1.3-1
MD AFW pump PP02A	3	DP, SPs, SPo, Q, V	3 mo	21	10.4.9-1
TD AFW pump PP01A	3	N, DP, SPs, SPo, Q, V	3 mo	21	10.4.9-1
MD AFW pump PP02B	3	DP, SPs, SPo, Q, V	3 mo	21	10.4.9-1
TD AFW pump PP01B	3	N, DP, SPs, SPo, Q, V	3 mo	21	10.4.9-1
ECW pump PP01A	3	DP, SP_S, SP_O, Q, V	3 mo	20	9.2.7-1
ECW pump PP02A	3	DP, SP_S, SP_O, Q, V	3 mo	20	9.2.7-1
ECW makeup pump PP03A	3	DP, SPS, SPO, Q, V	3 mo	20	9.2.7-1

Table 3.9-13 (2 of 90)

Pump	Safety Class	Test Parameter [h]	Test Freq	Test Config.[i]	Figure No.
ECW pump PP01B	3	DP, SP_S, SP_O, Q, V	3 mo	20	9.2.7-1
ECW pump PP02B	3	DP, SP_S, SP_O, Q, V	3 mo	20	9.2.7-1
ECW makeup pump PP03B	3	DP, SP_S, SP_O, Q, V	3 mo	20	9.2.7-1
DG 1 fuel oil transfer pump 01A/02A	3	Note 37	Note 37	Note 37	9.5.4-1
DG 2 fuel oil transfer pump 01B/02B	3	Note 37	Note 37	Note 37	9.5.4-1
DG 3 fuel oil transfer pump 01C/02C	3	Note 37	Note 37	Note 37	9.5.4-1
DG 4 fuel oil transfer pump 01D/02D	3	Note 37	Note 37	Note 37	9.5.4-1
DG 1 motor-driven fuel oil feed pump	3	Note 37	Note 37	Note 37	9.5.4-1
DG 2 motor-driven fuel oil feed pump	3	Note 37	Note 37	Note 37	9.5.4-1
DG 3 motor-driven fuel oil feed pump	3	Note 37	Note 37	Note 37	9.5.4-1
DG 4 motor-driven fuel oil feed pump	3	Note 37	Note 37	Note 37	9.5.4-1
DG 1 preheating water pump	3	Note 37	Note 37	Note 37	9.5.5-1
DG 2 preheating water pump	3	Note 37	Note 37	Note 37	9.5.5-1
DG 3 preheating water pump	3	Note 37	Note 37	Note 37	9.5.5-1
DG 4 preheating water pump	3	Note 37	Note 37	Note 37	9.5.5-1

Table 3.9-13 (3 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
RC-200	PZR POSRV	POS	SA EL	1	A/C		RVT	RO
RC-201	PZR POSRV	POS	SA EL	1	A/C		RVT	RO
RC-202	PZR POSRV	POS	SA EL	1	A/C		RVT	RO
RC-203	PRZ POSRV	POS	SA EL	1	A/C		RVT	RO
RC-0385	POSRV relief line vent	3W	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
RC-0386	POSRV relief line vent	3W	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
RG-0410	PZR gas vent	GL	S	1	B		S MT FS LPV	CS (12) CS (12) CS (12) 2 yr
RG-0411	PZR gas vent	GL	S	1	B		S MT FS LPV	CS (12) CS (12) CS (12) 2 yr
RG-0412	PZR gas vent	GL	S	1	B		S MT FS LPV	CS (12) CS (12) CS (12) 2 yr

Table 3.9-13 (4 of 90)

Valve No.	Valve Description	Valve Type [(a)]	Valve Act [(b)]	Safety Class [(c)]	Code Cat [(d)]	Valve Funct [(e)]	Test Reqd [(f)]	Test Freq [(g)]
RG-0413	PZR gas vent	GL	S	1	B		S MT FS LPV	CS [(12)] CS [(12)] CS [(12)] 2 yr
RG-0414	Reactor vessel gas vent	GL	S	1	B		S MT FS LPV	CS [(12)] CS [(12)] CS [(12)] 2 yr
RG-0415	Reactor vessel gas vent	GL	S	1	B		S MT FS LPV	CS [(12)] CS [(12)] CS [(12)] 2 yr
RG-0416	Reactor vessel gas vent	GL	S	1	B		S MT FS LPV	CS [(12)] CS [(12)] CS [(12)] 2 yr
RG-0417	Reactor vessel gas vent	GL	S	1	B		S MT FS LPV	CS [(12)] CS [(12)] CS [(12)] 2 yr
RG-0418	RCGV discharge to reactor drain tank vent	GL	S	2	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr
RG-0419	RCGV discharge to IRWST vent	GL	S	2	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr

Table 3.9-13 (5 of 90)

Valve No.	Valve Description	Valve Type [a]	Valve Act [b]	Safety Class [c]	Code Cat [d]	Valve Funct [e]	Test Reqd [f]	Test Freq [g]
RG-0420	RCGV discharge to IRWST vent	GL	S	2	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr
SI-100	IRWST return line check	CK	SA	2	C	CIC	S RF LT	3 mo 3 mo 2 yr
SI-101	IRWST return line check	CK	SA	2	C	CIC	S RF LT	3 mo 3 mo 2 yr
SI-113	SI line check	CK	SA	2	C	CIN	S RF	RO (16) RO (16)
SI-123	SI line check	CK	SA	2	C	CIN	S RF	CS (17) RO (17)
SI-133	SI line check	CK	SA	2	C	CIN	S RF	RO (16) RO (16)
SI-143	SI line check	CK	SA	2	C	CIN	S RF	CS (17) RO (17)
SI-157	CS pump suction check	CK	SA	2	C		S RF	3 mo 3 mo
SI-158	CS pump suction check	CK	SA	2	C		S RF	3 mo 3 mo

Table 3.9-13 (6 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-159	SC pump suction check	CK	SA	2	C		S RF	3 mo 3 mo
SI-160	SC pump suction check	CK	SA	2	C		S RF	3 mo 3 mo
SI-166	SI hot leg injection line relief	RV	SA	2	C		RVT	10 yr
SI-168	SCS line check	CK	SA	2	C		S RF	CS (17) 3 mo
SI-169	SCS line relief to RDT	RV	SA	1	C		RVT	5 yr
SI-178	SCS line check	CK	SA	2	C		S RF	CS (17) 3 mo
SI-179	SCS suction line relief	RV	SA	2	A/C	CIN	RVT	10 yr
SI-187	SCS test return line relief	RV	SA	2	C		RVT	10 yr
SI-188	SCS test return line relief	RV	SA	2	C		RVT	10 yr
SI-189	SCS suction line relief	RV	SA	2	A/C	CIN	RVT	10 yr
SI-211	SIT relief	RV	SA	2	C		RVT	10 yr
SI-215	SIT check	CK	SA	1	A/C	PIV	S LT RF	RR (19) 2 yr (38)
SI-217	SI line check	CK	SA	1	A/C	PIV	S LT RF	RR (20) 2 yr (38)
SI-221	SIT relief	RV	SA	2	C		RVT	10 yr
SI-225	SIT check	CK	SA	1	A/C	PIV	S LT RF	RR (19) 2 yr (38)

Table 3.9-13 (7 of 90)

Valve No.	Valve Description	Valve Type [a]	Valve Act [b]	Safety Class [c]	Code Cat [d]	Valve Funct [e]	Test Reqd [f]	Test Freq [g]
SI-227	SI line check	CK	SA	1	A/C	PIV	S LT RF	CS [20] 2 yr [38]
SI-231	SIT relief	RV	SA	2	C		RVT	10 yr
SI-235	SIT check	CK	SA	1	A/C	PIV	S LT RF	RR [19] 2 yr [38]
SI-237	SI line check	CK	SA	1	A/C	PIV	S LT RF	RR [20] 2 yr [38]
SI-241	SIT relief	RV	SA	2	C		RVT	10 yr
SI-245	SIT check	CK	SA	1	A/C	PIV	S LT RF	RR [19] 2 yr [38]
SI-247	SI line check	CK	SA	1	A/C	PIV	S LT RF	CS [20] 2 yr [38]
SI-285	SI miniflow line relief	RV	SA	2	C		RVT	10 yr
SI-286	SI miniflow line relief	RV	SA	2	C		RVT	10 yr
SI-287	SCS test return line relief	RV	SA	2	C		RVT	10 yr
SI-289	SCS test return line relief	RV	SA	2	C		RVT	10 yr
SI-292	SIT fill return line relief	RV	SA	3	C		RVT	10 yr
SI-293	SIT fill line isolation	GL	M	2	A	P, CIC	LT	2 yr

Table 3.9-13 (8 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-300	IRWST return line isolation	GT	EL	2	B	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
SI-301	IRWST return line isolation	GT	EL	2	B	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
SI-302	SI combined miniflow line isolation	GT	EL	2	B	CIC	S MT LPV LT	RO (21) RO (21) 2 yr 2 yr
SI-303	SI combined miniflow line isolation	GL	EL	2	B	CIC	S MT LPV LT	RO (21) RO (21) 2 yr 2 yr
SI-304	IRWST isolation	GT	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-305	IRWST isolation	GT	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-308	IRWST isolation	GT	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr

Table 3.9-13 (9 of 90)

Valve No.	Valve Description	Valve Type [(a)]	Valve Act [(b)]	Safety Class [(c)]	Code Cat [(d)]	Valve Funct [(e)]	Test Reqd [(f)]	Test Freq [(g)]
SI-309	IRWST isolation	GT	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-310	SDCHX outlet flow isolation	GL	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-311	SDCHX outlet flow isolation	GL	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-312	SDCHX bypass flow control	GL	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-313	SDCHX bypass flow control	GL	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-314	SCS test return line isolation	GL	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-315	SCS test return line isolation	GL	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-321	SI hot leg injection line isolation	GL	EL	2	A	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-322	Hot leg check valve leakage isolation	GL	AD	1	A	PIV	S MT FS LPV LT	3 mo 3 mo RO [(36)] 2 yr 2 yr

Table 3.9-13 (10 of 90)

Valve No.	Valve Description	Valve Type [(a)]	Valve Act [(b)]	Safety Class [(c)]	Code Cat [(d)]	Valve Funct [(e)]	Test Reqd [(f)]	Test Freq [(g)]
SI-331	SI hot leg injection line isolation	GL	EL	2	A	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-332	Hot leg check valve leakage isolation	GL	AD	1	A	PIV	S MT FS LPV LT	3 mo 3 mo RO [(36)] 2 yr 2 yr
SI-340	SCS/CSS pump suction cross connect	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-341	SCS/CSS pump discharge cross connect	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-342	SCS/CSS pump suction cross connect	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-343	SCS/CSS pump discharge cross connect	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-344	SC pump suction isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-346	SC pump suction isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr

Table 3.9-13 (11 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-347	CS pump suction isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI 391	Reactor cavity isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI 393	Reactor cavity isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI 395	IRWST return line isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-348	CS pump suction isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-404	SI pump discharge check	CK	SA	2	C		S RF	RO (16) 3 mo (16A)
SI-405	SI pump discharge check	CK	SA	2	C		S RF	RO (16) 3 mo (16A)
SI-409	SI line relief	RV	SA	2	C		RVT	10 yr
SI-417	SI line relief	RV	SA	2	C		RVT	10 yr
SI 422	SDCHX header relief	RV	SA	2	C		RVT	10 yr
SI 423	SDCHX header relief	RV	SA	2	C		RVT	10 yr

Table 3.9-13 (12 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-424	SI miniflow check	CK	SA	2	C		S RF	3 mo (22) 3 mo (22)
SI-426	SI miniflow check	CK	SA	2	C		S RF	3 mo 3 mo (22)
SI-434	SI pump discharge check	CK	SA	2	C		S RF	RO (16) 3 mo (16A)
SI-439	SI line relief	RV	SA	2	C		RVT	10 yr
SI-446	SI pump discharge check	CK	SA	2	C		S RF	RO (16) 3 mo (16A)
SI-448	SI miniflow check	CK	SA	2	C		S RF	3 mo 3 mo (16A)
SI-449	SI line relief	RV	SA	2	C		RVT	10 yr
SI-451	SI miniflow check	CK	SA	2	C		S RF	3 mo 3 mo (22)
SI 461	SC line relief	RV	SA	2	C		RVT	10 yr
SI 462	SC line relief	RV	SA	2	C		RVT	10 yr
SI-466	SC line relief	RV	SA	2	C		RVT	10 yr
SI-467	SC line relief	RV	SA	2	C		RVT	10 yr
SI-468	SI hot leg injection line relief	RV	SA	2	C		RVT	10 yr

Table 3.9-13 (13 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-469	SCS line relief to RDT	RV	SA	1	C		RVT	5 yr
SI-473	check valve leakage line relief	RV	SA	2	C		RVT	10 yr
SI-474	SIT fill line relief	RV	SA	2	A/C	CIC	RVT LT	10 yr 2 yr
SI-522	SI hot leg injection line check	CK	SA	1	A/C	PIV	S LT RF	RO (16) 2 yr (38)
SI-523	SI hot leg injection line check	CK	SA	1	A/C	CIN, PIV	S LT RF	RO (16) 2 yr (38)
SI-532	SI hot leg injection line check	CK	SA	1	A/C	PIV	S LT RF	RO (16) 2 yr (38)
SI-533	SI hot leg injection line check	CK	SA	1	A/C	CIN, PIV	S LT RF	RO (16) 2 yr (38)
SI-540	SI line check	CK	SA	1	A/C	PIV	S LT RF	RO (16) 2 yr (38)
SI-541	SI line check	CK	SA	1	A/C	PIV	S LT RF	CS (17) 2 yr
SI-542	SI line check	CK	SA	1	A/C	PIV	S LT RF	RO (16) 2 yr (38)

Table 3.9-13 (14 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-543	SI line check	CK	SA	1	A/C	PIV	S LT RF	CS (17) 2 yr (38)
SI-568	SC pump discharge check	CK	SA	2	C		S RF	3 mo 3 mo (15)
SI-569	SC pump discharge check	CK	SA	2	C		S RF	3 mo 3 mo (15)
SI-600	SCS line isolation	GL	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-601	SCS line isolation	GL	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-602	SI low flow control	GL	EL	2	B	CIN	S. MT LPV	3 mo 3 mo 2 yr
SI-603	SI low flow control	GL	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-604	SI hot leg injection isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr

Table 3.9-13 (15 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-605	SIT atmospheric vent isolation	GL	S	2	B		S MT FS LPV	CS (31) CS (31) CS (31) 2 yr
SI-606	SIT atmospheric vent isolation	GL	S	2	B		S MT FS LPV	CS (31) CS (31) CS (31) 2 yr
SI-607	SIT atmospheric vent isolation	GL	S	2	B		S MT FS LPV	CS (31) CS (31) CS (31) 2 yr
SI-608	SIT atmospheric vent isolation	GL	S	2	B		S MT FS LPV	CS (31) CS (31) CS (31) 2 yr
SI-609	SI hot leg isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-611	SIT fill and drain isolation	GL	AD	2	B		S MT FS LPV	3 mo 3 mo RO (36) 2 yr
SI-612	SIT nitrogen supply isolation	GL	AD	2	B		S FS LPV	EI (35) RO (35) 2 yr

Table 3.9-13 (16 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-613	SIT atmospheric vent isolation	GL	S	2	B		S MT FS LPV	CS (31) CS (31) CS (31) 2 yr
SI-614	SIT discharge isolation	GT	EL	1	B		S MT LPV	CS (24) CS (24) 2 yr
SI-616	SI line isolation	GL	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-618	Check valve leakage isolation	GL	AD	1	A	PIV	S MT FS LPV. LT	3 mo 3 mo RO (36) 2 yr 2 yr
SI 619	SIT nitrogen supply isolation	GL	AD	2	B		S FS LPV	EI (35) RO (35) 2 yr
SI-621	SIT fill and drain isolation	GL	AD	2	B		S MT FS LPV	3 mo 3 mo RO (36) 2 yr
SI-622	SIT nitrogen supply isolation	GL	AD	2	B		S FS LPV	EI (35) RO (35) 2 yr

Table 3.9-13 (17 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-623	SIT atmospheric vent isolation	GL	S	2	B		S MT FS LPV	CS (31) CS (31) CS (31) 2 yr
SI-624	SIT discharge isolation	GT	EL	1	B		S MT LPV	CS (24) CS (24) 2 yr
SI-626	SI line isolation	GL	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-628	Check valve leakage isolation	GL	AD	1	A	PIV	S MT FS LPV LT	3 mo 3 mo RO (36) 2 yr 2 yr
SI-629	SIT nitrogen supply isolation	GL	AD	2	B		S FS LPV	EI (35) RO (35) 2 yr
SI-631	SIT fill and drain isolation	GL	AD	2	B		S MT FS LPV	3 mo 3 mo RO(36) 2 yr
SI-632	SIT nitrogen supply isolation	GL	AD	2	B		S FS LPV	EI (35) RO (35) 2 yr

Table 3.9-13 (18 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-633	SIT atmospheric vent isolation	GL	S	2	B		S MT FS LPV	CS (31) CS (31) CS (31) 2 yr
SI-634	SIT discharge isolation	GT	EL	1	B		S MT LPV	CS (24) CS (24) 2 yr
SI-636	SI line isolation	GL	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-638	Check valve leakage isolation	GL	AD	1	A	PIV	S MT FS LPV LT	3 mo 3 mo RO (36) 2 yr 2 yr
SI-639	SIT nitrogen supply isolation	GL	AD	2	B		S FS LPV	EI (35) RO (35) 2 yr
SI-641	SIT fill and drain isolation	GL	AD	2	B		S MT FS LPV	3 mo 3 mo RO (36) 2 yr
SI-642	SIT nitrogen supply isolation	GL	AD	2	B		S FS LPV	EI (35) RO (35) 2 yr

Table 3.9-13 (19 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-643	SIT atmospheric vent isolation	GL	S	2	B		S MT FS LPV	CS (31) CS (31) CS (31) 2 yr
SI-644	SIT discharge isolation	GL	EL	1	B		S MT LPV	CS (24) CS (24) 2 yr
SI-646	SI line isolation	GL	EL	2	B	CIN	S MT LPV	3 mo 3 mo 2 yr
SI-648	Check valve leakage isolation	GL	AD	1	A	PIV	S MT FS LPV LT	3 mo 3 mo RO (36) 2 yr 2 yr
SI-649	SIT nitrogen supply isolation	GL	AD	2	B		S FS LPV	EI (35) RO (35) 2 yr
SI-651	SCS suction line isolation	GT	EL	1	A	PIV	S MT LPV LT	CS (25) CS (25) 2 yr 2 yr
SI-652	SCS suction line isolation	GT	EL	1	A	PIV	S MT LPV LT	CS (25) CS (25) 2 yr 2 yr

Table 3.9-13 (20 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-653	SCS suction line isolation	GT	EL	1	A	PIV, CIN	S MT LPV	CS (25) CS (25) 2 yr
SI-654	SCS suction line isolation	GT	EL	1	A	PIV, CIN	S MT LPV LT	CS (25) CS (25) 2 yr 2 yr
SI-655	SCS suction line isolation	GT	EL	2	B	CIN	S MT LPV	CS (25) CS (25) 2 yr
SI-656	SCS suction line isolation	GT	EL	2	B	CIN	S MT LPV	CS (25) CS (25) 2 yr
SI-661	RDT isolation	GL	AD	2	B		S FS LPV	3 mo RO (36) 2 yr
SI-670	SIT drain line isolation	GL	AD	2	B		S FS LPV	3 mo RO (36) 2 yr
SI-682	SIT fill line isolation	GL	AD	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr

Table 3.9-13 (21 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI-688	SCS test return line isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-690	SCS warmup line flow control	GL	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-691	SCS warmup line flow control	GL	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI-693	SCS test return line isolation	GT	EL	2	B		S MT LPV	3 mo 3 mo 2 yr
SI 704	SIS fill check	CK	SA	2	C		S RF	3 mo 3 mo
SI 705	SIS fill check	CK	SA	2	C		S RF	3 mo 3 mo
SI 706	SIS fill check	CK	SA	2	C		S RF	3 mo 3 mo
SI 707	SIS fill check	CK	SA	2	C		S RF	3 mo 3 mo
SI 712	SCS fill check	CK	SA	2	C		S RF	3 mo 3 mo
SI 713	SCS fill check	CK	SA	2	C		S RF	3 mo 3 mo
SI 801	External emergency injection line chek	CK	SA	2	C		S RF	3 mo 3 mo

Table 3.9-13 (22 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SI 803	External emergency injection line isolation	GT	M	2	B		S FS	3 mo 3 mo
CS-0001	Containment spray containment isolation	GT	EL	2	B		S MT FS LPV	3 mo 3 mo RO (2 yr) RO (2 yr)
CS-0002	Containment spray containment isolation	GT	EL	2	B		S MT FS LPV	3 mo 3 mo RO (2 yr) RO (2 yr)
CS-0003	Containment spray containment isolation	GT	EL	2	A	CIC	LT S MT FS LPV	RO (2 yr) RO (2 yr) 3 mo RO (2 yr) RO (2 yr)
CS-0004	Containment spray containment isolation	GT	EL	2	A	CIC	LT S FS LPV	RO (2 yr) RO (2 yr) RO (2 yr) RO (2 yr)
CS-0005	Containment spray IRWST return isolation	GT	EL	2	B		S FS LPV	3 mo 2 yr 2 yr
CS-0006	Containment spray IRWST return isolation	GT	EL	2	B		S FS LPV	3 mo 2 yr 2 yr

Table 3.9-13 (23 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CS-0007	Containment spray IRWST return isolation	GT	EL	2	B		S FS LPV	3 mo 2 yr 2 yr
CS-0008	Containment spray IRWST return isolation	GT	EL	2	B		S FS LPV	3 mo 2 yr 2 yr
CS-1001	CS pump 1A check	CK	SA	2	C	P	S F	3 mo 3 mo
CS-1002	CS pump 1B check	CK	SA	2	C	P	S RF	3 mo 3 mo
CS-1005	CSHX 1A relief	RV	SA	2	C	P	RVT	RO (10 yr)
CS-1006	CSHX 1B relief	RV	SA	2	C	P	RVT	RO (10 yr)
CS-1007	Containment isolation check	CK	SA	2	A/C	CIC	S LT RF	RO (2 yr)
CS-1008	Containment isolation check	CK	SA	2	A/C	CIC	S LT RF	RO (2 yr)
CS-1011	Refueling pool isolation	GT	M	3	B	P	LPV	2 yr
CS-1012	Refueling pool isolation	GT	M	3	B	P	LPV	2 yr
CS-1013	ECSBS containment isolation	GT	M	2	A	P, CIC	LT	2 yr
CS-1014	ECSBS containment isolation check	CK	SA	2	A/C	CIC	S LT RF	RO (2 yr)
CS-1021	CS miniflow HX 2A relief	RV	SA	3	C	P	RVT	10 yr

Table 3.9-13 (24 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CS-1022	CS miniflow HX 2B relief	RV	SA	3	C	P	RVT	10 yr
CS-1023	IRWST return line relief	RV	SA	3	C	P	RVT	10 yr
CS-1024	IRWST return line relief	RV	SA	3	C	P	RVT	10 yr
IW-0001	Reactor cavity flooding isolation	GT	EL	2	B		MT S LPV	RO (34) RO (34) 2 yr
IW-0002	Reactor cavity flooding isolation	GT	EL	2	B		MT S LPV	RO (34) RO (34) 2 yr
IW-0003	Reactor cavity flooding isolation	GT	EL	2	B		MT S LPV	RO (34) RO (34) 2 yr
IW-0004	Reactor cavity flooding isolation	GT	EL	2	B		MT S LPV	RO (34) RO (34) 2 yr
IW-0010	IRWST level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0011	IRWST level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr

Table 3.9-13 (25 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
IW-0012	HVT level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0013	HVT level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0014	HVT level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0015	HVT level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0016	HVT level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0017	HVT level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr

Table 3.9-13 (26 of 90)

Valve No.	Valve Description	Valve Type [(a)]	Valve Act [(b)]	Safety Class [(c)]	Code Cat [(d)]	Valve Funct [(e)]	Test Reqd [(f)]	Test Freq [(g)]
IW-0018	Reactor cavity level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0019	Reactor cavity level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0020	Reactor cavity level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0021	Reactor cavity level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0022	IRWST level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0023	IRWST level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr

Table 3.9-13 (27 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
IW-0024	IRWST level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0025	IRWST level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0026	IRWST level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0027	IRWST level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0028	HVT level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0029	HVT level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0030	HVT level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr

Table 3.9-13 (28 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
IW-0031	HVT level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0032	Reactor cavity level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0033	Reactor cavity level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0034	Reactor cavity level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0035	Reactor cavity level instrument isolation	GL	S	2	A	CIC	LT MT FS LPV	2 yr 3 mo 3 mo 2 yr
IW-0005	BAMP suction isolation	GT	EL	2	A	CIC	LT MT S LPV	2 yr 3 mo 3 mo 2 yr
IW-0006	BAMP suction isolation	GT	EL	2	A	CIC	LT MT S LPV	2 yr 3 mo 3 mo 2 yr

Table 3.9-13 (29 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
IW-1003	BAMP suction line relief	RV	SA	2	C	PIV CIN	LT RVT	2 yr 10 yr
CV-189	IRWST makeup line check	CK	SA	2	A/C	CIC	S LT RF	3 mo 2 yr 3 mo
CV-255	Seal injection containment isolation	GL	EL	2	A	CIC	S MT LPV LT	CS (6) CS (6) 2 yr 2 yr
CV-363	Shutdown purification line check	CK	SA	2	A/C	CIC	LT RF	2 yr CS (23)
CV-362	Shutdown purification line isolation	GT	M	2	A	P, CIC	LT LPV	2 yr 2 yr
CV-494	Resin sluice supply header to reactor drain header check	CK	SA	2	A/C	CIC	S LT RF	3 mo 2 yr 3 mo

Table 3.9-13 (30 of 90)

Valve No.	Valve Description	Valve Type [a]	Valve Act [b]	Safety Class [c]	Code Cat [d]	Valve Funct [e]	Test Reqd [f]	Test Freq [g]
CV-505	RCP CBO containment isolation	GL	AD	2	A	CIC	S MT FS LPV LT	CS [7] CS [7] CS [7] 2 yr 2 yr
CV-506	RCP CBO containment isolation	GL	AD	2	A	CIC	S MT FS LPV LT	CS [7] CS [7] CS [7] 2 yr 2 yr
CV-509	IRWST makeup line containment isolation	GT	EL	2	A	CIC	S MT LPV LT	3 mo 3 mo 3 mo 2 yr
CV-515	Letdown isolation	GL	AD	1	A	TIV	S MT FS LPV LT	CS [8] CS [8] CS [8] 2 yr 2 yr
CV-516	Letdown isolation	GL	AD	1	A		S MT FS LPV LT	CS [8] CS [8] CS [8] 2 yr 2 yr

Table 3.9-13 (31 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CV-522	Letdown containment isolation	GL	AD	2	A	CIC	S MT FS LPV LT	CS (8) CS (8) CS (8) 2 yr 2 yr
CV-523	Letdown containment isolation	GL	AD	2	A	CIC	S MT FS LPV LT	CS (8) CS (8) CS (8) 2 yr 2 yr
CV-524	Charging containment isolation	GL	EL	2	A	CIC	S MT LPV LT	CS (9) CS (9) 2 yr 2 yr
CV-560	Reactor drain tank effluent containment isolation	GL	AD	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
CV-561	Reactor drain tank effluent containment isolation	GL	AD	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr

Table 3.9-13 (32 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CV-576	Charging flow restricting	GL	EL	2	A		LT LPV S MT	CS (41) 2yr CS (41) CS (41)
CV-577	Charging flow restricting	GL	EL	2	A		LT LPV MT S	CS (41) 2 yr CS (41) CS (41)
CV-580	Resin sluice supply header to reactor drain header isolation	GT	AD	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
CV-747	Charging line check	CK	SA	2	A/C	CIC	LT RF	2 yr CS (10)
CV-835	Seal injection containment isolation	CK	SA	2	A/C	CIC	LT RF	2 yr CS (11)
SD-0001	SG 1 blowdown isolation	GT	EL	2	B	P	LPV	2 yr
SD-0002	SG 2 blowdown isolation	GT	EL	2	B	P	LPV	2 yr
SD-0003	SG 3 blowdown isolation	GT	EL	2	B	P	LPV	2 yr
SD-0004	SG 4 blowdown isolation	GT	EL	2	B	P	LPV	2 yr
SD-0005	SG 1 blowdown to flash tank	GT	AD	2	A	CIN	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr

Table 3.9-13 (33 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SD-0007	SG 1 blowdown to flash tank	GT	EL	2	A	CIN	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
SD-0006	SG 1 blowdown to flash tank	GT	AD	2	A	CIN	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
SD-0008	SG 1 blowdown to flash tank	GT	EL	2	A	CIN	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
SD-1115	SG 1 wet layup recirculation isolation	CK	SA	2	A/C	CIN	RF LT	CS (27) CS (27)
SD-1116	SG 2 wet layup recirculation isolation	CK	SA	2	A/C	CIN	RF LT	CS (27) CS (27)
SD-1113	SG 1 wet layup recirculation isolation	GT	M	2	A	P, CIN	LT	CS
SD-1114	SG 2 wet layup recirculation isolation	GT	M	2	A	P, CIN	LT	CS
CC-0131	Essential central chiller condenser 2A isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0132	Essential central chiller condenser 2B isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr

Table 3.9-13 (34 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CC-0181	EDG 1A inlet isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0182	EDG 1B inlet isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0383	Essential central chiller condenser 1A isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0384	Essential central chiller condenser 1B isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0901	Essential central chiller condenser 1A outlet throttling	3W	AD	3	B		S FS MT	3 mo 3 mo 3 mo
CC-0902	Essential central chiller condenser 1B outlet throttling	3W	AD	3	B		S FS MT	3 mo 3 mo 3 mo
CC-0905	Essential central chiller condenser 2A outlet throttling	3W	AD	3	B		S FS MT	3 mo 3 mo 3 mo
CC-0906	Essential central chiller condenser 2B outlet throttling	3W	AD	3	B		S FS MT	3 mo 3 mo 3 mo
CC-1031	CCW HX 1A relief	RV	SA	3	C		RVT	10 yr
CC-1032	CCW HX 1B relief	RV	SA	3	C		RVT	10 yr

Table 3.9-13 (35 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CC-1033	CCW HX 2A relief	RV	SA	3	C		RVT	10 yr
CC-1034	CCW HX 2B relief	RV	SA	3	C		RVT	10 yr
CC-1035	CCW HX 3A relief	RV	SA	3	C		RVT	10 yr
CC-1036	CCW HX 3B relief	RV	SA	3	C		RVT	10 yr
CC-1109	CCW surge tank 1A N2 supply check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1110	CCW surge tank 1B N2 supply check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1111	CCW surge tank 1A relief	RV	SA	3	C		RVT	10 yr
CC-1112	CCW surge tank 1B relief	RV	SA	3	C		RVT	10 yr
CC-1303	CCW makeup pump 3A discharge check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1304	CCW makeup pump 3B discharge check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1309	CCW surge tank 1A makeup check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1310	CCW surge tank 1B makeup check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1317	CCW surge tank 1B demi. makeup check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1318	CCW surge tank 1A demi. makeup check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1319	CCW surge tank 1A demi. makeup check	CK	SA	3	C		S RF	3 mo 3 mo

Table 3.9-13 (36 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CC-1320	CCW surge tank 1B demi. makeup check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1325	CCW makeup to AFW storage tank A check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1326	CCW makeup to AFW storage tank B check	CK	SA	3	C		S RF	3 mo 3 mo
CC-0027	CCW HXS bypass A isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0028	CCW HXS bypass B isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0143	Non-essential supply header 1A isolation	BF	EL	3	B		S MT LPV	CS (3) CS (3) 2 yr
CC-0147	Non-essential return header 1B isolation	BF	EL	3	B		S MT LPV	CS (3) CS (3) 2 yr
CC-0021	CCW HX 01A outlet throttling	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0023	CCW HX 02A outlet throttling	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0025	CCW HX 03A outlet throttling	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr

Table 3.9-13 (37 of 90)

Valve No.	Valve Description	Valve Type [a]	Valve Act [b]	Safety Class [c]	Code Cat [d]	Valve Funct [e]	Test Reqd [f]	Test Freq [g]
CC-0351	SCS HX 1 outlet isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0389	SFP HX A outlet isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0097	CS HX 1 outlet isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0145	Non-essential supply header 1 isolation	BF	EL	3	B		S MT LPV	CS (3) CS (3) 2 yr
CC-0149	Non-essential return header 1 isolation	BF	EL	3	B		S MT LPV	CS (3) CS (3) 2 yr
CC-0231	CCW supply to RCP 1A, 1B, 2A, 2B isolation	BF	EL	2	B	CIC	S MT LPV LT	CS (1) CS (1) 2 yr 2 yr
CC-1001	CCW pump 1A discharge check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1003	CCW pump 2A discharge check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1099	CCW supply to RCP 1A, 1B, 2A, 2B check	CK	SA	2	A/C	CIC	S RF LT	CS (2) RO (2) 2 yr

Table 3.9-13 (38 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CC-0011	CCW surge tank 01A makeup supply header isolation	BL GL	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-1515	SC HX 1 header relief	RV	SA	3	C		RVT	10 yr
CC-1547	SC miniflow HX 1 header relief	RV	SA	3	C		RVT	10 yr
CC-0249	CCW return from RCP 1A, 1B, 2A, 2B isolation	BF	EL	2	A	CIC	S MT LPV LT	CS (1) CS (1) 2 yr 2 yr
CC-1247	CS miniflow HX 1 header relief	RV	SA	3	C		RVT	10 yr
CC-0250	CCW return from RCP 1A, 1B, 2A, 2B isolation	BF	EL	2	A	CIC	S MT LPV LT	CS (1) CS (1) 2 yr 2 yr
CC-1575	SFP cooling HX 1 header relief	RV	SA	3	C		RVT	10 yr
CC-1215	CS HX 1 header relief	RV	SA	3	C		RVT	10 yr
CC-1100	CCW return from RCP 1A, 1B, 2A, 2B isolation	CK	SA	2	A/C	CIC	S LT RF	RO (2) 2 yr CS (2)
CC-1569	Essential water chiller condenser 1A header relief	RV	SA	3	C		RVT	10 yr
CC-1107	CCW surge tank 01 vacuum relief	RV	SA	3	C		RVT	10 yr
CC-1269	Essential water chiller condenser 2A header relief	RV	SA	3	C		RVT	10 yr

Table 3.9-13 (39 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CC-0028	CCW HX bypass header 2 isolation	BF	EL	3	A		S MT LPV	3 mo 3 mo 2 yr
CC-0144	Non-essential supply header 2 isolation	BF	EL	3	B		S MT LPV	CS (3) CS (3) 2 yr
CC-0148	Non-essential return header 2 isolation	BF	EL	3	B		S MT LPV	CS (3) CS (3) 2 yr
CC-0022	CCW HX 01B outlet throttling	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0024	CCW HX 02B outlet throttling	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0026	CCW HX 03B outlet throttling	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0352	SCS HX 2 outlet isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0390	SFP HX B outlet isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0098	CS HX 1 outlet isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr

Table 3.9-13 (40 of 90)

Valve No.	Valve Description	Valve Type [a]	Valve Act [b]	Safety Class [c]	Code Cat [d]	Valve Funct [e]	Test Reqd [f]	Test Freq [g]
CC-0146	Non-essential supply header 2 isolation	BF	EL	3	B		S MT LPV	CS [(3)] CS [(3)] 2 yr
CC-0150	Non-essential return header 2 isolation	BF	EL	3	B		S MT LPV	CS [(3)] CS [(3)] 2 yr
CC-1003	CCW pump 2A discharge check	CK	SA	3	C		S RF	3 mo 3 mo
CC-1004	CCW pump 2B discharge check	CK	SA	3	C		S RF	3 mo 3 mo
CC-0937	Cross connection supply header isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0938	Cross connection supply header isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0939	Cross connection return header isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0940	Cross connection return header isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-0012	CCW surge tank 01B makeup supply header isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
CC-1516	SC HX 2 header relief	RV	SA	3	C		RVT	10 yr
CC-1548	SC miniflow HX 2 header relief	RV	SA	3	C		RVT	10 yr

Table 3.9-13 (41 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CC-1248	CS miniflow HX 2 header relief	RV	SA	3	C		RVT	10 yr
CC-1576	SFP cooling HX 2 header relief	RV	SA	3	C		RVT	10 yr
CC-1215	CS HX 1 header relief	RV	SA	3	C		RVT	10 yr
CC-1216	CS HX 2 header relief	RV	SA	3	C		RVT	10 yr
CC-0296	CCW supply to letdown HX isolation	BF	E	2	A	CIC	S MT LPV LT	CS (4) CS (4) 2 yr 2 yr
CC-0297	CCW supply to letdown HX isolation	BF	EL	2	A	CIC	S MT LPV LT	CS (4) CS (4) 2 yr 2 yr
CC-0301	CCW return from letdown HX isolation	BF	EL	2	A	CIC	S MT LPV LT	CS (4) CS (4) 2 yr 2 yr
CC-0302	CCW return from letdown HX isolation	BF	EL	2	A	CIC	S MT LPV LT	CS (4) CS (4) 2 yr 2 yr
CC-1570	Essential water chiller condenser 1B header relief	RV	SA	3	C		RVT	10 yr
CC-1270	Essential water chiller condenser 2B header relief	RV	SA	3	C		RVT	10 yr
CC-1685	CCW supply to letdown HX isolation	CK	SA	2	A/C	CIC	S LT RF	CS (5) 2 yr RO (5)

Table 3.9-13 (42 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CC-1686	CCW return from letdown HX isolation	CK	SA	2	A/C	CIC	S LT RF	RO (5) 2 yr CS (5)
CC-1108	CCW surge tank 01B vacuum relief	RV	SA	3	C		RVT	10 yr
SX-0043	ESW blowdown isolation	GT	S	3	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr
SX-0044	ESW blowdown isolation	GT	S	3	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr
SX-0045	ESW pump 1A discharge isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
SX-0046	ESW pump 1B discharge isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
SX-0047	ESW pump 2A discharge isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
SX-0048	ESW pump 2B discharge isolation	BF	EL	3	B		S MT LPV	3 mo 3 mo 2 yr
SX-1001	ESW pump 1A discharge	CK	SA	3	C		S RF	3 mo 3 mo

Table 3.9-13 (43 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
SX-1001	ESW pump 2A discharge	CK	SA	3	C		S RF	3 mo 3 mo
SX-1002	ESW pump 1B discharge	CK	SA	3	C		S RF	3 mo 3 mo
SX-1004	ESW pump 2B discharge	CK	SA	3	C		S RF	3 mo 3 mo
SX-1041	CCW HX 1A cold side relief	RV	SA	3	C		RVT	10 yr
SX-1042	CCW HX 1B cold side relief	RV	SA	3	C		RVT	10 yr
SX-1043	CCW HX 2A cold side relief	RV	SA	3	C		RVT	10 yr
SX-1044	CCW HX 2B cold side relief	RV	SA	3	C		RVT	10 yr
SX-1045	CCW HX 3A cold side relief	RV	SA	3	C		RVT	10 yr
SX-1046	CCW HX 3B cold side relief	RV	SA	3	C		RVT	10 yr
FC-1005	FC pump A discharge check	CK	SA	3	C		S RF	3 mo 3 mo
FC-1006	FC pump B discharge check	CK	SA	3	C		S RF	3 mo 3 mo
FC-1013	SFPC HX A relief	RV	SA	3	C	P	RVT	10yr
FC-1014	SFPC HX B relief	RV	SA	3	C	P	RVT	10 yr
FC-1142	Refueling pool cleanup suction isolation	GT	M	2	A	P, CIC	LT	2 yr

Table 3.9-13 (44 of 90)

Valve No.	Valve Description	Valve Type [(a)]	Valve Act [(b)]	Safety Class [(c)]	Code Cat [(d)]	Valve Funct [(e)]	Test Reqd [(f)]	Test Freq [(g)]
FC-1143	Refueling pool cleanup suction isolation	GT	M	2	A	P, CIC	LT	2 yr
FC-1144	Refueling pool cleanup discharge isolation	GT	M	2	A	P,CIC	LT	2 yr
FC-1145	Refueling pool cleanup discharge check valve	CK	SA	2	A/C	CIC	S LT RF	RO (2 yr)
FC-1217	IRWST return isolation	GT	M	3	B	P	LPV	2 yr
GW-0001	Reactor drain tank gas space to GWMS	GL	S	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
GW-0002	Reactor drain tank gas space to GWMS	GL	EL	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
DE-0005	Containment building isolation	GL	M	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
DE-0006	Containment building isolation	GL	AD	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr

Table 3.9-13 (45 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
PX-0001	Hot leg sample	GL	S	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
PX-0002	Hot leg sample	GL	S	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
PX-0003	PZR liquid sample	GL	S	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
PX-0004	PZR liquid sample	GL	S	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
PX-0005	PZR steam space sample	GL	S	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr

Table 3.9-13 (46 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
PX-0006	PZR steam space sample	GL	S	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
PX-0008	SI pump 1&3 miniflow sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0009	SI pump 2&4 miniflow sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0012	Purification filter sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0013	Purification filter sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0014	Debor. IX outlet sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0016	SC pump 1 miniflow sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0017	SC pump 2 miniflow sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0020	SI tank sample containment isolation	GL	S	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr

Table 3.9-13 (47 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
PX-0021	SI tank sample containment isolation	GL	S	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
PX-0022	SI tank 1A sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0023	SI tank 1B sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0024	SI tank 1C sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0025	SI tank 1D sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0026	CS pump 1 miniflow sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0027	CS pump 2 miniflow sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-0034	Sample return to VCT isolation	GL	S	3	B	P	FS LPV	3 mo 2 yr
PX-0041	Containment air sample containment isolation	GT	EL	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr

Table 3.9-13 (48 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
PX-0042	Containment air sample containment isolation	GT	EL	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
PX-0043	Containment air sample return containment isolation	GT	EL	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
PX-0053	PASS sample return	GL	S	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
PX-0063	VCT gas sample isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-1025	Sample return to VCT isolation	GL	S	2	B	P	FS LPV	3 mo 2 yr
PX-1005	Containment isolation	CK	SA	2	A/C	CIC	S RF LT	RO (42) RO (42) 2 yr
PX-1020	Containment air sample return containment isolation	CK	SA	2	A/C	CIC	S RF LT	RO (43) RO (43) 2 yr
AF-0035	Motor-driven AFW pump PP02A flow modulating	GL	S	3	B		S MT LPV	3 mo 3 mo 2 yr

Table 3.9-13 (49 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
AF-0036	Turbine-driven AFW pump PP01B flow modulating	GL	S	3	B		S MT LPV	3 mo 3 mo 2 yr
AF-0037	Steam-driven AFW pump PP01A flow modulating	GL	S	3	B		S MT LPV	3 mo 3 mo 2 yr
AF-0038	Motor-driven AFW pump PP02B flow modulating	GL	S	3	B		S MT LPV	3 mo 3 mo 2 yr
AF-0043	Motor-driven AFW pump PP02A AFW isolation	GT	EL	2	B	TIV, CIN	S MT LPV LT	3 mo 3 mo 2 yr 2 yr (39,45)
AF-0044	Turbine-driven AFW pump PP01B AFW isolation	GT	EL	2	B	TIV, CIN	S MT LPV LT	3 mo 3 mo 2 yr 2 yr (39,45)
AF-0045	Turbine-driven AFW pump PP01A AFW isolation	GT	EL	2	B	TIV, CIN	S MT LPV LT	3 mo 3 mo 2 yr 2 yr (39,45)

Table 3.9-13 (50 of 90)

Valve No.	Valve Description	Valve Type [(a)]	Valve Act [(b)]	Safety Class [(c)]	Code Cat [(d)]	Valve Funct [(e)]	Test Reqd [(f)]	Test Freq [(g)]
AF-0046	Motor-driven AFW pump PP02B AFW isolation	GT	EL	2	B	TIV, CIN	S MT LPV LT	3 mo 3 mo 2 yr 2 yr (39,45)
AF-1003A	Motor-driven AF pump PP02A discharge	CK	SA	3	C		S RF	CS (44) 3 mo (44)
AF-1003B	Motor-driven AF pump PP02B discharge	CK	SA	3	C		S RF	CS (44) 3 mo (44)
AF-1004A	Turbine-driven AF pump PP01A discharge	CK	SA	3	C		S RF	CS (44) 3 mo (44)
AF-1004B	Turbine-driven AF pump PP01B discharge	CK	SA	3	C		S RF	CS (44) 3 mo (44)
AF-1007A	Motor-driven AFW pump PP02A AFW isolation	CK	SA	2	C		S RF LT	CS (44) 3 mo (44) 2 yr (39)
AF-1007B	Motor-driven AFW pump PP02B AFW isolation	CK	SA	2	C		S RF LT	CS (44) 3 mo (44) 2 yr (39)
AF-1008A	Turbine-driven AFW pump PP01A AFW isolation	CK	SA	2	C		S RF LT	CS (44) 3 mo (44) 2 yr (39)
AF-1008B	Turbine-driven AFW pump PP01B AFW isolation	CK	SA	2	C		S RF LT	CS (44) 3 mo (44) 2 yr (39)

Table 3.9-13 (51 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
AF-1010A	Chemical injection for AFW	CK	SA	3	C		S RF	CS 3 mo
AF-1010B	Chemical injection for AFW	CK	SA	3	C		S RF	CS 3 mo
AF-1011A	Chemical injection for AFW	CK	SA	3	C		S RF	CS 3 mo
AF-1011B	Chemical injection for AFW	CK	SA	3	C		S RF	CS 3 mo
AT-0007	AF pump turbine TA01A emergency steam drain	GL	AD	3	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr
AT-0008	AF pump turbine TA01B emergency steam drain	GL	AD	3	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr
AT-0009	AF pump turbine TA01A steam isolation	GL	AD	3	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr
AT-0010	AF pump turbine TA01B steam isolation	GL	AD	3	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr
AT-0011	AF pump turbine TA01A steam bypass	GL	AD	3	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr

Table 3.9-13 (52 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
AT-0012	AF pump turbine TA01B steam isolation bypass	GL	AD	3	B		S MT FS LPV	3 mo 3 mo 3 mo 2 yr
CA-0013	Condenser vacuum exhaust gas	GT	EL	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
CA-1023	Condenser vacuum exhaust gas	CK	SA	2	A/C	CIC	S LT RF	3 mo 2 yr 3 mo
DG-4022A	Start Air Receiver 40A Inlet	CK	SA	3	A/C		Note 37	Note 37
DG-4022B	Start Air Receiver 40B Inlet	CK	SA	3	A/C		Note 37	Note 37
DG-4022C	Start Air Receiver 40C Inlet	CK	SA	3	A/C		Note 37	Note 37
DG-4022D	Start Air Receiver 40D Inlet	CK	SA	3	A/C		Note 37	Note 37
DG-4030A	Start Air Receiver 41A Inlet	CK	SA	3	A/C		Note 37	Note 37
DG-4030B	Start Air Receiver 41B Inlet	CK	SA	3	A/C		Note 37	Note 37
DG-4030C	Start Air Receiver 41C Inlet	CK	SA	3	A/C		Note 37	Note 37
DG-4030D	Start Air Receiver 41D Inlet	CK	SA	3	A/C		Note 37	Note 37
DG-4034A	Air Receiver 40A Discharge	CK	SA	3	C		Note 37	Note 37
DG-4034B	Air Receiver 40B Discharge	CK	SA	3	C		Note 37	Note 37
DG-4034C	Air Receiver 40C Discharge	CK	SA	3	C		Note 37	Note 37
DG-4034D	Air Receiver 40D Discharge	CK	SA	3	C		Note 37	Note 37
DG-4038A	Air Receiver 41A Discharge	CK	SA	3	C		Note 37	Note 37

Table 3.9-13 (53 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
DG-4038B	Air Receiver 41B Discharge	CK	SA	3	C		Note 37	Note 37
DG-4038C	Air Receiver 41C Discharge	CK	SA	3	C		Note 37	Note 37
DG-4038D	Air Receiver 41D Discharge	CK	SA	3	C		Note 37	Note 37
DG-4043A	Overspeed Air Receiver 42A inlet	CK	SA	3	C		Note 37	Note 37
DG-4043B	Overspeed Air Receiver 42B inlet	CK	SA	3	C		Note 37	Note 37
DG-4043C	Overspeed Air Receiver 42C inlet	CK	SA	3	C		Note 37	Note 37
DG-4043D	Overspeed Air Receiver 42D inlet	CK	SA	3	C		Note 37	Note 37
DG-4114A	Lube oil/LT water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-4114B	Lube oil/LT water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-4114C	Lube oil/LT water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-4114D	Lube oil/LT water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-4217A	HT/CC water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-4217B	HT/CC water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-4217C	HT/CC water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-4217D	HT/CC water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-4250A	CC/LT water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37

Table 3.9-13 (54 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
DG-4250B	CC/LT water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-4250C	CC/LT water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-4250D	CC/LT water heat exchanger temperature control 3-way valve	3W	SA	3	C		Note 37	Note 37
DG-5023A	Start Air Receiver 40A Relief Valve	RV	SA	3	C		RVT	10 yr
DG-5023B	Start Air Receiver 40B Relief Valve	RV	SA	3	C		RVT	10 yr
DG-5023C	Start Air Receiver 40C Relief Valve	RV	SA	3	C		RVT	10 yr
DG-5023D	Start Air Receiver 40D Relief Valve	RV	SA	3	C		RVT	10 yr
DG-5031A	Start Air Receiver 41A Relief Valve	RV	SA	3	C		RVT	10 yr
DG-5031B	Start Air Receiver 41B Relief Valve	RV	SA	3	C		RVT	10 yr
DG-5031C	Start Air Receiver 41C Relief Valve	RV	SA	3	C		RVT	10 yr
DG-5031D	Start Air Receiver 41D Relief Valve	RV	SA	3	C		RVT	10 yr
FP-0030	Fire water supply	GL	S	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
FP-1440	Fire water supply	CK	SA	2	A/C	CIC	LT RF	2 yr RO (32)
FW-0121	SG 1 main FW economizer isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS (13) CS (13) CS (13) 2 yr 2 yr(39)

Table 3.9-13 (55 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
FW-0122	SG 1 main FW economizer isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS (13) CS (13) CS (13) 2 yr 2 yr(39)
FW-0123	SG 2 main FW economizer isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS (13) CS (13) CS (13) 2 yr 2 yr(39)
FW-0124	SG 2 main FW economizer isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS (13) CS (13) CS (13) 2 yr 2 yr(39)
FW-0131	SG 1 Main FW downcomer isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS (13) CS (13) CS (13) 2 yr 2 yr(39)
FW-0132	SG 1 main FW downcomer isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS (13) CS (13) CS (13) 2 yr 2 yr(39)

Table 3.9-13 (56 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
FW-0133	SG 2 main FW downcomer isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS (13) CS (13) CS (13) 2 yr 2 yr(39)
FW-0134	SG 2 main FW downcomer isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS (13) CS (13) CS (13) 2 yr 2 yr(39)
FW-0138	Feedwater chemical injection valve	GL	AD	2	B	CIN	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
FW-0139	Feedwater chemical injection valve	GL	AD	2	B	CIN	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
FW-1035	SG 1 economizer FW line check valve	CK	SA	2	C		RF	CS (29)
FW-1036	SG 1 economizer FW line check valve	CK	SA	2	C		RF	CS (29)
FW-1037	SG 1 economizer FW line check valve	CK	SA	2	C		RF	CS (29)
FW-1039	SG 1 downcomer FW line check valve	CK	SA	2	C	CIN	RF	CS (29)
FW-1040	SG 1 downcomer FW line check valve	CK	SA	2	C		RF LT	CS (29) 2 yr(39)
FW-1042	SG 2 economizer FW line check valve	CK	SA	2	C		RF	CS (29)
FW-1043	SG 1 economizer FW line check valve	CK	SA	2	C		RF	CS (29)

Table 3.9-13 (57 of 90)

Valve No.	Valve Description	Valve Type [(a)]	Valve Act [(b)]	Safety Class [(c)]	Code Cat [(d)]	Valve Funct [(e)]	Test Reqd [(f)]	Test Freq [(g)]
FW-1044	SG 2 economizer FW line check valve	CK	SA	2	C		RF	CS [(29)]
FW-1046	SG 2 downcomer FW line check valve	CK	SA	2	C	CIN	RF LT	CS [(29)] 2 yr[(39)]
FW-1047	SG 2 downcomer FW line check valve	CK	SA	2	C		RF LT	CS [(29)] 2 yr [(39)]
FW-1050	SG 1 feedwater chemical injection	CK	SA	2	C		S RF	3 mo
FW-1052	SG 1 feedwater chemical injection	CK	SA	2	C		S RF	3 mo
IA-0020	Instrumentation air supply	GL	AD	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
IA-1601	Instrumentation air supply	CK	SA	2	A/C	CIC	S LT RF	3 mo 2 yr 3 mo
MS-011	SG 1 Main Steam Isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS [(14)] CS [(14)] CS [(14)] 2 yr 2 yr[(39)]
MS-012	SG 1 Main Steam Isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS [(14)] CS [(14)] CS [(14)] 2 yr 2 yr [(39)]

Table 3.9-13 (58 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
MS-013	SG 2 Main Steam Isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS (14) CS (14) CS (14) 2 yr 2 yr (39)
MS-014	SG 2 Main Steam Isolation	GT	EH	2	B	CIN	S MT FS LPV LT	CS (14) CS (14) CS (14) 2 yr 2 yr (39)
MS-015	SG 1 Main Steam Isolation Valve Bypass	GT	EH	2	B	CIN	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr (39)
MS-016	SG 1 Main Steam Isolation Valve Bypass	GT	EH	2	B	CIN	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr (39)
MS-017	SG 2 Main Steam Isolation Valve Bypass	GT	EH	2	B	CIN	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr (39)

Table 3.9-13 (59 of 90)

Valve No.	Valve Description	Valve Type [(a)]	Valve Act [(b)]	Safety Class [(c)]	Code Cat [(d)]	Valve Funct [(e)]	Test Reqd [(f)]	Test Freq [(g)]
MS-018	SG 2 Main Steam Isolation Valve Bypass	GT	EH	2	B	CIN	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr [(39)]
MS-090	SG1 Main Steam Drip Leg Isolation Valve	GL	AD	2	B	CIN	S MT FS LPV LT	CS [(14)] CS [(14)] CS [(14)] 2 yr 2 yr[(39)]
MS-091	SG1 Main Steam Drip Leg Isolation Valve	GL	AD	2	B	CIN	S MT FS LPV LT	CS [(14)] CS [(14)] CS [(14)] 2 yr 2 yr [(39)]
MS-092	SG2 Main Steam Drip Leg Isolation Valve	GL	AD	2	B	CIN	S MT FS LPV LT	CS [(14)] CS [(14)] CS [(14)] 2 yr 2 yr [(39)]
MS-093	SG2 Main Steam Drip Leg Isolation Valve	GL	AD	2	B	CIN	S MT FS LPV LT	CS [(14)] CS [(14)] CS [(14)] 2 yr 2 yr [(39)]

Table 3.9-13 (60 of 90)

Valve No.	Valve Description	Valve Type [a]	Valve Act [b]	Safety Class [c]	Code Cat [d]	Valve Funct [e]	Test Reqd [f]	Test Freq [g]
MS-101	SG 1 Atmospheric Dump Valve	GL	EH	2	B	CIN	S MT FS LPV LT	2 yr 2 yr 2 yr 2 yr 2 yr [39]
MS-102	SG 1 Atmospheric Dump Valve	GL	EH	2	B	CIN	S MT FS LPV LT	2 yr 2 yr 2 yr 2 yr 2 yr [39]
MS-103	SG 2 Atmospheric Dump Valve	GL	EH	2	B	CIN	S MT FS LPV LT	2 yr 2 yr 2 yr 2 yr 2 yr[39]
MS-104	SG 2 Atmospheric Dump Valve	GL	EH	2	B	CIN	S MT FS LPV LT	2 yr 2 yr 2 yr 2 yr 2 yr [39]
MS-105	SG 1 MS ADV Isolation Valve	GT	EL	2	B	CIN	S MT LPV LT	3 mo 3 mo 2 yr 2 yr [39]

Table 3.9-13 (61 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
MS-106	SG 1 MS ADV Isolation Valve	GT	EL	2	B	CIN	S MT LPV LT	3 mo 3 mo 2 yr 2 yr (39)
MS-107	SG 2 MS ADV Isolation Valve	GT	EL	2	B	CIN	S MT LPV LT	3 mo 3 mo 2 yr 2 yr (39)
MS-108	SG 2 MS ADV Isolation Valve	GT	EL	2	B	CIN	S MT LPV LT	3 mo 3 mo 2 yr 2 yr (39)
MS-109	AF Pump turbine TA01B steam supply	GL	AD	2	B	-	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
MS-110	AF Pump turbine TA01A steam supply	GL	AD	2	B	-	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
MS-111	AF Pump turbine TA01B warmup	GL	AD	2	B	-	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
MS-112	AF Pump turbine TA01A steam supply	GL	AD	2	B	-	S MT FS LPV	3 mo 3 mo 3 mo 2 yr

Table 3.9-13 (62 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
MS-1301	SG 1 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1302	SG 1 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1303	SG 1 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1304	SG 1 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1305	SG 1 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1306	SG 1 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1307	SG 1 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1308	SG 1 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1309	SG 1 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1310	SG 1 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1311	SG 2 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1312	SG 2 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1313	SG 2 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1314	SG 2 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)

Table 3.9-13 (63 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
MS-1315	SG 2 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1316	SG 2 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1317	SG 2 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1318	SG 2 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1319	SG 2 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
MS-1320	SG 2 Main Steam Safety Valve	RV	SA	2	C	CIN	RVT LT	5 yr 2 yr (39)
NT-0004	Nitrogen supply	GL	AD	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
NT-1016	Nitrogen supply	CK	SA	2	A/C	CIC	S LT RF	3 mo 2 yr 3 mo
SA-0001	Service air supply	GL	AD	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
SA-1401	Service air supply	CK	SA	2	A/C	CIC	S LT RF	3 mo 2 yr 3 mo

Table 3.9-13 (64 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
VQ-0011	Containment High Volume Purge Supply 1	BF	EH	2	A	CIC	S MT FS LPV LT	CS (26) CS (26) CS (26) 2 yr 2 yr
VQ-0012	Containment High Volume Purge Supply 1	BF	EL	2	A	CIC	S MT FS LPV LT	CS (26) CS (26) CS (26) 2 yr 2 yr
VQ-0013	Containment High Volume Purge Exhaust 1	BF	EL	2	A	CIC	S MT FS LPV LT	CS (26) CS (26) CS (26) 2 yr 2 yr
VQ-0014	Containment High Volume Purge Exhaust 1	BF	EH	2	A	CIC	S MT FS LPV LT	CS (26) CS (26) CS (26) 2 yr 2 yr
VQ-0031	Containment Low Volume Purge Supply	BF	AD	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr

Table 3.9-13 (65 of 90)

Valve No.	Valve Description	Valve Type [a]	Valve Act [b]	Safety Class [c]	Code Cat [d]	Valve Funct [e]	Test Reqd [f]	Test Freq [g]
VQ-0032	Containment Low Volume Purge Supply	BF	AD	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
VQ-0033	Containment Low Volume Purge Exhaust	BF	AD	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
VQ-0034	Containment Low Volume Purge Exhaust	BF	AD	2	A	CIC	S MT FS LPV LT	3 mo 3 mo 3 mo 2 yr 2 yr
WM-1751	Demineralized water supply	GL	M	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
WM-1752	Demineralized water supply	CK	SA	2	A/C	CIC	S LT RF	RO (33) 2 yr RO (33)
WO-906A	Essential chilled water flow control control room AHU HV01A	3W	AD	3	B		S MT LPV	2 yr 2 yr 2 yr

Table 3.9-13 (66 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
WO-906B	Essential chilled water flow control control room AHU HV01B	3W	AD	3	B		S MT LPV	2 yr 2 yr 2 yr
WO-906C	Essential chilled water flow control control room AHU HV01C	3W	AD	3	B		S MT LPV	2 yr 2 yr 2 yr
WO-906D	Essential chilled water flow control control room AHU HV01D	3W	AD	3	B		S MT LPV	2 yr 2 yr 2 yr
WO-917A	Essential chilled water flow control EDG normal AHU HV11C	3W	AD	3	B		S MT LPV	2 yr 2 yr 2 yr
WO-917B	Essential chilled water flow control EDG normal AHU HV11D	3W	AD	3	B		S MT LPV	2 yr 2 yr 2 yr
WO-918A	Essential chilled water flow control EDG normal AHU HV11A	3W	AD	3	B		S MT LPV	2 yr 2 yr 2 yr
WO-918B	Essential chilled water flow control EDG normal AHU HV11B	3W	AD	3	B		S MT LPV	2 yr 2 yr 2 yr
WO-1001A	Essential chilled water compression tank A relief	RV	SA	3	C		RVT	10 yr
WO-1001B	Essential chilled water compression tank B relief	RV	SA	3	C		RVT	10 yr
WO-1003A	Essential chilled water compression tank A MDS makeup	CK	SA	3	C		RF	3 mo

Table 3.9-13 (67 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
WO-1003B	Essential chilled water compression tank B MDS makeup	CK	SA	3	C		RF	3 mo
WO-1010A	Essential chilled water pump PP01A discharge	CK	SA	3	C		S RF	3 mo 3 mo
WO-1010B	Essential chilled water pump PP01B discharge	CK	SA	3	C		S RF	3 mo 3 mo
WO-1011A	Essential chilled water makeup pump A discharge	CK	SA	3	C		S RF	3 mo 3 mo
WO-1011B	Essential chilled water makeup pump B discharge	CK	SA	3	C		S RF	3 mo 3 mo
WO-1014A	Essential chilled water pump PP02A discharge	CK	SA	3	C		S RF	3 mo 3 mo
WO-1014B	Essential chilled water pump PP02B discharge	CK	SA	3	C		S RF	3 mo 3 mo
WO-1022A	Essential chilled water makeup pump A discharge	CK	SA	3	C		S RF	3 mo 3 mo
WO-1022B	Essential chilled water makeup pump B discharge	CK	SA	3	C		S RF	3 mo 3 mo
WO-1031A	Essential chilled water compression tank A nitrogen supply	CK	SA	3	C		RF	3 mo
WO-1031B	Essential chilled water compression tank B nitrogen supply	CK	SA	3	C		RF	3 mo
WO-1032A	Essential chilled water compression tank A MDS makeup	CK	SA	3	C		RF	3 mo
WO-1032B	Essential chilled water compression tank B MDS makeup	CK	SA	3	C		RF	3 mo

Table 3.9-13 (68 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
WI-012	PCW Containment Return	GT	AD	2	A	CIC	S MT FS LPV LT	3 mo(28) 3 mo(28) 3 mo(28) 2 yr 2 yr
WI-013	PCW Containment Supply	GT	AD	2	A	CIC	S MT FS LPV LT	3 mo(28) 3 mo(28) 3 mo(28) 2 yr 2 yr
WI-014	PCW Containment Return	RV	SA	2	A		RVT	10yr
WI-015	PCW Containment Return	GT	EL	2	A	CIC	S MT FS LPV LT	3 mo(28) 3 mo(28) 3 mo(28) 2 yr 2 yr
WI-1043	PCW Containment Supply	CK	SA	2	A/C	CIC	S RF LT	3 mo 3 mo 2 yr
PR-0431	Containment radiation monitor sample inlet	GT	EL	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
PR-0432	Containment radiation monitor sample inlet	GT	EL	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr

Table 3.9-13 (69 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
PR-0434	Containment radiation monitor sample outlet	GT	EL	2	A	CIC	S MT LPV LT	3 mo 3 mo 2 yr 2 yr
PR-1433	Containment radiation monitor sample outlet	CK	SA	2	A/C	CIC	S RF LT	3 mo RO 2 yr
CM-0001	Containment H2 sample Ch. A inlet isolation	GL	S	2	A	CIC	S FS LT MT PV	3 mo 3 mo 2 yr 3 mo 2 yr
CM-0002	Containment H2 sample Ch. B inlet isolation	GL	S	2	A	CIC	S FS LT MT LPV	3 mo 3 mo 2 yr 3 mo 2 yr
CM-0003	Containment H2 analyzer Ch. A inlet isolation	GL	S	2	A	CIC	S FS LT MT LPV	3 mo 3 mo 2 yr 3 mo 2 yr
CM-0004	Containment H2 analyzer Ch. B inlet isolation	GL	S	2	A	CIC	S FS LT MT LPV	3 mo 3 mo 2 yr 3 mo 2 yr

Table 3.9-13 (70 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CM-0009	Containment H2 analyzer Ch. A outlet isolation	GL	S	2	A	CIC	S FS LT MT LPV	3 mo 3 mo 2 yr 3 mo 2 yr
CM-0010	Containment H2 analyzer Ch. B outlet isolation	GL	S	2	A	CIC	S FS LT MT LPV	3 mo 3 mo 2 yr 3 mo 2 yr
CM-0011	Containment IRWST H2 sample Ch. A inlet isolation	GL	S	2	A	CIC	FS LT S MT LPV	3 mo 2 yr 3 mo 3 mo 2 yr
CM-0012	Containment IRWST H2 sample Ch. B inlet isolation	GL	S	2	A	CIC	FS LT S MT LPV	3 mo 2 yr 3 mo 3 mo 2 yr
CM-0013	Containment H2 sample Ch. A inlet isolation	GL	S	2	A	CIC	S FS LT MT LPV	3 mo 3 mo 2 yr 3 mo 2 yr

Table 3.9-13 (71 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CM-0014	Containment H2 sample Ch. B inlet isolation	GL	S	2	A	CIC	S FS LT MT LPV	3 mo 3 mo 2 yr 3 mo 2 yr
CM-0017	Containment pressure monitor Ch. A isolation	GL	S	2	A	CIC	S FS LT MT LPV	3 mo 3 mo 2 yr 3 mo 2 yr
CM-0018	Containment pressure monitor Ch. B isolation	GL	S	2	A	CIC	FS LT S MT LPV	3 mo 2 yr 3 mo 3 mo 2 yr
CM-0019	Containment pressure monitor Ch. C isolation	GL	S	2	A	CIC	FS LT S MT LPV	3 mo 2 yr 3 mo 3 mo 2 yr
CM-0020	Containment pressure monitor Ch. D isolation	GL	S	2	A	CIC	FS LT S MT LPV	3 mo 2 yr 3 mo 3 mo 2 yr

Table 3.9-13 (72 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
CM-0023	Containment H2 sample Ch. A inlet isolation	GL	S	2	A	CIC	S FS LT MT LPV	3 mo 3 mo 2 yr 3 mo 2 yr
CM-0024	Containment H2 sample Ch. B inlet isolation	GL	S	2	A	CIC	S FS LT MT LPV	3 mo 3 mo 2 yr 3 mo 2 yr
CM-1013	Containment H2 sample Ch. A outlet isolation	CK	SA	2	A/C	CIC	S RF LT	RO RO 2 yr
CM-1014	Containment H2 sample Ch. B outlet isolation	CK	SA	2	A/C	CIC	S RF LT	RO RO 2 yr

Table 3.9-13 (73 of 90)

Valve No.	Valve Description	Valve Type (a)	Valve Act (b)	Safety Class (c)	Code Cat (d)	Valve Funct (e)	Test Reqd (f)	Test Freq (g)
PS-0031	SG 1 blowdown hot leg	GT	AD	2	B	CIN	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
PS-0032	SG 2 blowdown hot leg	GT	AD	2	B	CIN	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
PS-0033	SG 1 downcomer	GT	AD	2	B	CIN	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
PS-0034	SG 2 downcomer	GT	AD	2	B	CIN	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
PS-0035	SG 1 blowdown cold leg	GT	AD	2	B	CIN	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
PS-0036	SG 2 blowdown cold leg	GT	AD	2	B	CIN	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
PS-0257	SG 1 blowdown hot leg	GT	AD	2	B	CIN	S MT FS LPV	3 mo 3 mo 3 mo 2 yr
PS-0258	SG 2 blowdown hot leg	GT	AD	2	B	CIN	S MT FS LPV	3 mo 3 mo 3 mo 2 yr

Table 3.9-13 (74 of 90)

Notes:

(a) Valve Type:

GL -	Globe	BF -	Butterfly
GT -	Gate	PK -	Packless
CK -	Check	PL -	Plug
RV -	Relief	POS -	Pilot Operated Safety Relief Valve
BL -	Bal	3W -	3Way
BL -	Ball		

(b) Valve Actuator:

EL -	Electric motor	S -	Solenoid
SA -	Self actuating	EH -	Electro-hydraulic
AD -	Air diaphragm	P -	Piston
M -	Manual		

(c) Safety Classification as defined in Subsection 3.2.3.

(d) Valve ASME Code Category A, B, C, or D as defined in ASME OM Code, ISTC 1300.

(e) Valve Function:

CIC - Containment Isolation Valve as listed in Table 6.2.4-1, which is Type-C leakage rate tested in accordance with ANSI/ANS 56.8.
CIN - Containment Isolation Valve as listed in Table 6.2.4-1, which is not Type-C leakage rate tested in accordance with ANSI/ANS 56.8.
PIV - Pressure Isolation Valve.
TIV - Temperature Isolation Valve.
P - Passive valves as defined by ASME OM Code, ISTA-2000 are denoted by a P in this column. All other valves are active valves.

Table 3.9-13 (75 of 90)

(f) Required Valve Tests per ASME OM Code, ISTC and Mandatory Appendix I; and additional required testing:

LT - Valve Leakage Rate Test (per ASME OM Code, ISTC): Subsections ASME OM Code, ISTC for valves with function CIC in (e) above.
Subsection ASME OM Code, ISTC for valves with function PIV in (e) above.
Reactor Coolant System PIVs are leakage rate tested in accordance with Technical Specifications Surveillance Requirement 3.4.13.1.
Subsection ASME OM Code, ISTC for Category A valves except the valves with function TIV.
Subsection ASME OM Code, ISTC for valves with function TIV.

LPV - Valve Position Verification (ASME OM Code, ISTC)

S - Valve Stroke Exercise in the forward flow direction:
Category A or B (ASME OM Code, ISTC)
Category C (ASME OM Code, ISTC)

RF - Reverse Flow Exercise for A/C and C valves (ASME OM Code, ISTC). “RF” testing is performed at the same testing frequency as the corresponding "S" test, unless otherwise described.

MT - Valve Stroke Time Test of Category A or B power operated valves (ASME OM Code, ISTC)

FS - Valves Test for fail-safe actuation of Category A or B valves (ASME OM Code, ISTC)

RVT - Relief Valve Test (ASME OM Code, ISTC)

Table 3.9-13 (76 of 90)

(g) Pump or valve test exclusions, alternatives, and frequency per ASME OM Code, ISTB and ASME OM Code, ISTC. For valves whose test frequency exceeds the normal frequency, see the note (as indicated in parenthesis beside the test frequency) for additional information/justification.

CS - Cold Shutdown

The following condition applies for all testing performed during cold shutdown:

Cold shutdown testing in accordance with the requirements of ASME OM Code, ISTC. See the note for additional information/justification. Valve exercising during cold shutdown commences until all testing is complete or the plant is ready to return to power. A completion of all valve testing is not a prerequisite to return to power. Any testing not completed by the end of one cold shutdown is performed during subsequent cold shutdowns, starting from the last test performed at the previous cold shutdown. In case of frequent shutdowns, testing is not performed more often than once every 3 months.

RO - Refueling Outage.

All Refueling Outage valve testing is to be completed prior to returning the plant to operation.

RR - Partially stroke valve at or when proceeding to/starting up from cold shutdown. Fully stroke valve during each refueling outage. Some valves may require mechanical exercising or disassembly during each refueling outage to verify operability. All RR testing measures are completed prior to returning the plant to operation.

QC - Partially stroke valve every 3 months. Fully stroke valve during cold shutdown.

EI - Valve operates in the course of plant operation at a frequency that satisfies test requirements. Additional exercising not required provided the test parameters are analyzed and recorded at an operational interval not exceeding the test interval requirement.

Category A or B (ASME OM Code, ISTC)

Category C (ASME OM Code, ISTC)

Table 3.9-13 (77 of 90)

(h) Pump Test Parameters as defined in ASME OM Code, ISTB-5000:

N -	Speed	V -	Vibration
DP -	Differential Pressure	SPs -	Static Suction Pressure
Q -	Flow Rate	SPo -	Operating Suction Pressure
SPc -	Calculated Suction Pressure		

Note: If ASME OM-ISTB pump tests cannot be performed on the CCW or ESW pumps due to inability to repeat pump tests single point flow conditions, pump curve testing will be used to assess pump degradation, as described in Subsection 3.9.6.1.

(i) Typical test configurations for pumps and valves requiring special valve arrangements and/or test connections are shown in Figure 3.9-15. When referenced, these typical test configurations constitute design requirements for the affected pump/valve to be reflected in affected documentation by the COL applicant during later detailed design.

(1) Valves: CC-0231, CC-0249, CC-0250

During normal operations, these valves are open to supply/return cooling water to/from the reactor coolant pump (RCP) coolers. Failure of these valves in the closed position could lead to pump damage or failure and force a unit shutdown. Therefore, these valves will be tested during cold shutdown when the RCPs are not operating.

(2) Valves: CC-1099, CC-1100

These valves provide containment isolation and overpressure protection for the component cooling water (CCW) supply and return lines to/from the RCPs (refer to Figure 9.2.2-1). Since these CCW lines are to remain in service during plant operation, it is impractical to perform S or RF testing on the valves on a quarterly test frequency.

The reverse stroke (RF) test of CC-1099 is impractical to perform without isolating CC-1071and CC-1070, and then pressurizing against the check valve seat in the reverse flow direction via test connection CC-2083. The resultant leakage is then measured through test connection CC-2081. Since this method of testing requires access to areas of high radiation and contamination, a test of this type can be performed only during refueling. This method of testing is the same as will be employed for the ANSI/ANS 56.8-1994 Type-C leakage rate tests. Therefore, LT testing accomplishes and satisfies the reverse flow testing requirements for CC-1099.

Table 3.9-13 (78 of 90)

The reverse stroke (RF) test of CC-1100 is impractical to perform without isolating CC-02491and CC-1085 and then pressurizing against the check valve seat in the reverse flow direction via test connection CC-2086. The resultant leakage is then measured through test connection CC-2086. Since this method of testing requires access to areas of high radiation and contamination, a test of this type can be performed only during refueling. This method of testing is the same as will be employed for the ANSI/ANS 56.8-1994 Type-C leakage rate tests. Therefore, LT testing accomplishes and satisfies the reverse flow testing requirements for CC-1100.

(3) Valves: CC-0143, CC-0144, CC-0145, CC-0146, CC-0147, CC-0148, CC-0149, CC-0150

These valves close on receipt of a safety injection actuation signal to isolate the non-essential component cooling water (CCW) loops. The non-essential cooling loops provide cooling of the non-essential chillers. When Division 1 non-essential chilled water (NECW) is secured for testing of Division 1 NECW valves and Division 1 non-essential header CCW valves CC-0143, CC-0145, CC-0147, and CC-0149 may then be stroke tested. These valves will use the same test frequency as the NECW valves, as described in (28).

The Division 2 non-essential CCW header services the letdown heat exchanger in addition to the Division 2 non-essential chillers. Closing the Division 2 non-essential CCW header valves during plant operation could result in unnecessary reactor coolant system transients. Also, failure to cool the high temperature letdown flow leaving the regenerative heat exchanger can lead to cavitation at the letdown orifices, which has been known to cause line failure. Therefore, valves CC-0144, CC-0146, CC-0148, and CC-0150 will be tested during cold shutdown.

(4) Valves: CC-0296, CC-0297, CC-0301, CC-0302

These valves isolate cooling water to/from the letdown heat exchanger and close on a containment isolation actuation signal. For reasons stated in Note (3) above, testing these valves during normal operations is not practical. Therefore, these valves will be tested during cold shutdown.

(5) Valves: CC-1685, CC-1686

These valves provide containment isolation and overpressure protection for the component cooling water (CCW) supply and return lines to/from the letdown heat exchanger (refer to Figure 9.2.2-1 (4 of 4)). Since these CCW lines are to remain in service during plant operation, it is impractical to perform S or RF testing on the valves on a quarterly test frequency.

Valve CC-1685 is forward stroke tested during cold shutdown by isolating CC-0297 while keeping CC-0296 open to allow a CCW header pressure to stroke CC-1685. The reverse stroke (RF) test of CC-1685, however, is impractical to perform without isolating CC-0297, CC-1683, and CC-1681, and then pressurizing against the check valve seat in the reverse direction via test connection CC-2663. The resultant leakage is then measured through test connection CC-2662. Since this method of testing requires access to areas of high radiation and contamination, a test of this type can be performed only during refueling. This method of testing is the same as will be employed for the ANSI/ANS 56.8-1994 Type-C leakage rate tests. Therefore, LT testing accomplishes and satisfies the reverse flow testing requirement for CC-1685.

Valve CC-1686 is reverse flow stroke tested during cold shutdown. With CCW letdown supply and return lines in service, CC-0301 and CC-1682 are isolated, thus backseating check valve CC-1686 against CCW header pressure. Any leakage past CC-1686 may then be measured via test connection CC-2667. The forward stroke (S) of CC-1686, however, is impractical to perform without isolating CC-0301, CC-1682, and CC-1684, while keeping CC-0302 open and injecting a test flow through test connection CC-2666. The resultant outleakage is then measured at test connection CC-2668. Since this valve testing methodology requires containment entries to areas of high radiation and contamination, the forward stroke testing of CC-1686 will be performed during refueling.

(6) Valves: CV-255

This valve isolates seal injection water to the RCP seals. Valve closure during normal operations with the RCPs operating would result in damage to pump seals. Therefore, this valve will be tested during cold shutdown when the RCPs are not operating.

(7) Valves: CV-505, CV-506

These valves close on receipt of a containment spray actuation signal to isolate the RCP HP leakage line. During normal operations, these valves are open to maintain seal injection flow across the RCP seals. Closure of these valves during normal operations would inhibit seal water flow across the RCP seals, which would result in damage to the pump seals. Therefore, these valves will be tested during cold shutdown when the RCPs are not operating.

(8) Valves: CV-515, CV-516, CV-522, CV-523

These valves are normally open to pass letdown flow from the RCS to the chemical and volume control system (CVCS). Stroking these valves during normal operations could result in unnecessary RCS transients. In addition, these valves experience high stresses when cycled due to the high pressure environment in which they operate. Repeated cycling of the valves at this high pressure could severely affect valve integrity over the expected operating life of the valves. In addition, failure of these valves in the closed position could result in a loss of pressurizer level control, forcing a unit shutdown. Therefore, these valves will be tested during cold shutdown when the effects of valve operation are minimized. Globe valve CV-515 performs a temperature isolation valve (TIV) function. This valve isolates the letdown line on a high temperature, as sensed downstream of the letdown heat exchanger by dedicated temperature monitors (refer to Figure 9.3.4-1, Sheet 1).

The setpoints of these temperature monitors and associated valve isolation actuation circuitry are such that the design temperature limits of the interfacing CVCS piping and components will not be exceeded prior to the closure of CV-515. Temperature monitors are also used to evaluate the integrity of CV-515 in this closed position. Each refueling outage, an integrity evaluation of CV-515 performed by isolating the letdown line using CV-515, and then subjecting the valve to reactor coolant system pressure and temperature and analyzing the resultant temperature differential across the valve over time. RCS pressure and temperature may be actually lower than plant at-power RCS pressure and temperature levels to avoid valve duty stress, provided these parameters are analyzed and extrapolated to full RCS pressure and temperature.

Table 3.9-13 (80 of 90)

(9) Valves: CV-524

This valve functions as a containment isolation valve and isolates charging flow to the RCS. During normal operations, this charging flow is used to cool the letdown flow in the regenerative heat exchanger and to provide makeup to the RCS. For reasons stated in Note (3), it is not practical to test this valve during normal operations. In addition, failure of this valve in the closed position could result in a loss of pressurizer level control forcing a unit shutdown. Therefore, this valve will be tested during cold shutdown.

(10) Valves: CV-747

This valve functions as a containment isolation valve. Testing requires that charging flow be isolated. As stated in Note (9) above, this is not practical during normal operations. Therefore, this valve will be tested during cold shutdown.

(11) Valves: CV-835

This valve functions as a containment isolation valve. Testing requires that seal injection to the RCP's be isolated. As stated in Note (6), this is not practical during normal operations. Therefore, this valve will be tested during cold shutdown.

(12) Valves: RG-0410, RG-0411, RG-0412, RG-0413, RG-0414, RG-0415, RG-0416, RG-0417

These valves are closed during normal plant operating to maintain the reactor coolant pressure boundary (RCPB). These valves are active valves and are designed to be used during a safety grade cooldown of the RCS. Opening these valves during normal operation leaves only one Class 1 valve, which does not maintain the RCPB according to 10 CFR 50.2 and ANSI/ANS 51.1 definitions. While there is a third valve downstream of the two reactor coolant gas vent system (RCGVS) valves, the piping and the third valve are Class 2. In order to maintain the integrity of the RCPB, these valves are to be tested during plant shutdown periods only and not during reactor operation.

These valves will be tested as cold shutdown valves.

(13) Valves: FW-0121, FW-0122, FW-0123, FW-0124, FW-0131, FW-0132, FW-0133, FW-0134

These valves isolate main feedwater to the SGs upon receipt of a main steam isolation signal (MSIS). Closure of these valves during normal operations would isolate feedwater to the SGs, which may result in a severe transient in the SG and a unit trip. Therefore, these valves will be tested during cold shutdown.

Table 3.9-13 (81 of 90)

(14) Valves: MS-011, MS-012, MS-013, MS-014, MS-090, MS-091, MS-092, MS-093

These valves are main steam isolation valves (MSIVs) and main steam drip leg isolation valves, which isolate the main steam lines upon receipt of an MSIS. Performance of either a full stroke or partial stroke test during normal operations may cause severe transients in the main steam lines and result in a unit trip. The valves will therefore be full stroke tested on a cold shutdown frequency basis, but with the unit in MODE 3 and at operating temperature and pressure, so as to replicate design conditions under which valve closure is to be achieved.

(15) Valves: SI-568, SI-569

These valves are to close to prevent reverse flow when either the SC/CS pumps are use to provide containment spray or to provide shutdown cooling flow. These valves are tested by operating the SC/CS pump in any train, opening the discharge crossover isolation valve between the two systems, and isolating the suction of the off-line pump. Closure of the check valve on reverse flow in the discharge of the off-line pump is verified by monitoring pressure increase upstream of the valve.

(16) Valves: SI-113, SI-133, SI-404, SI-405, SI-434, SI-446, SI-522, SI-523, SI-532, SI-533, SI-540, SI-542

Check valves SI-113, SI-133, SI-404, SI-405, SI-434, SI-446, SI-540, and SI-542 are to be provided with sufficient flow from the SI pumps to stroke to their full open position. The flow for stroke testing these valves passes through the DVI nozzles and into the RCS. The SI pump discharge pressure is not sufficient to overcome normal RCS operating pressure. In addition, any flow from safety injection through these valves and into the RCS during power operations would produce an undesirable temperature transient at the DVI nozzles. The valves are also not full or partial stroked during cold shutdown, since this may result in low temperature overpressurization of the RCS. Since it is impractical to full stroke test these check valves during plant operation or to perform full/partial stroke test during cold shutdown conditions, these valves are full stroke tested each refueling outage.

Check valves SI-520, SI-522, SI-523, SI-532, and SI-533 are to be provided with sufficient flow from the SI pumps to stroke to their full open position. The flow for stroke testing these valves passes through the RCS hot legs (hot leg injection). The SI pump discharge pressure is not sufficient to overcome normal RCS operating pressure. In addition, any flow from safety injection through these valves and into the RCS during power operations would produce an undesirable temperature transient at the shutdown cooling line connections to the hot legs. The valves are also not full or partial stroked during cold shutdown, since this may result in low temperature overpressurization of the RCS. Since it is impractical to full stroke test these check valves during plant operation or to perform full/partial stroke test during cold shutdown conditions, these valves are full stroke tested each refueling outage.

Check valves SI-113 and SI-133 are not reverse flow tested quarterly, since testing of these valves during power operations would require containment entries by plant personnel to high radiation and airborne contamination areas. These valves are not reverse flow tested every cold shutdown, because of the extensive test equipment setup which could extend the cold shutdown. These valves are reverse flow tested during refueling.

Table 3.9-13 (82 of 90)

(16A) Valves: SI-404, SI-405, SI-434, SI-446

These valves are reverse flow tested by pressurizing the volume of piping between these valves and their respective SI pump discharge maintenance isolation valve (SI-476, SI-478, SI-435, and SI-447) with water, and using either pressure decay or volumetric analysis to determine valve reverse seating function.

(17) Valves: SI-123, SI-143, SI-541, SI-543, SI-168, SI-178

Check valves SI-123, SI-143, SI-541, SI-543, SI-168, and SI-178 are to be provided with sufficient flow to stroke to their full open position. This test flow ultimately passes through the DVI nozzles and into the RCS. Neither the SI nor the SC/CS pump discharge pressures are sufficient to overcome normal RCS operating pressure in order to establish the flow required to perform a partial or full stroke test of these valves. In addition, any flow from safety injection or shutdown cooling to the RCS during power operations would produce an undesirable temperature transient at the DVI nozzles. During cold shutdown, the SI pumps may not be used for stroke testing these valves, because this could result in low temperature overpressurization of the reactor vessel. A full flow stroke test of these valves during cold shutdown is achievable by use of the SC/CS pumps.

Valves SI-123 and SI-143 are not reverse flow tested quarterly, since testing of these valves during power operations would require containment entries by testing personnel to high radiation and airborne contamination areas. These valves are not reverse flow tested every cold shutdown because of the extensive test equipment setup which could extend the cold shutdown. These valves are reverse flow tested during refueling.

(18) Valves: CS-1007, CS-1008

These valves are required to open to pass flow from the containment spray (CS) pumps to the containment atmosphere. These valves cannot be stroked open with CS flow, since this would result in spraying down containment. These valves will be equipped with external means to exercise the valve obturator and to measure the force required to exercise the valve open and closed (this performs both the S and RF test). Since the valves are located in a containment area subject to moderate to high radiation and contamination levels, the valves will be exercised each refueling outage, instead of during cold shutdown or plant operation (every 3 months).

(19) Valves: SI-215, SI-225, SI-235, SI-245

These SIT outlet check valves are to be provided with sufficient flow from the SITs to the RCS to stroke to their full open position. During normal operations, the SI tanks are not capable of providing flow to the RCS, due to RCS pressure and tank pressure limitations. Also, providing flow to the DVI nozzles during plant operations would cause undesirable temperature transients at the DVI nozzles. The SITs may be used, however, to provide flow to partially stroke these valves, with minimal temperature transient impact to the DVI nozzles, when proceeding to or starting up from cold shutdown. In this configuration a full flow stroke test is impractical, due to significant inventory additions to the RCS should the SIT water level become too low. During refueling and with the reactor head removed, full flow testing of these valves is practical. In this condition, SIT flow may be obtained which is sufficient to full stroke the SIT outlet check valve, with minimal risk of injecting nitrogen into the RCS. Therefore, these valves will be partially stroke tested during cold shutdown and full stroke tested during refueling.

Table 3.9-13 (83 of 90)

(20) Valves: SI-217, SI-227, SI-237, SI-247

Providing flow to the DVI nozzles in order to stroke these check valves during plant operation is not practicable, since RCS pressure during normal operations is significantly higher than the discharge pressures of the SI pumps, SC/CS pumps, or SITs. In addition, any flow from these sources to the RCS during power operations would produce an undesirable temperature transient at the DVI nozzles.

Check Valves SI-217 and SI-237:

Full stroke testing of these check valves is not practical at cold shutdown for several reasons. First, SI pumps 4 and 3 are not capable of providing sufficient flow to full stroke their respective DVI check valve (SI-217/SI-237). Secondly, such use of SI pumps during cold shutdown condition is not practical, since it could result in low temperature overpressurization of the reactor vessel. Thirdly, use of water inventory from their respective SITs to full stroke test these check valves during cold shutdown when RCS pressure is low is impractical, because of the risk of injecting nitrogen into the RCS, and because the RCS is not capable of accepting the added SIT inventory from a full stroke test.

However, a partial stroke test of these check valves may be achieved with minimal temperature transient impact to their respective DVI nozzles, when proceeding to or starting up from cold shutdown, by use of water inventory from their respective SITs to establish flow through these valves. During refueling and with the reactor head removed, full flow testing of these valves is practical. In this condition, SIT flow may be obtained which is sufficient to full stroke the respective DVI check valve, with minimal risk of injecting nitrogen into the RCS. Therefore, these valves will be partially stroke tested during cold shutdown and full stroke tested during refueling.

Check Valves SI-227 and SI-247:

These check valves have the same testing limitations as SI-217 and SI-237, above, except that a full stroke test of the check valves is practical during cold shutdown by operating their respective SC/CS pump. SI-227 and SI-247 are tested as cold shutdown valves.

(21) Valves: SI-302, SI-303

Closing of these valves to perform stroke testing renders both SI pumps in the respective valve trains inoperable, which is in violation of Technical Specifications 3.5.2 and 3.5.3. Technical Specification 3.5.3, however, allows inoperability of both train SI pumps during refueling subject to prescribed RCS parameters.

(22) Valves: SI-424, SI-426, SI-448, SI-451

Each of these valves is reverse flow tested by isolating its associated pump and operating the other divisional SI pump in miniflow to provide reverse flow against the tested valve. In this test alignment, the operating pump is to remain in miniflow condition, fully capable of supplying design basis accident flow. Procedural measures are implemented thus so that only one train SI pump will be inoperable (i.e., the isolated SI pump).

Table 3.9-13 (84 of 90)

(23) Valves: CV-363

This check valve functions as a containment isolation valve and isolates the shutdown cooling purification line. During normal operation, CV-363 and manual valve CV-362 are in the closed position to isolate this line. When shutdown cooling purification is used during cold shutdown, these valves are open. To provide reasonable assurance of the operability of CV-363 for its containment isolation function, this valve is reverse flow tested. Reverse flow testing of CV-363 is not practical quarterly, since during unit operation, opening of CV-362 or other venting path could result in an inter-system LOCA.

The appropriate interval for such testing is during cold shutdown when the shutdown purification line is secured prior to unit startup.

(24) Valves: SI-614, SI-624, SI-634, SI-644

These valves are to be open to pass flow from the SITs to the RCS. Technical Specifications do not permit testing these valves during normal operation since all four SITs are to be operable. Normal shutdown/startup procedures require these valves to be closed when proceeding to cold shutdown and to be opened when starting up from cold shutdown. Testing of these valves will be performed at this time.

(25) Valves: SI-651, SI-652, SI-653, SI-654, SI-655, SI-656

These valves are to open to align the SC/CS pump suction to the RCS. These valves are interlocked such that they cannot be opened when RCS pressure is above the operating pressure of the SCS. Therefore, these valves cannot be tested during normal operations. Testing will be performed during cold shutdown when valves can be manipulated.

(26) Valves: VQ-0011, VQ-0012, VQ-0013, VQ-0014

These valves are to close on receipt of a CIAS to perform their containment isolation function. During normal operations, these valves are closed and Technical Specifications do not permit opening. Therefore, these valves will be tested during cold shutdown.

(27) Valves: SD-1115, SD-1116

These valves are to close on reverse flow in the SG wet layup recirculation line to perform their containment isolation function. The recirculation lines are isolated during normal operations and are only used when the SGs are in wet layup conditions such as during cold shutdown. Therefore, these valves will be tested during cold shutdown when the recirculation system is stopped.

(28) Valves: WI-012, WI-013, WI-015

During normal operations, these valves are open provide plant chilled water (PCW) to the containment. Stroke testing of any one of these valves will require interruption of at least one division of PCW to the containment. To maintain containment air temperatures within the 120 ^{o}F Technical Specifications limit year-round, the containment coolers are required to operate with the two units in standby. There may be periods during the year, however, when two out of four containment cooler operations (one PCW division operating, one PCW division secured) provides sufficient cooling to maintain containment temperature within the Technical Specifications limit, due to less severe site climate and heat sink characteristics (e.g., non-summer months). For these periods of the year, the valves will be quarterly stroke tested.

For other periods of the year during which at least two of the four containment coolers are to be kept in operation to maintain containment temperature within the 48.9 °C (120 °F) Technical Specification limit, the valves will be tested as cold shutdown valves.

(29) Valves: FW-1035, FW-1036, FW-1037, FW-1039, FW-1040, FW-1042, FW-1043, FW-1044, FW-1046, FW-1047

These main feedwater system check valves are located on the feedwater inlet lines to the steam generators. These check valves have only a safety function to close. Since these valves are to remain open during power operations to maintain steam generator level and prevent reactor trip and plant shutdown, quarterly reverse flow testing is impractical. As described in Section 10.4.7 and Figure 10.4.7-1, the feedwater split between the economizer feedwater lines and the downcomer feedwater line always maintains some flow through the downcomer feedwater line, even though the two economizer feedwater lines are sized to collectively provide 100 percent required flow to the steam generator they service. The 10 percent of required steam generator flow that passes through the downcomer line at full power operation is used to maintain the downcomer line at a constant temperature to protect the line from thermal transient (water hammer) damage. Thus, the downcomer feedwater line may not be isolated during power operations in order to perform a reverse flow test on its feedwater check valves. Flow testing of these valves is performed on a cold shutdown (CS) frequency basis while the plant is in Mode 3 (hot standby), at which condition adequate steam generator pressure exists to perform reverse flow tests on the valves.

(30) Valves: DS-112, DS-117, DS-118, DS-122, DS-127, DS-128, DS-212, DS-217, DS-218, DS-222, DS-227, DS-228

These valves are tested by isolating one of the starting air receiver tanks and starting the diesel generator using the remaining operational tank to direct full flow through the tested valves. In addition to the diesel generator starting using the one air receiver tank, positive means in accordance with KEPIC MOC are provided to verify conclusively that check valves DS-117, DS-118, DS-127, DS-128, DS-217, DS-218, DS-227, and DS-228 stroke open. Positive means are also employed to verify reverse flow seating in accordance with KEPIC MOC for check valves DS-112, DS-117, DS-118, DS-122, DS-127, DS-128, DS-212, DS-217, DS-218, DS-222, DS-227, and DS-228.

Table 3.9-13 (86 of 90)

(31) Valves: SI-605, SI-606, SI-607, SI-608, SI-613, SI-623, SI-633, SI-634

These solenoid valves are both stroke tested (S) and fail-safe tested (FS) during cold shutdown because opening any of these valves will result in depressurizing the affected SIT, thus causing the SIT to be inoperable. These valves cannot be tested during plant operations, since plant Technical Specification 3.5.1 requires all SITs to remain operable in Mode 1 (power operations). Technical Specification LCO 3.5.1 (required action) for inoperability of any SIT requires restoration of that SIT to operable status in one hour, or commence unit shutdown. Since this LCO is too stringent to allow valve stroke or fail-safe testing of these valves during plant operations, this testing will be performed during cold shutdown.

(32) Valves: FP-0030, FP-1440

The safety function of valves FP-0030 and FP-1440 in the forward stroke direction is to relieve thermal pressure to the containment fire water supply piping and thus prevent damage to the containment penetration as a result of containment heatup following a LOCA. It is impractical to perform a forward stroke test for check valves FP-0030 and FP-1440 during power operations or cold shutdown for several reasons: significant radiation and contamination exposure to test personnel in containment, the necessity of disabling the sprinkler system within containment to perform the test which jeopardizes system response to containment fire, and the extensive restoration/draining of the fire supply headers inside of containment post-testing to their normal “dry” status which would result in extending the cold shutdown.

The reverse flow safety function is containment isolation. Reverse flow testing of these check valves is impractical during power operations or cold shutdown for several reasons: significant radiation and contamination exposure to test personnel in containment, the necessity of disabling the sprinkler systems to fill the “dry” fire water supply piping in the reverse flow test volume in order to establish backpressure on the check valve seat, which jeopardizes system response to containment fire, and the extensive restoration/draining of the fire supply headers inside of containment post-testing to their normal “dry” status, which would result in extending the cold shutdown.

(33) Valves: WM-1752

The safety function of valve WM-1752 in the forward stroke direction is to relieve thermal pressure to the containment demineralized water piping as a result of containment heatup following a LOCA. Verification of this safety function requires forward stroke testing, and use of demineralized water within containment. However, during power operations and cold shutdown, there are no users of demineralized water within containment to establish this flow, without necessitating containment entry to areas of high radiation dose and airborne contamination present during power operations and cold shutdown to manipulate manual valves at decontamination sinks, etc. The forward stroke test will be performed during refueling for ALARA purposes. Similarly, the RF test will require containment entry to areas of high radiation dose and airborne contamination present during power operations and cold shutdown. For ALARA purposes, this test will be performed during refueling.

Table 3.9-13 (87 of 90)

(34) Valves: IW-0001, IW-0002, IW-0003, IW-0004

Valves IW-0001 and IW-0002 are motor-operated holdup volume tank (HVT) flooding valves; valves IW-0003 and IW-0004 are reactor cavity flooding valves. The valves are normally closed and remain closed throughout the recovery period of any design basis accident. The valves are opened only for a severe accident, which requires flooding of the reactor cavity in the event of the reactor vessel breach. The opening of the valves allows water to flow from the IRWST to the reactor cavity to cover core debris. Operability of the valves is not required for shutting down the reactor, maintaining cold shutdown, or mitigating the consequences of any design basis accident.

Testing of the HVT flooding valves requires that the manual valves located upstream be closed to prevent the flow of water from the IRWST to the HVT. Closing the manual valves is not practical during operations at power because containment entry would be required. The reactor cavity flooding valves are not tested during operations at power because, in the event of a design basis accident, the failure of the valves in the open position would provide an open flow path from the HVT to the reactor cavity. This would adversely affect the operability of the IRWST in mitigating the consequences of a design basis accident; operability of the IRWST is required for all plant operating modes from power operation through, and including, cold shutdown. The HVT flooding valves and the reactor cavity flooding valves will be tested during each refueling outage. This will limit personnel radiation exposure and minimize the potential impact on the operability of the IRWST.

(35) Valves: SI-612, SI-619, SI-622, SI-629, SI-632, SI-639, SI-642, SI-649

These air-operated SIT nitrogen pressure control valves are stroked in the course of plant operation as a matter of normal operation and pressure control of the SITs at a frequency that satisfies test requirements of quarterly testing. Fail-safe (FS) actuation on a 3-month basis, however, is impractical during plant operations (quarterly test frequency) or cold shutdown because such testing involves entries to containment to proximity of the SITs (high radiation dose and airborne contamination area) to fail air to the air diaphragm valve actuators. Therefore, the FS test for these valves will be performed on a refueling outage basis for ALARA purposes.

(36) Valves: SI-322, SI-332, SI-611, SI-618, SI-621, SI-628, SI-631, SI-638, SI-648, SI-661, SI-670

These air-operated valves are stroked on a quarterly frequency. Fail-safe (FS) actuation testing on a 3-month basis, however, is impractical during plant operations (quarterly test frequency) or cold shutdown because such testing involves entries to containment to proximity of the SITs (high radiation dose and airborne contamination area) to fail air to the air diaphragm valve actuators. Therefore, the FS test for these valves will be performed on a refueling outage basis for ALARA purposes.

(37) Although these emergency diesel generator support system components are Safety Class 3, they are procured, tested, and maintained as part of the emergency diesel generators themselves, which are tested for operability and reliability by the plant Technical Specifications. Therefore, these components are tested by Technical Specifications Surveillance Requirements of Technical Specification Section 3.8.

Table 3.9-13 (88 of 90)

(38) Pressure isolation valves (PIVs) are not reverse flow tested quarterly, since testing of these valves during power operation would require containment entries to high radiation and airborne contamination areas. PIVs are not reverse flow tested every cold shutdown, because of the extensive test equipment setup, which could extend the cold shutdown. The RF function is verified, however, by leakage testing each valve in the reverse flow direction during unit startup for the testing frequency outlined in Technical Specification Surveillance Requirement 3.4.13.1. This surveillance requirement states that leakage testing of these valves is required every 18 months and prior to entering Mode 2 whenever the plant has been in Mode 5 (cold shutdown) for 7 days or more, if leakage testing has not been performed in the previous 9 months and within 24 hours following valve actuation due to automatic or manual action or flow through the valve(s).

(39) Inservice Testing/Monitoring for Valves on Piping Connected to the Steam Generator Secondary Side

Steam generator (SG) main steam isolation valves and SG main steam isolation bypass valves are tested for gross leakage each refueling outage. Testing is performed by isolating these valves with the steam generators under steam pressures created by normal startup/shutdown and measuring downstream steam header pressure and temperature.

SG main steam safety valves are tested each refueling outage for gross leakage by means of walkdown/temperature/acoustic monitoring with main steam lines pressurized.

SG atmospheric dump valves (ADV) are tested for gross leakage each refueling outage by temperature/acoustic monitoring of the ADV lines downstream of the ADVs with main steam lines pressurized.

Steam generator blowdown valves are tested for gross leakage each refueling outage. Testing is performed by isolating these valves individually against steam generator pressure and then monitoring the steam generator blowdown tank for an increase in tank level, which would be indicative of gross valve leakage.

Steam generator sampling line valves are tested for gross leakage each refueling outage. Testing is performed by isolating these valves individually against steam generator pressure and then monitoring sample line flow for gross valve leakage.

Main feedwater containment isolation valves are tested for gross leakage each refueling outage. Testing is performed by individually subjecting these valves to steam generator pressure experienced during unit startup/shutdown, and then measuring resultant valve leakage through the provided test connection. The startup feedwater pump may be used for establishing and maintaining steam generator inventory for this gross leakage test.

Table 3.9-13 (89 of 90)

Auxiliary feedwater (AF) containment isolation valves are tested for gross leakage each refueling outage. The AF isolation check valves are leakage tested by individually subjecting these valves to steam generator pressures experienced during unit startup/shutdown and then measuring resultant valve leakage through the provided test connection. The outside-containment AF isolation valves are leakage tested by pressurizing the piping between these valves and their inside-containment containment isolation check valves while the steam generators are at startup/shutdown pressures. Valve leakage is then measured through the provided test connection. These AF valves also employ installed temperature instrumentation to detect leakage past these valves.

(40) Safety Injection System

For inservice testing of the safety injection pumps during refueling outages, a walkdown visual examination of safety injection system piping and components outside containment will be conducted to verify the leak tight integrity of the system.

(41) Valves: CV-575, CV-576, CV-577

These valves limit charging flow to RCS. It is not practical to test these values during normal operations. In addition, failure of these values in the closed position could result in a loss of pressurizer level control. Therefore, these valves will be tested during cold shutdown.

(42) Valve: PX-1005

This valve is installed to isolate the containment building and protect the overpressure of sample collecting piping of the post-accident primary sampling system (Figure 9.3.2-1). PX-1005 is closed for normal plant operating, and the right-direction and reverse-direction stroke test of PX-1005 is operated by using test-fitting. The right-direction and reverse-direction stroke test of PX-1005 is not operated every quarter because an operator has to enter the high radiation and radiation contamination air-particle areas in containment building during power operations. This valve is not tested during cold shutdown because cold shutdown can be extended by the installation of test equipment but tested during refueling operation.

(43) Valve: PX-1020

This valve is installed to isolate the containment building of sample collecting piping of containment atmosphere (Figure 9.3.2-1). PX-1020 is closed for normal plant operation, and the right-direction and reverse-direction stroke test of PX-1020 is operated by using test-fitting. The right-direction and reverse-direction stroke test of PX-1005 is not operated every quarter because an operator has to enter the high radiation and radiation contamination air-particle areas in containment building during power operations. This valve is not tested during cold shutdown because cold shutdown can be extended by the installation of test equipment but tested during refueling operation.

Table 3.9-13 (90 of 90)

(44) Valves: AF-1003A, AF-1003B, AF-1004A, AF-1004B, AF-1007A, AF-1007B, AF-1008A, AF-1008B

When auxiliary feedwater (AF) system operation is required, these valves are to open to provide flow to the steam generators (SGs).

Testing of these valves requires AF injection into the SGs, which is not practical during normal operations due to the effects of thermal shock to the SG feedwater nozzles and potential overcooling of the RCS. Testing during cold shutdown is not desirable because the SG is in wet layup conditions. Therefore, these valves will be tested following cold shutdown prior to entering Mode 2 which allows normal SG water levels to be established and the system aligned for standby readiness.

Reverse flow for reverse flow testing of valves AF-1004A, AF-1004B, AF1003A, and AF-1003B is obtained from operating the motor-driven AF pump in the opposite division as the tested valve and manipulating manual crossover valves AF-2330D, AF-2330C, AF-2330A, and AF-2330B, as appropriate, to align flow from the motor driven AF pump to the valve under test.

Reverse flow for reverse flow testing of valves AF-1008A, AF-1008B, AF-1007A, and AF-1007B is obtained by opening AF pump to AF isolation valves AF-0043, AF-0044, AF-0045, and AF-0046, respectively.

(45) Valves: AF-0043, AF-0044, AF-0045, AF-0046

Gate valves AF-0043, AF-0044, AF-0045, and AF-0046, similarly provide the high/low temperature interfaces between main feedwater system piping (design temperature: 570 °F) and the auxiliary feedwater system piping (design temperature: 140 °F), as illustrated in Figure 10.4.9-1, Sheet 1. These valves remain open during plant operation, but in the case of the backleakage into the auxiliary feedwater system piping, which can result in exceeding piping design temperature and steam binding of the auxiliary feedwater pumps (Re: Response to Generic Safety Issue 093), these valves close and leakage through check valves is tested. Dedicated temperature monitors, which alarm to the control room, are located upstream of these valves and are used to detect any high temperature backleakage from the main feedwater system. The setpoints of these temperature monitors and associated alarms will be such that the design temperature limit of the interfacing auxiliary feedwater system piping will not be exceeded prior to the initiation of the alarm.

Table 3.9-14

RCS Pressure Isolation Valves Associated with the Reactor Coolant System

Detailed testing information is contained in Table 3.9-13 and Technical Specification Surveillance Requirement 3.4.13.1.

Valve	Description
SI-215	SI tank 4 discharge check
SI-217	DVI nozzle 1B check valve
SI-225	SI tank 2 discharge check
SI-227	DVI nozzle 2B check valve
SI-235	SI tank 3 discharge check
SI-237	DVI nozzle 2A check valve
SI-245	SI tank 1 discharge check
SI-247	DVI nozzle 1A check valve
SI-322	Hot leg injection 1 bleed off isolation
SI-332	Hot leg injection 2 bleed off isolation
SI-522	Hot leg injection loop 1 check
SI-523	Hot leg injection loop 1 check
SI-532	Hot leg injection loop 2 check
SI-533	Hot leg injection loop 2 check
SI-540	SI pump #4 discharge check
SI-541	SI pump #2 discharge check
SI-542	SI pump #3 discharge check
SI-543	SI pump #1 discharge check
SI-618	SI line 4 leakage return
SI-628	SI line 2 leakage return
SI-638	SI line 3 leakage return
SI-648	SI line 1 leakage return
SI-651	SC pump #1 suction
SI-652	SC pump #2 suction
SI-653	SC pump #1 suction
SI-654	SC pump #2 suction

Table 3.9-15 (1 of 2)

Arrangement Comparison of Reactor Internals

Palo Verde	APR1400
Core Support Barrel	
A. Right circular cylinder, three sections, supported by heavy external flange top end, heavy internal flange bottom end.	A. Right circular cylinder, three sections, supported by heavy external flange top end, heavy internal flange bottom end.
B. Two outlet nozzles through barrel.	B. Two outlet nozzles through barrel.
C. Top flange seats on reactor vessel ledge, and has four alignment keys attached.	C. Top flange seats on reactor vessel ledge, and has four alignment keys attached.
D. Bottom flange supports lower support structure, fuel and core shroud.	D. Bottom flange supports lower support structure, fuel and core shroud.
E. Top flange supports upper guide structure assembly.	E. Top flange supports upper guide structure assembly.
F. Six amplitude-limiting devices (snubbers) attached to lower barrel.	F. Six amplitude-limiting devices (snubbers) attached to lower barrel.
Lower Support Structure	
A. Made up of interlocked grid beams with surrounding short cylinder and perforated bottom plates attached to the bottoms of the beams.	A. Made up of interlocked grid beams with surrounding short cylinder and perforated bottom plates attached to the bottom of the beams.
B. Fuel support pins are attached to the top end of the grid beams.	B. Fuel support pins are attached to the top end of the grid beams.
C. The core shroud assembly is attached to the top of the LSS cylinder.	C. The core shroud assembly is attached to the top of the LSS cylinder.
D. The LSS cylinder rests on and is attached to the CSB bottom (internal) flange.	D. The LSS cylinder rests on and is attached to the CSB bottom (internal) flange.

Table 3.9-15 (2 of 2)

Palo Verde	APR1400
Upper Guide Structure	
A. Right circular cylinder supported by a heavy external flange type end, and heavy plate attached to bottom end.	A. Right circular cylinder supported by a heavy external flange top end, and heavy plate attached to bottom end.
B. Heavy plate in A is perforated with flow holes and guide tubes.	B. Heavy plate in A is perforated with flow holes and guide tubes.
C. A second heavy plate supports the bottom ends of the guide tubes, is also perforated.	C. A second heavy plate supports the bottom ends of the guide tubes, is also perforated.
D. This second heavy plate C engages guide lugs on the core shroud.	D. This second heavy plate C engages guide lugs on the core shroud.
E. The guide tubes are welded to the two heavy plates above.	E. The guide tubes are welded to the two heavy plates above.
CEA Shroud Assembly / Inner Barrel Assembly	
A. A series of large-diameter tubes are connected by full-length webs	A. A series of large-diameter tubes are connected by full-length webs.
B. All welded construction.	B. All welded construction.
C. Blank	C. The shroud tube and web assembly is connected to an external cylinder.
D. The tube and web assembly is supported by the UGS support plate via tie rods. Incorporates four (4) interlocking snubbers to the upper UGS.	D. The tube, web, and cylinder assembly is supported by the UGS upper flange.
E. Material – austenitic stainless steel	E. Material – austenitic stainless steel

Table 3.9-16

Nominal Dimensional Comparison of Reactor Internals

Component	Palo Verde	APR1400
Core Support Barrel		
Length, mm (in)	9,734.6 (383-1/4)	9,715.5 (382-1/2)
Diameter (ID), mm (in)	3,987.8 (157)	3,987.8 (157)
Thickness upper, mm (in)	76.2 (3)	76.2 (3)
Thickness middle, mm (in)	66.7 (2-5/8)	66.7 (2-5/8)
Thickness lower, mm (in)	76.2 (3)	76.2 (3)
Outlet nozzles, (qty)	2	2
Outlet nozzle diameter (ID), mm (in)	1,184.3 (46-5/8)	1,184.3 (46-5/8)
Snubbers, (qty)	6	6
Lower Support Structure		
Cylinder height, mm (in)	412.8 (16-1/4)	412.8 (16-1/4)
Cylinder diameter (OD), mm (in)	3,970.3 (156-5/16)	3,970.3 (156-5/16)
Main beams, (Qty)	16	16
Beam thickness, mm (in)	44.5 (1-3/4)	44.5 (1-3/4)
Beam height, mm (in)	669.9 (26-3/8)	669.9 (26-3/8)
UGS Support Barrel Assembly		
Length, mm (in)	4,924.4 (193-7/8)	4,924.4 (193-7/8)
Diameter flange (OD), mm (in)	4,559.3 (179-1/2)	4,559.3 (179-1/2)
Diameter barrel (OD), mm (in)	3,962.4 (156)	3,962.4 (156)
Barrel thickness, mm (in)	76.2 (3)	76.2 (3)
CEA Guide Tubes, (Qty)	804	820
Fuel alignment plate diameter, mm (in)	3,962.4 (156)	3,962.4 (156)
Plate thickness, mm (in)	114.3 (4-1/2)	139.7 (5-1/2)

Table 3.9-17

Comparison of Operating Condition for Reactor Internals

Operating Condition	Palo Verde	APR1400
RV design pressure, kg/cm^{2}A (psia)	175.8 (2,500)	175.8 (2,500)
RV normal operating pressure, kg/cm^{2}A (psia)	158.2 (2,250)	158.2 (2,250)
Normal operating coolant inlet temperature, °C (°F)	296.1 (565)	290.6 (555)
Normal operating coolant outlet temperature, °C (°F)	327.3 (621.2)	323.9 (615)
Design temperature, °C (°F)	343.3 (650)	343.3 (650)
Normal operating coolant flow rate, L/min (gpm)	1,687,000 (445,600)	1,689,000 (446,300)
Mechanical design coolant flow rate (with core), L/min (gpm)	1,965,000 (519,124)	1,943,000 (513,200)
Mechanical design coolant flow rate (without core), L/min (gpm)	2,066,000 (545,860)	2,112,000 (557,900)
Reactor coolant pump frequency (Hz)	20	20
Reactor coolant pump blade passing frequency (Hz)	120	120

Table 3.9-18 (1 of 2)

List of Active Pumps

Pump	ASME Class	System (subsection)
Safety Injection Pump 1	2	6.3.2
Safety Injection Pump 2	2	6.3.2
Safety Injection Pump 3	2	6.3.2
Safety Injection Pump 4	2	6.3.2
Shutdown Cooling Pump 1	2	6.3.2
Shutdown Cooling Pump 2	2	6.3.2
SFP cooling pump 1	3	9.1.3
SFP cooling pump 2	3	9.1.3
CCW pump 1A	3	9.2.2
CCW pump 1B	3	9.2.2
CCW pump 2A	3	9.2.2
CCW pump 2B	3	9.2.2
CCW makeup pump 3A	3	9.2.2
CCW makeup pump 3B	3	9.2.2
ESW pump 1A	3	9.2.2
ESW pump 1B	3	9.2.2
ESW pump 2A	3	9.2.2
ESW pump 2B	3	9.2.2
ECW pump PP01A	3	9.2.7
ECW pump PP02A	3	9.2.7
ECW makeup pump PP03A	3	9.2.7
ECW pump PP01B	3	9.2.7
ECW pump PP02B	3	9.2.7
ECW makeup pump PP03B	3	9.2.7

Table 3.9-18 (2 of 2)

Pump	ASME Class	System (subsection)
Essential chilled water pump PP01A	3	9.2.7
Essential chilled water pump PP02A	3	9.2.7
Essential chilled water makeup pump PP03A	3	9.2.7
Essential chilled water pump PP01B	3	9.2.7
Essential chilled water pump PP02B	3	9.2.7
Essential chilled water makeup pump PP03B	3	9.2.7
DG 1 fuel oil transfer pump 01A/02A	3	9.5.5
DG 1 fuel oil transfer pump 01B/02B	3	9.5.5
MD AFW pump PP02A	3	10.4.9
TD AFW pump PP01A	3	10.4.9
MD AFW pump PP02B	3	10.4.9
TD AFW pump PP01B	3	10.4.9

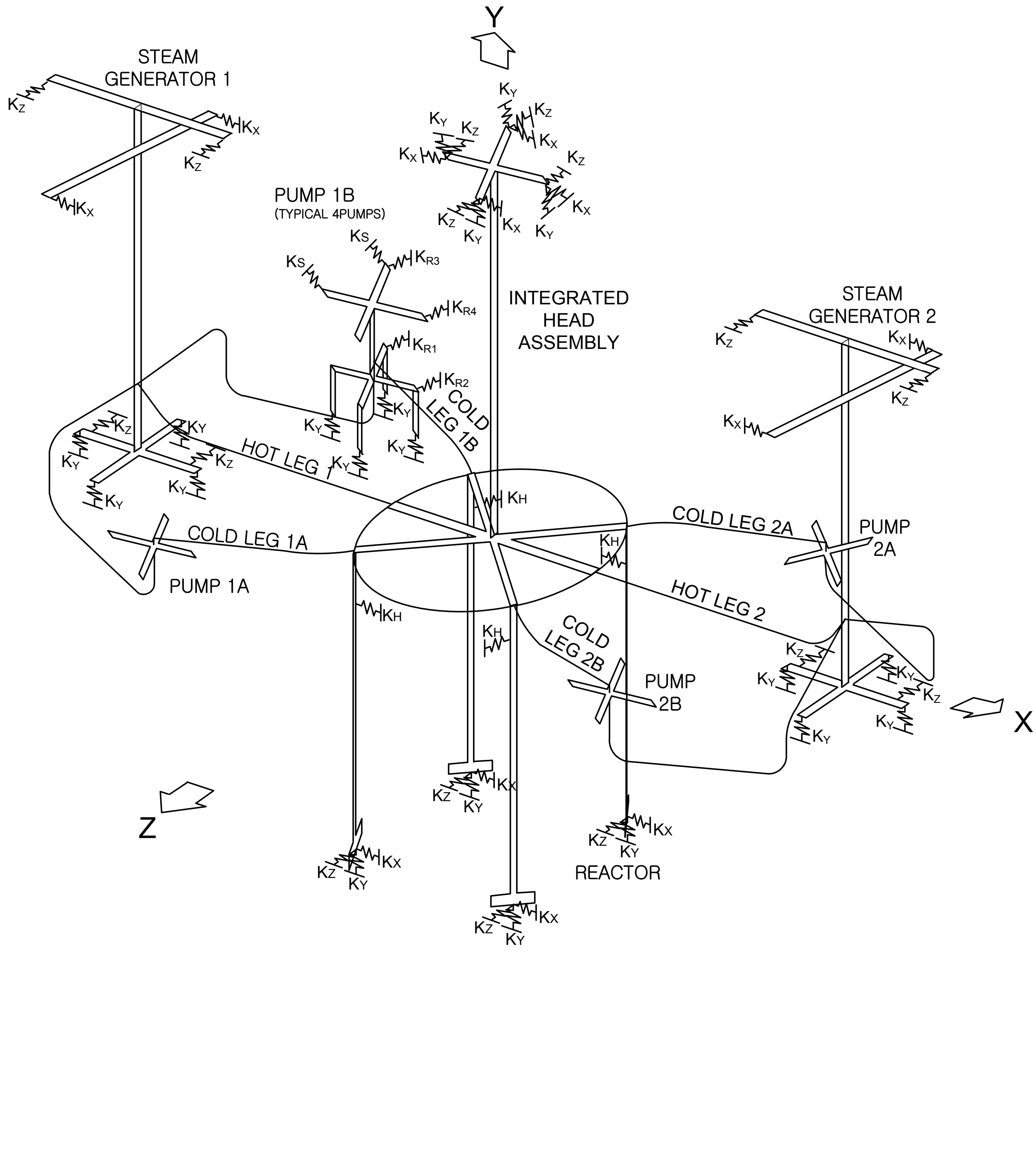


Figure 3.9-1 Reactor Coolant System Supports Diagram

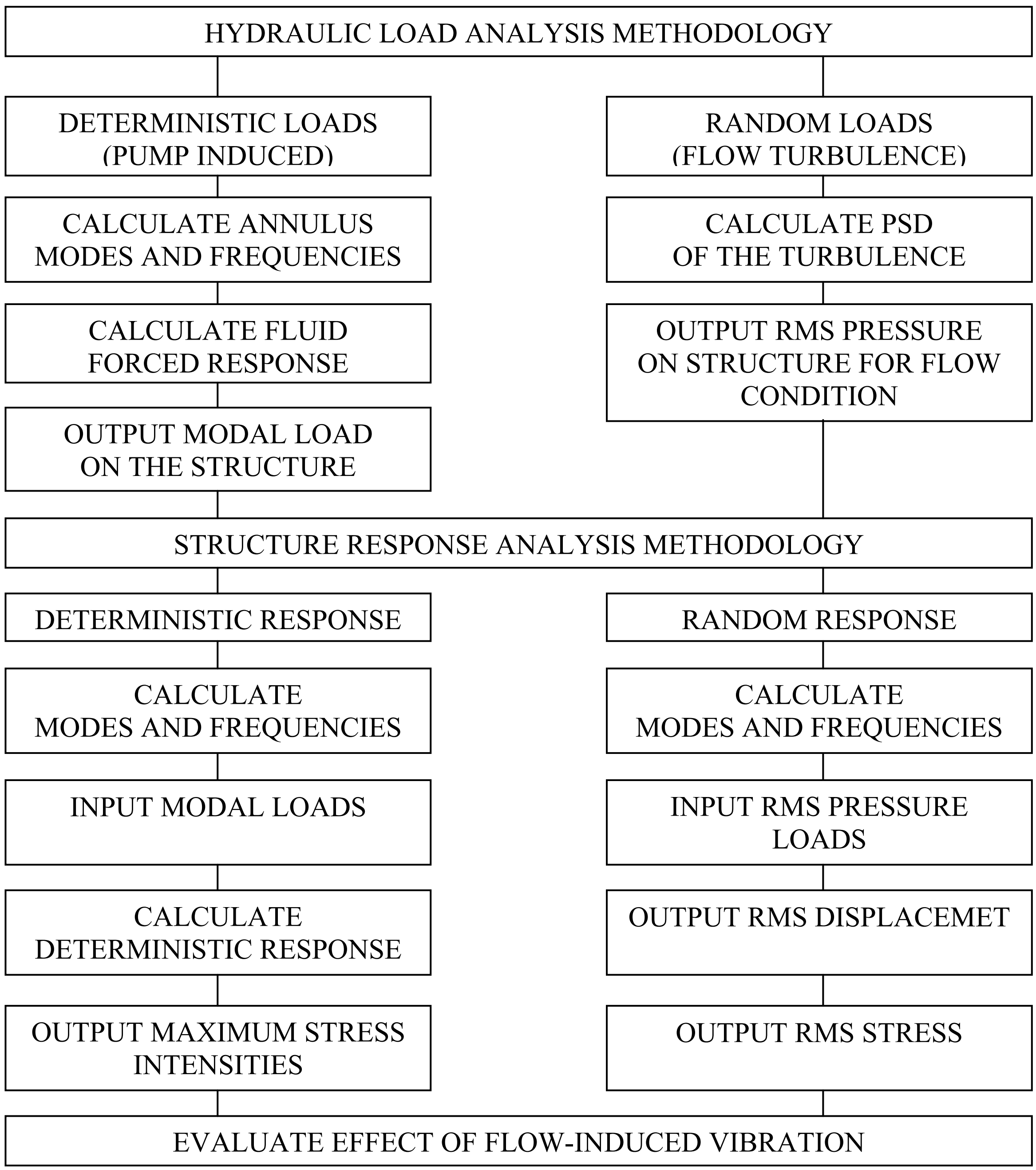


Figure 3.9-2 Summary of Analytical Methodology

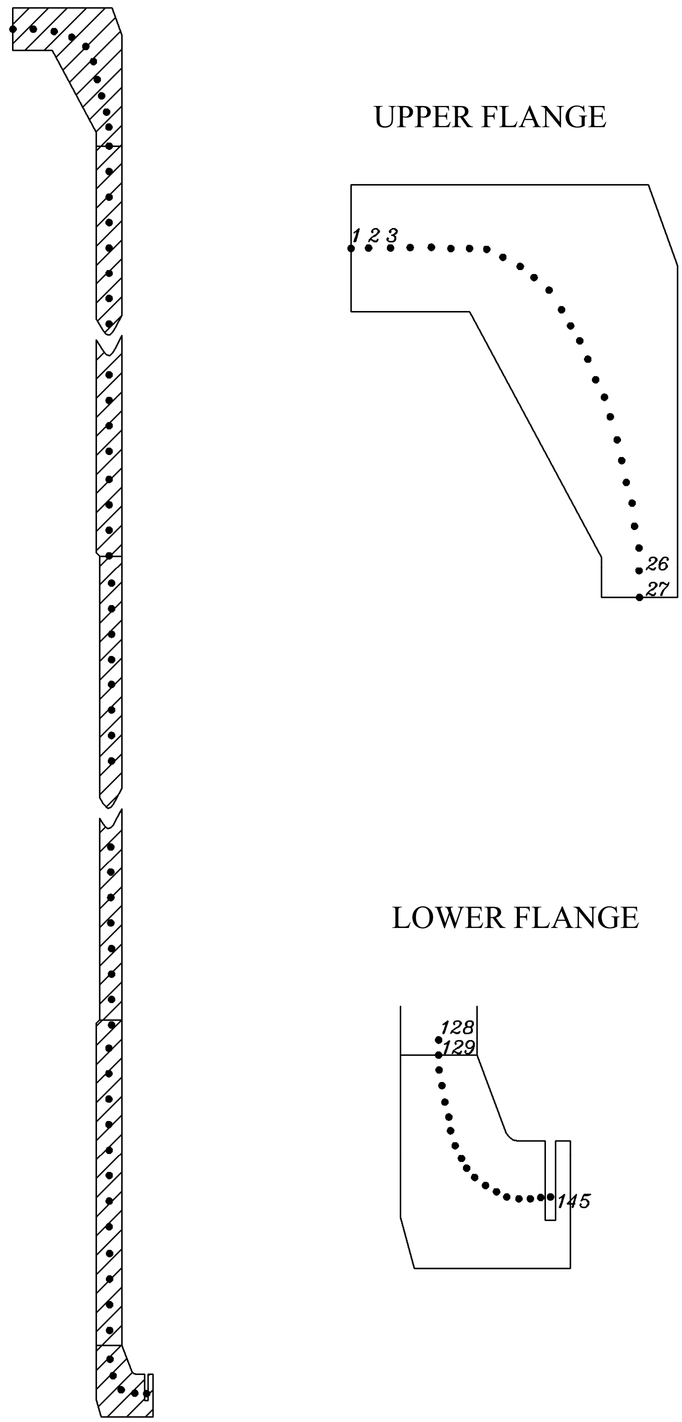


Figure 3.9-3 ASHSD Finite Element Model of the Core Support Barrel

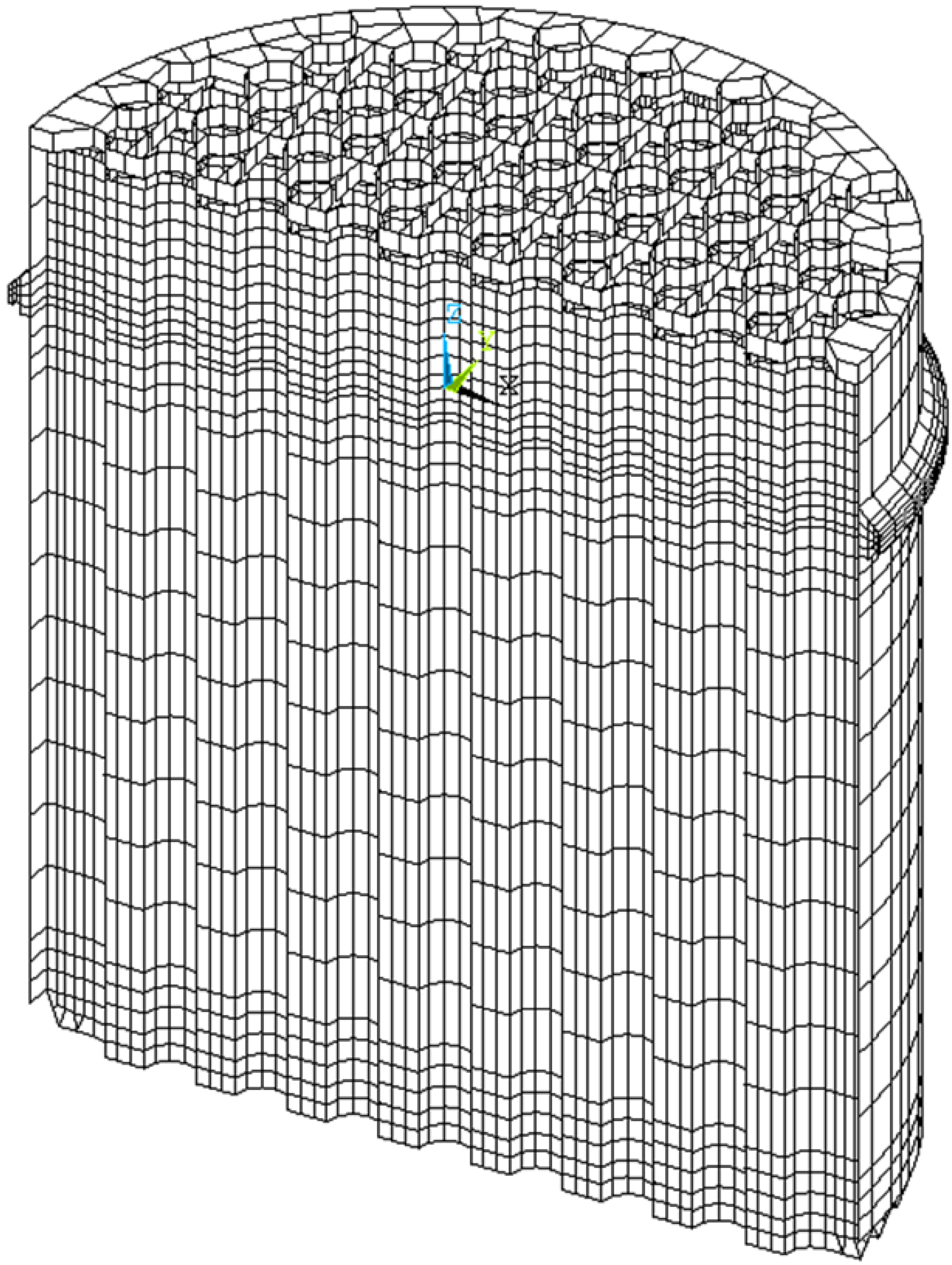


Figure 3.9-4 Inner Barrel Assembly of Finite Element Model

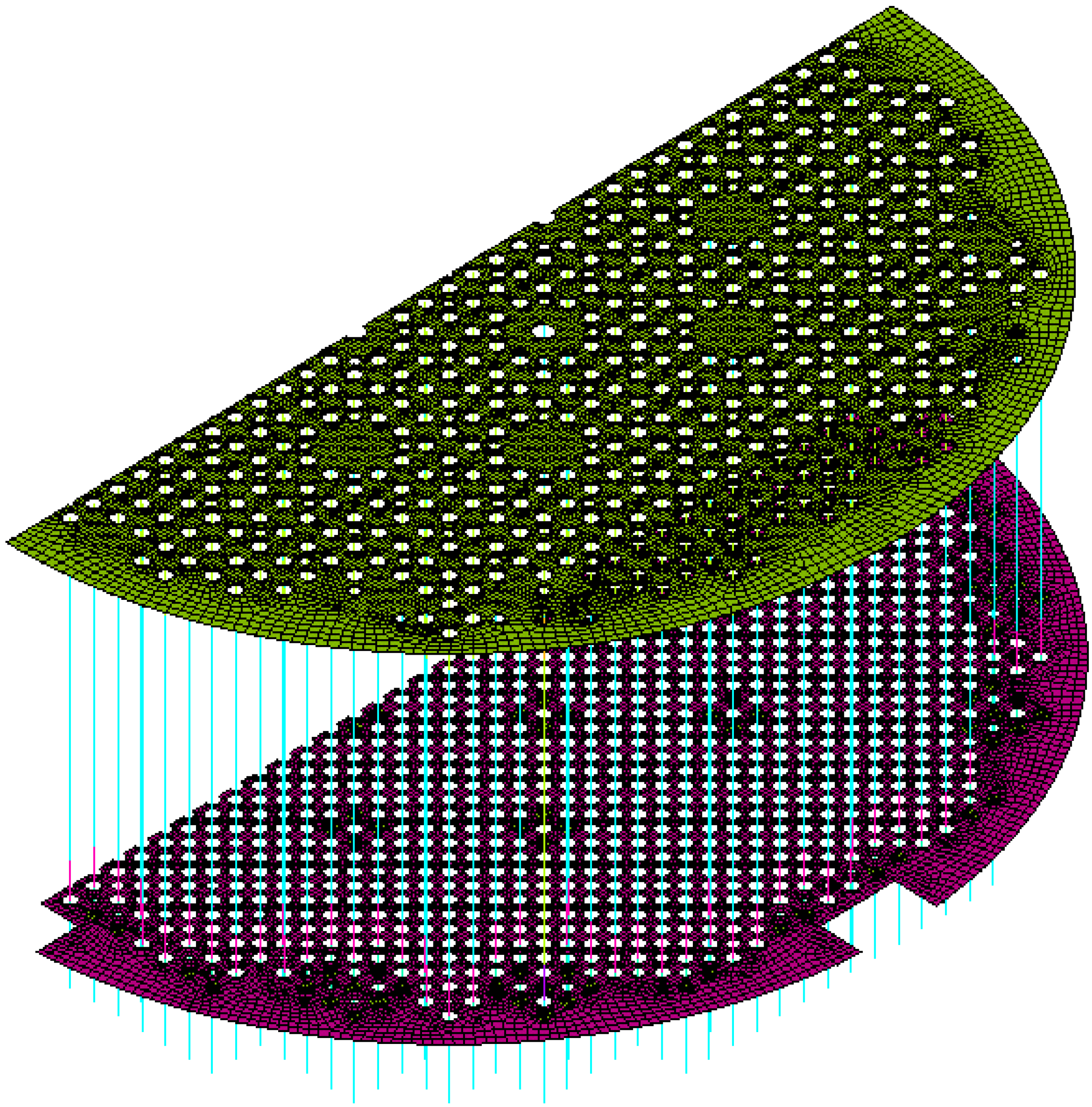


Figure 3.9-5   Finite Element Model of the CEA Guide Tubes

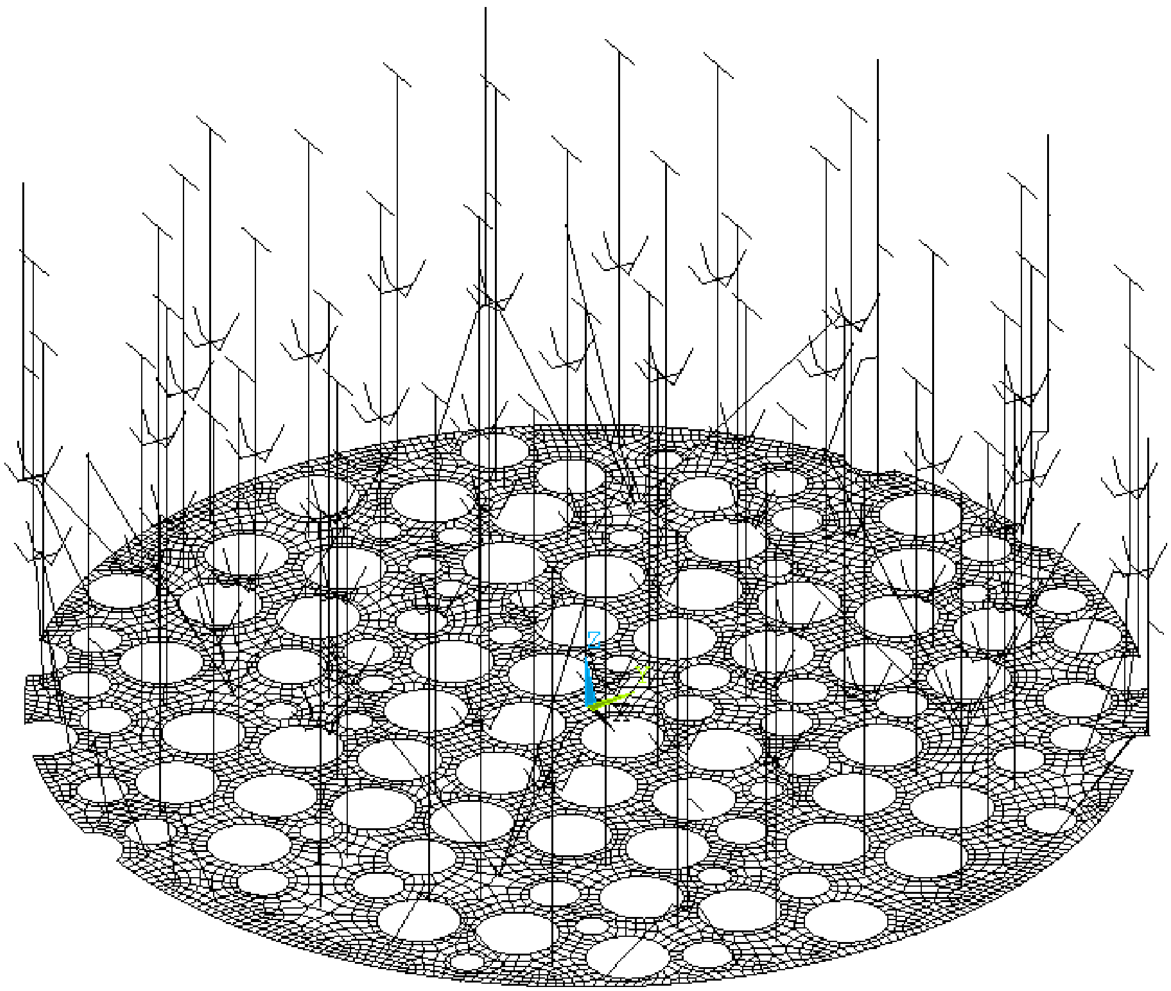


Figure 3.9-6 Lower Support Structure / In-Core Instrumentation Nozzle Assembly Finite Element Model

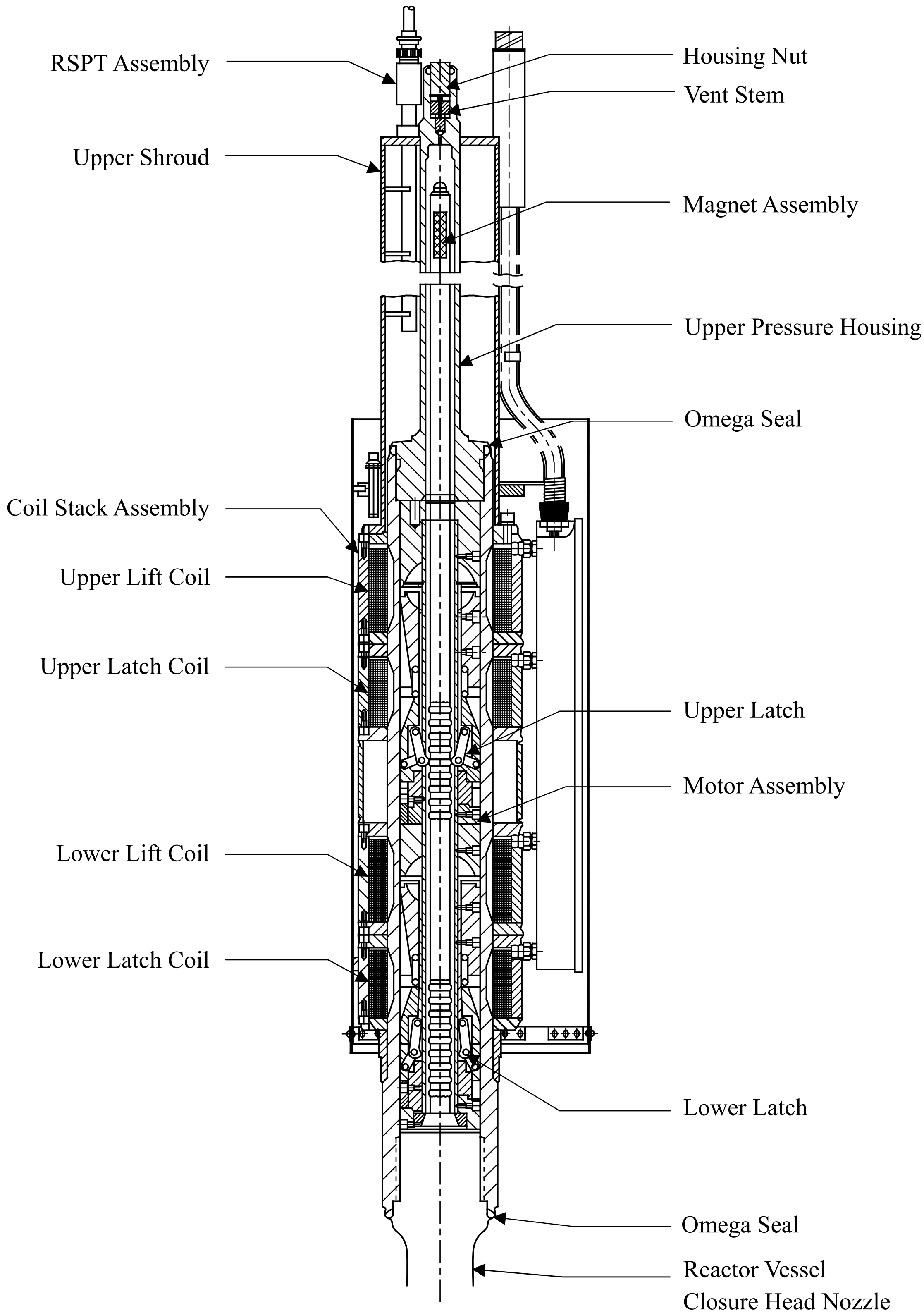


Figure 3.9-7 Control Element Drive Mechanism

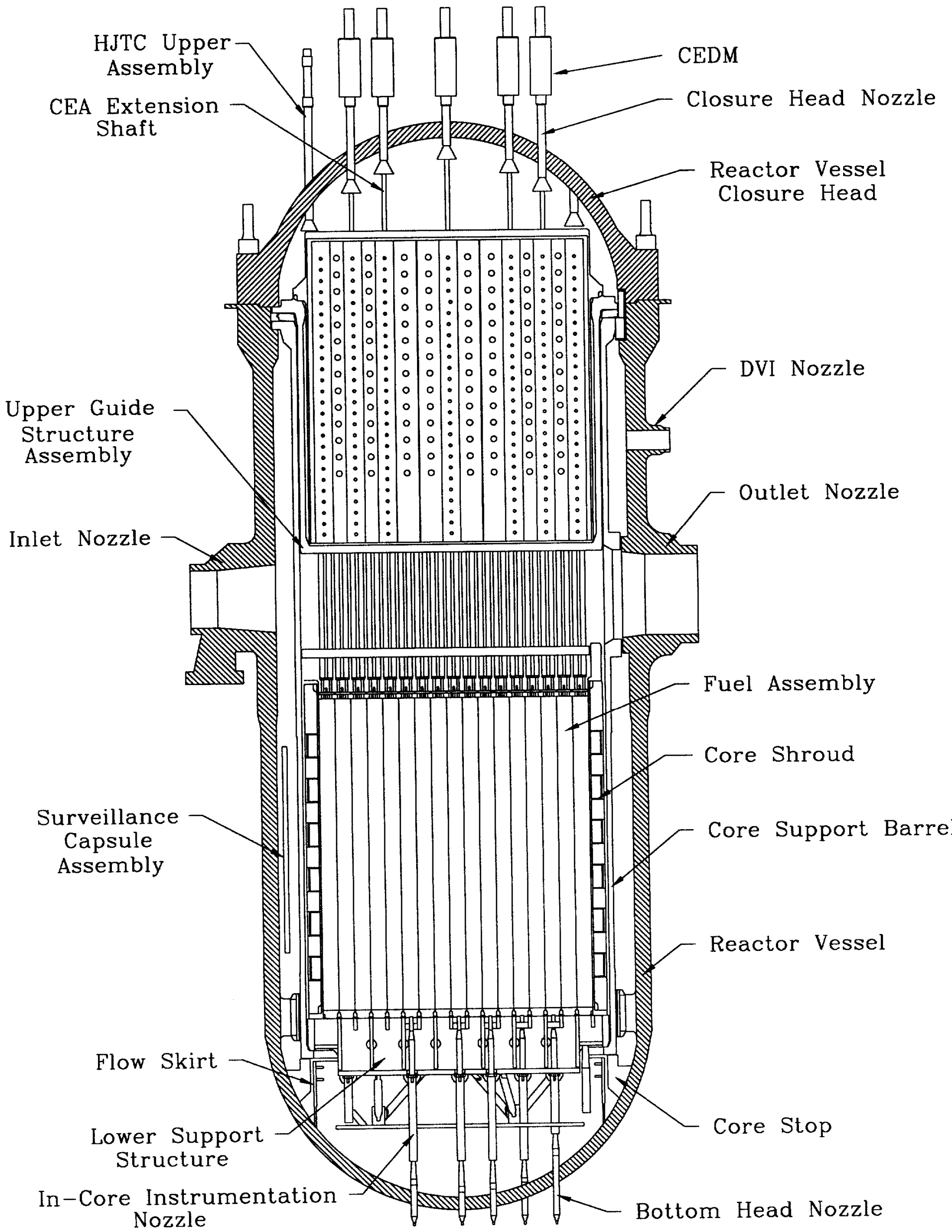


Figure 3.9-8 Reactor Internals Arrangement

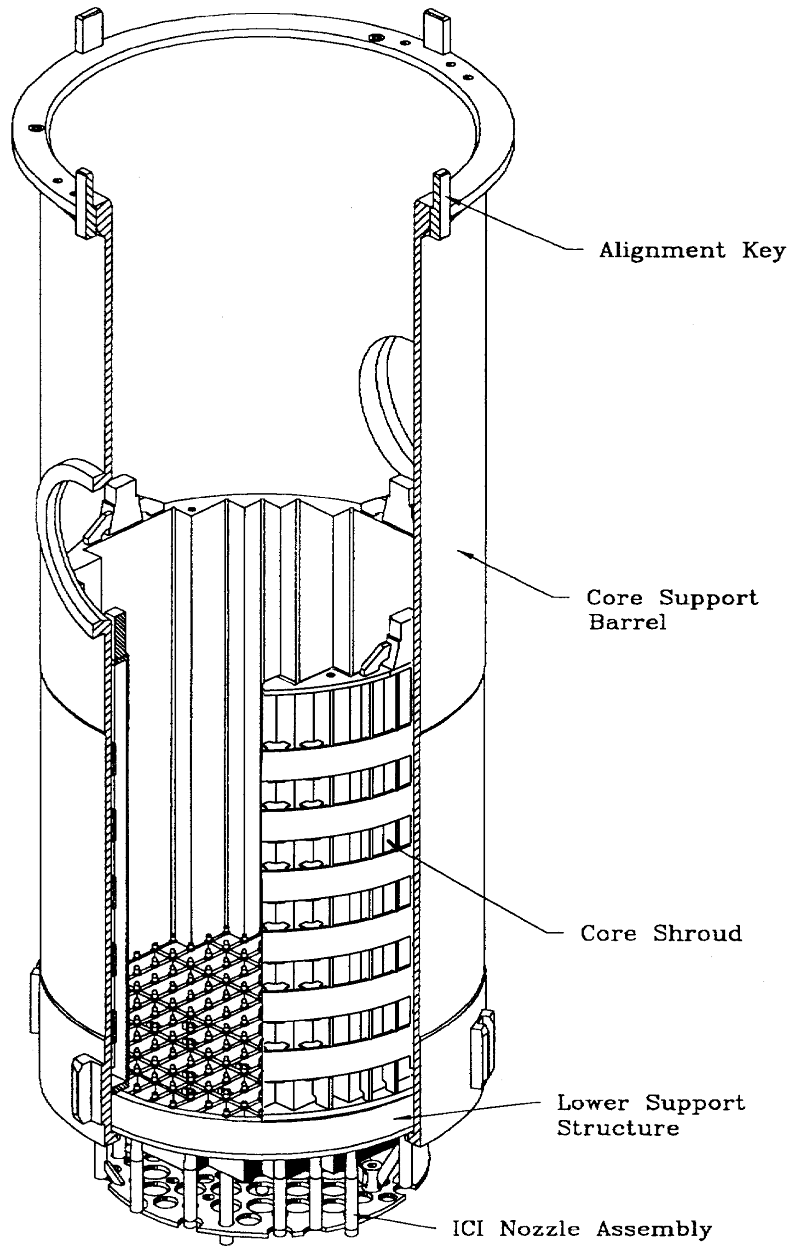


Figure 3.9-9 Core Support Barrel Assembly

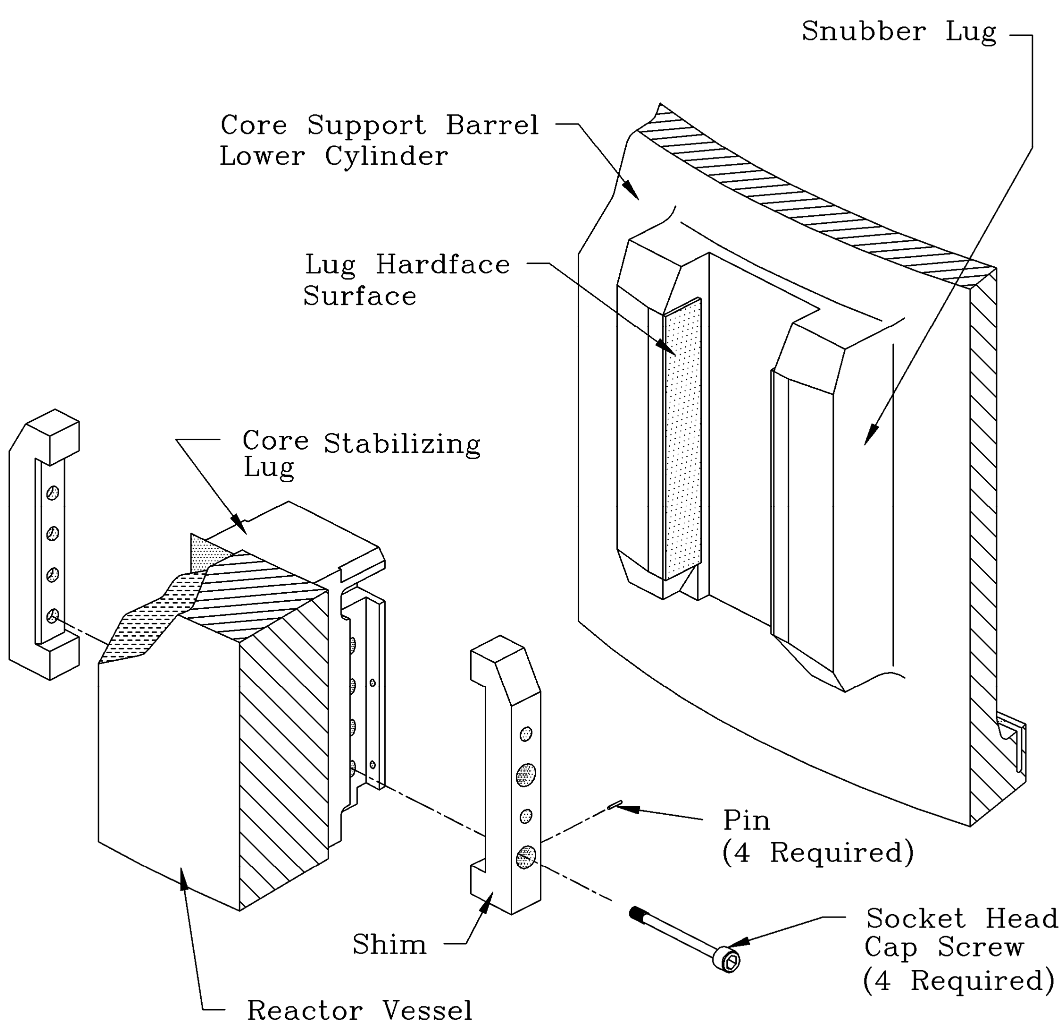


Figure 3.9-10 Reactor Vessel / Core Support Barrel Snubber Assembly

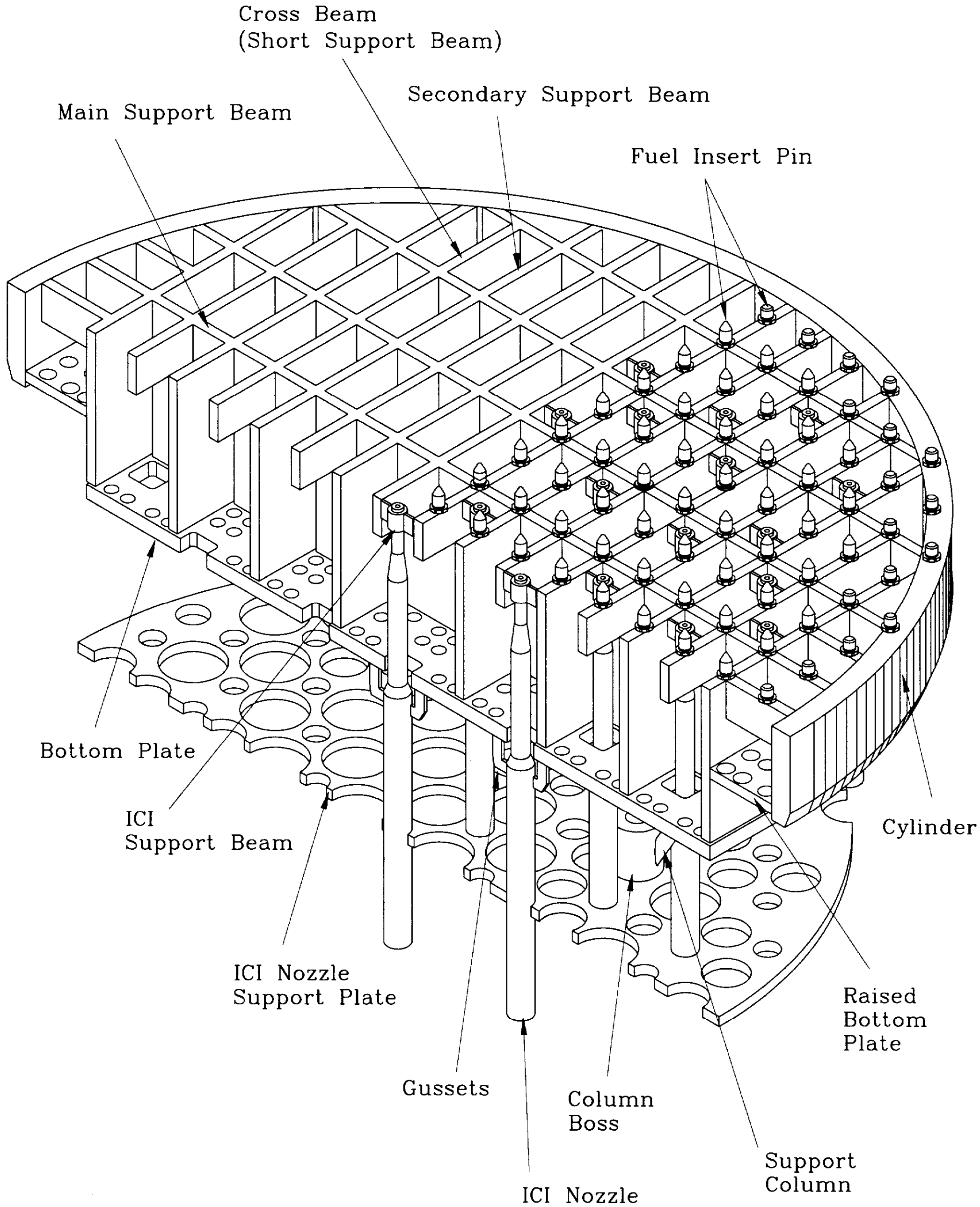


Figure 3.9-11 Lower Support Structure /ICI Nozzle Assembly

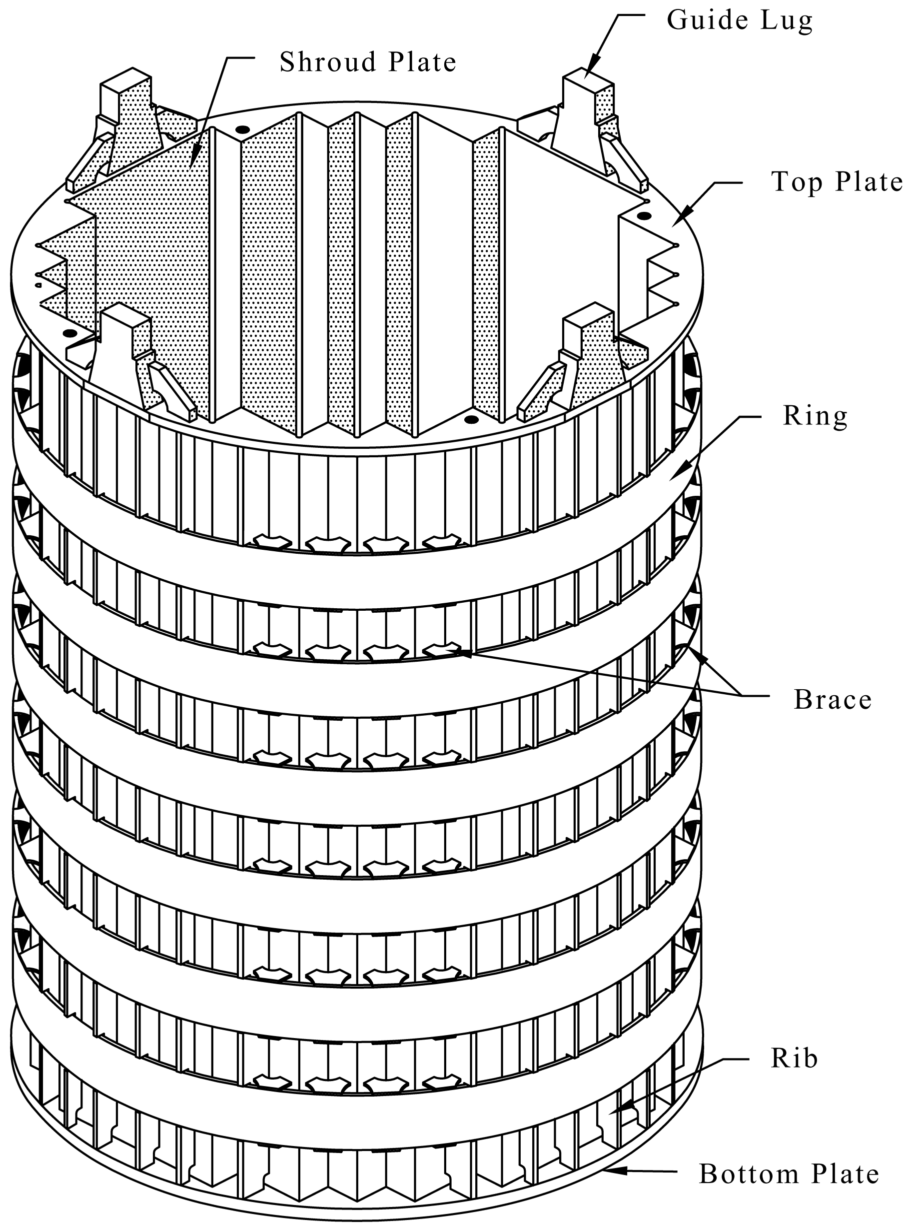


Figure 3.9-12 Core Shroud

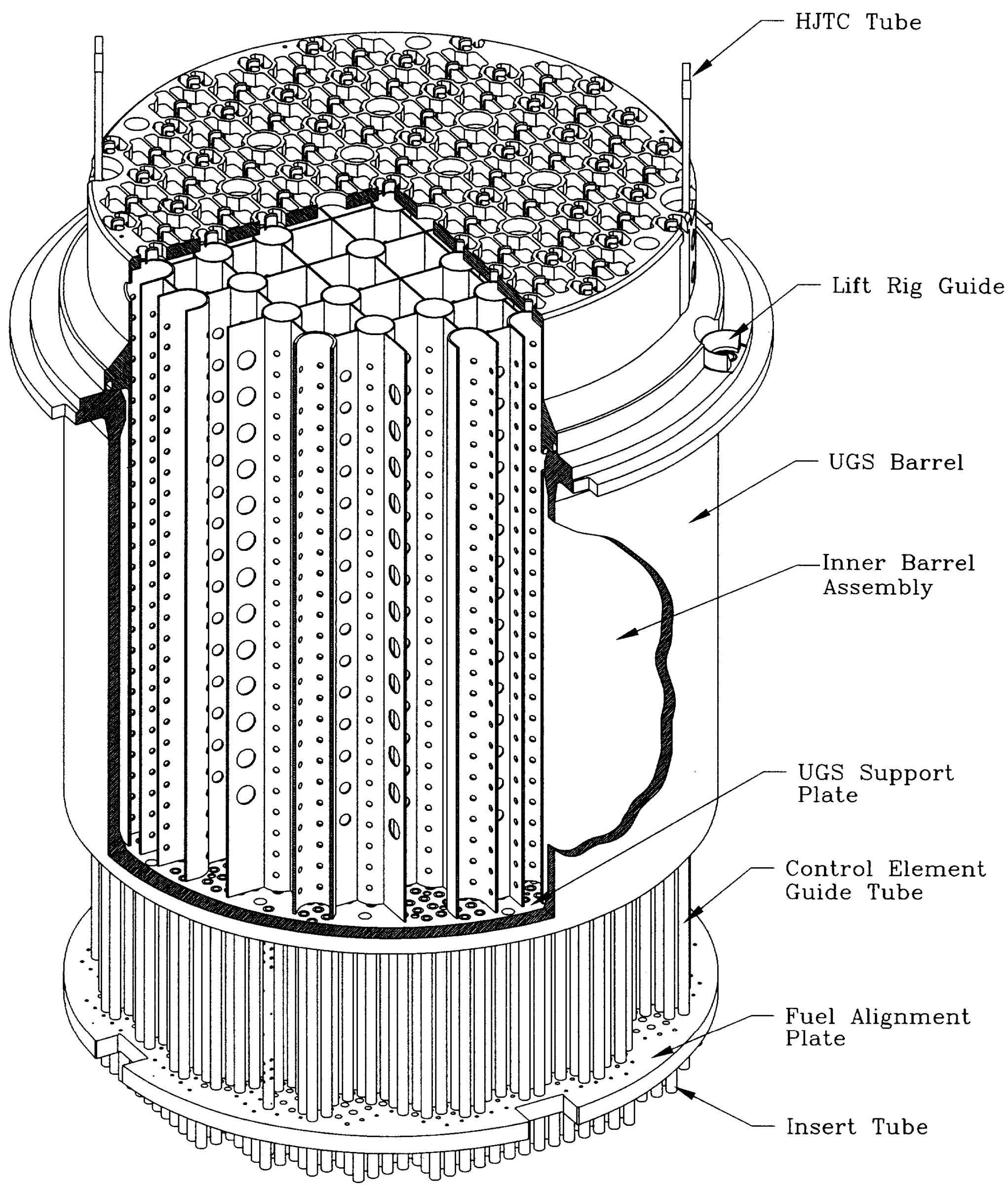


Figure 3.9-13 Upper Guide Structure Assembly

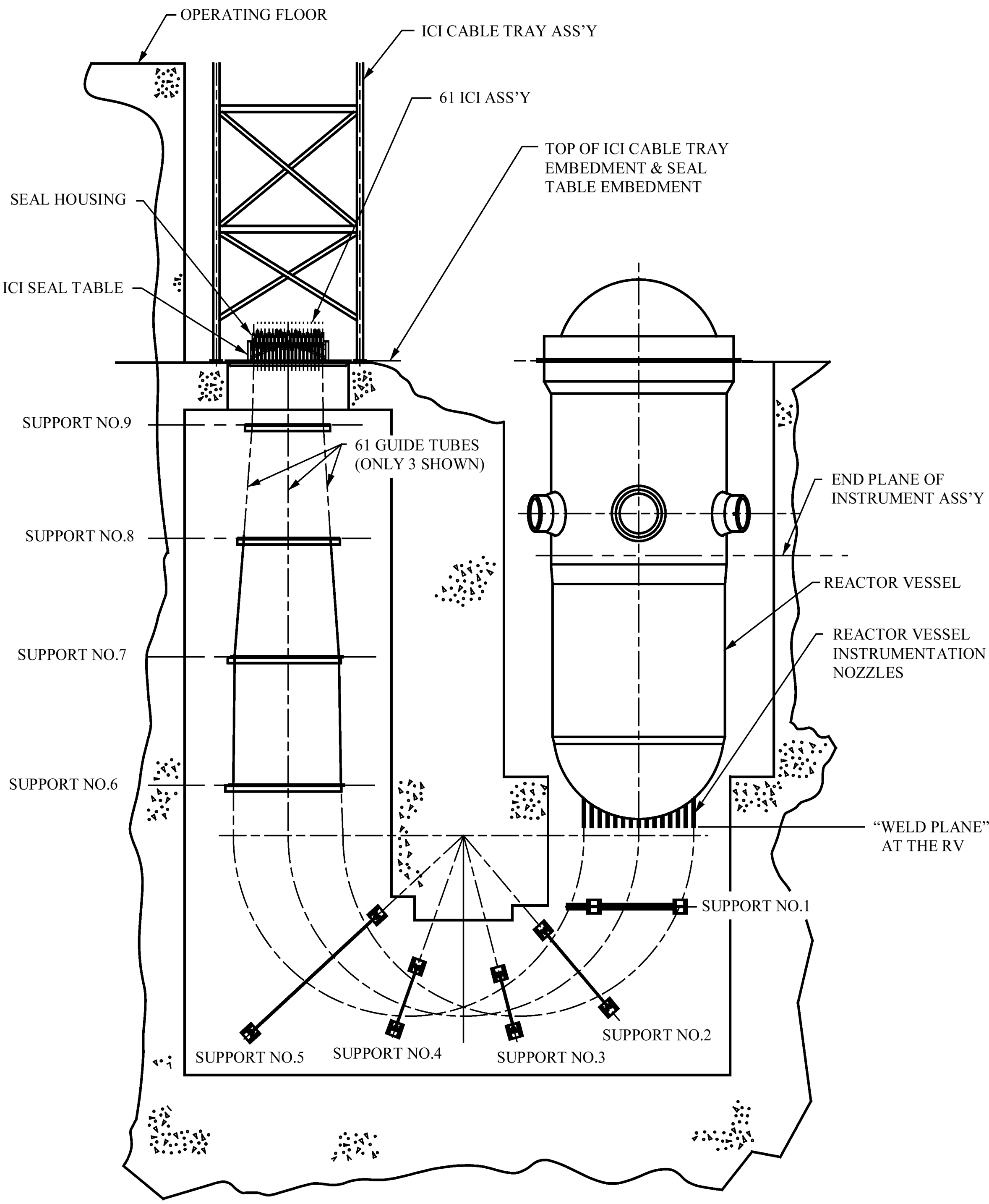


Figure 3.9-14 In-Core Instrumentation Support System

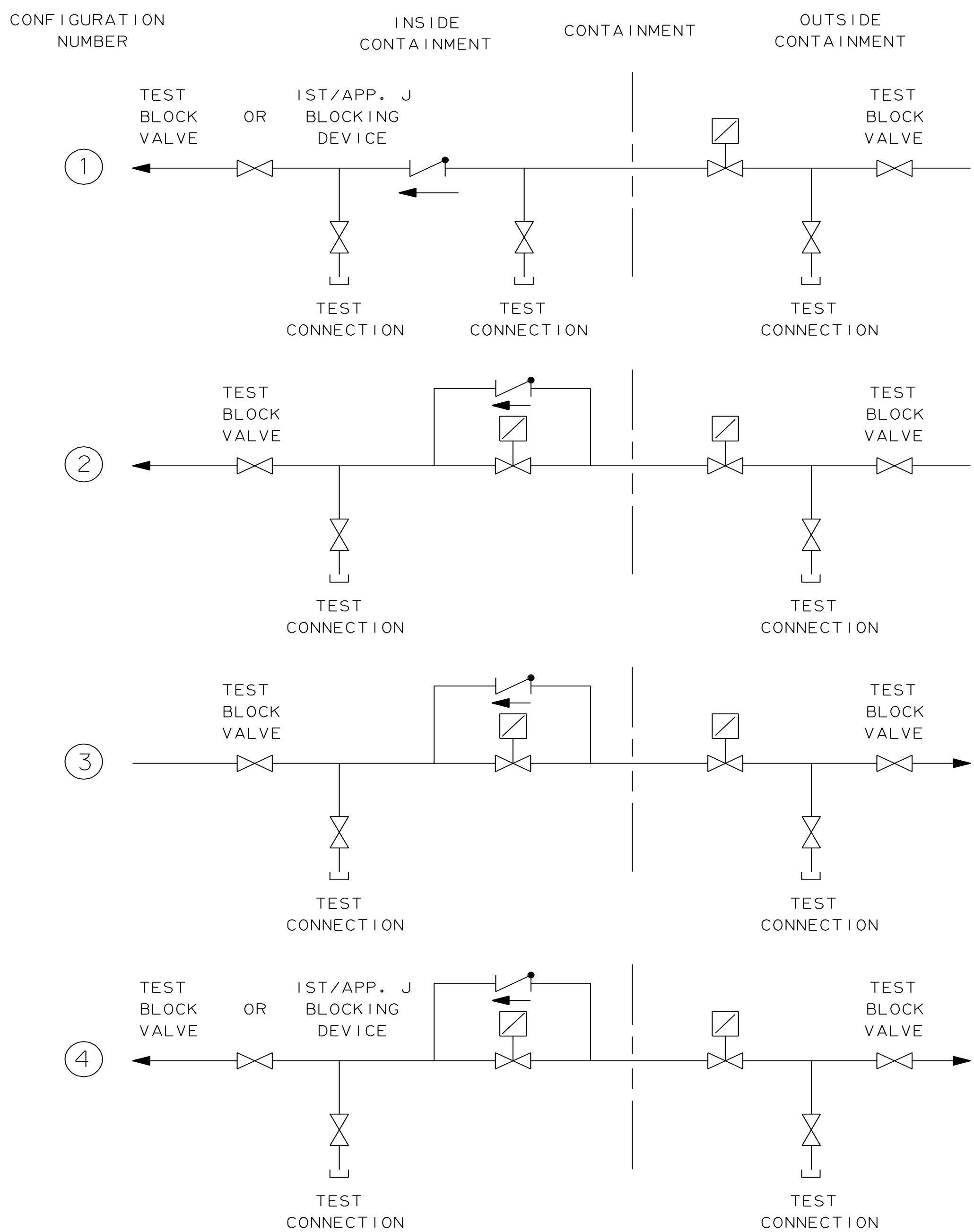


Figure 3.9-15 Typical Inservice Testing Connections (1 of 9)

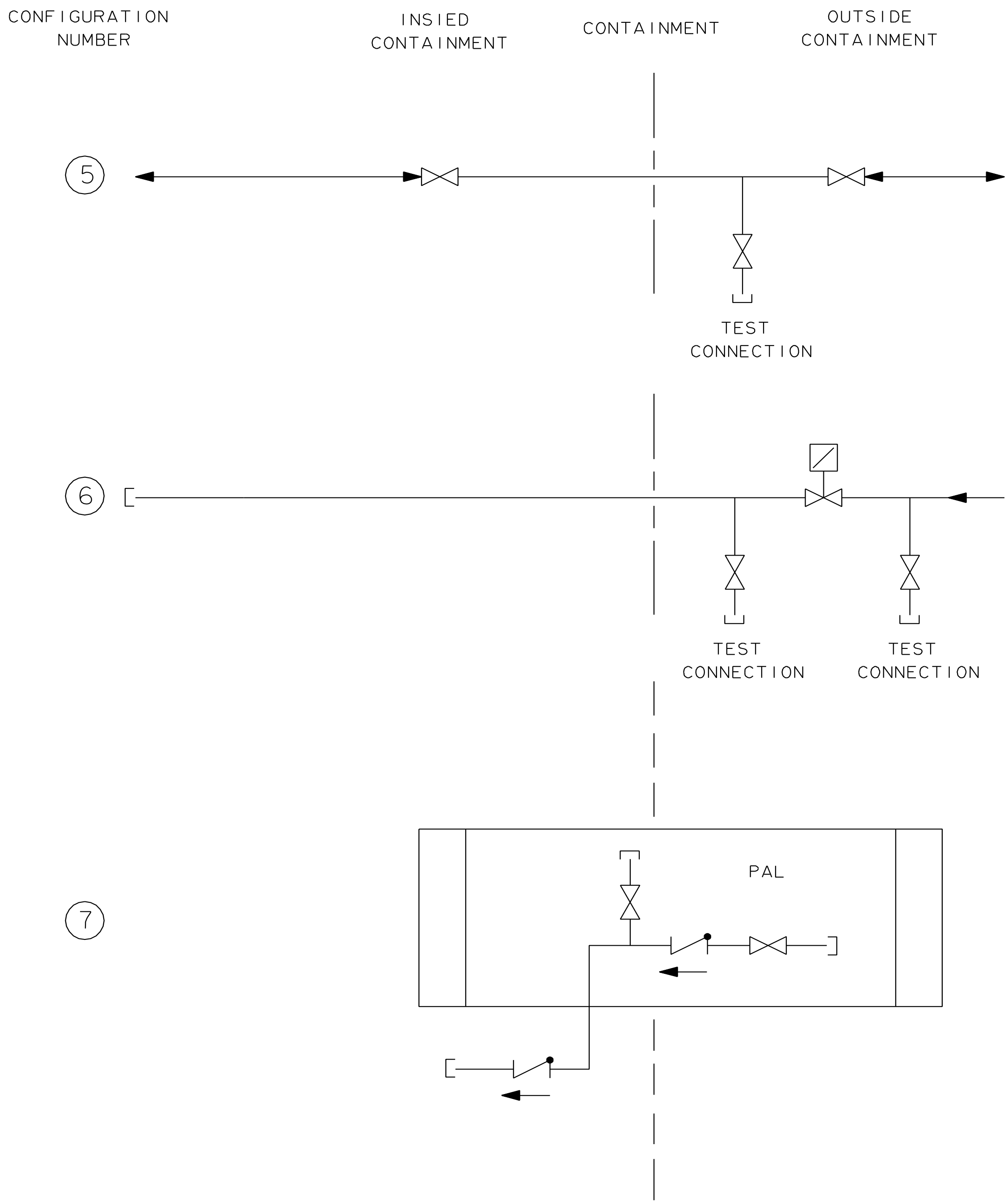


Figure 3.9-15 Typical Inservice Testing Connections (2 of 9)

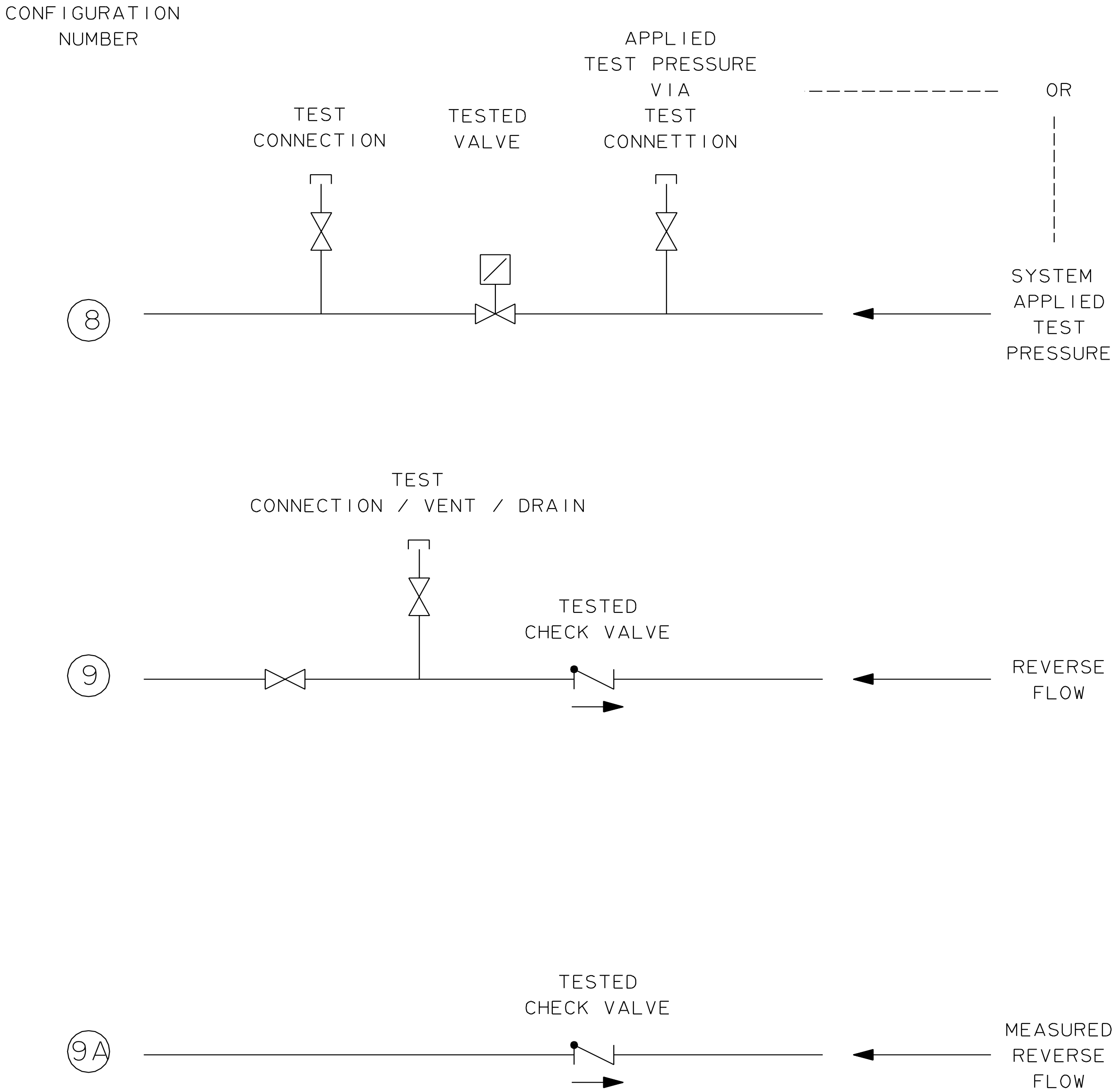


Figure 3.9-15 Typical Inservice Testing Connections (3 of 9)

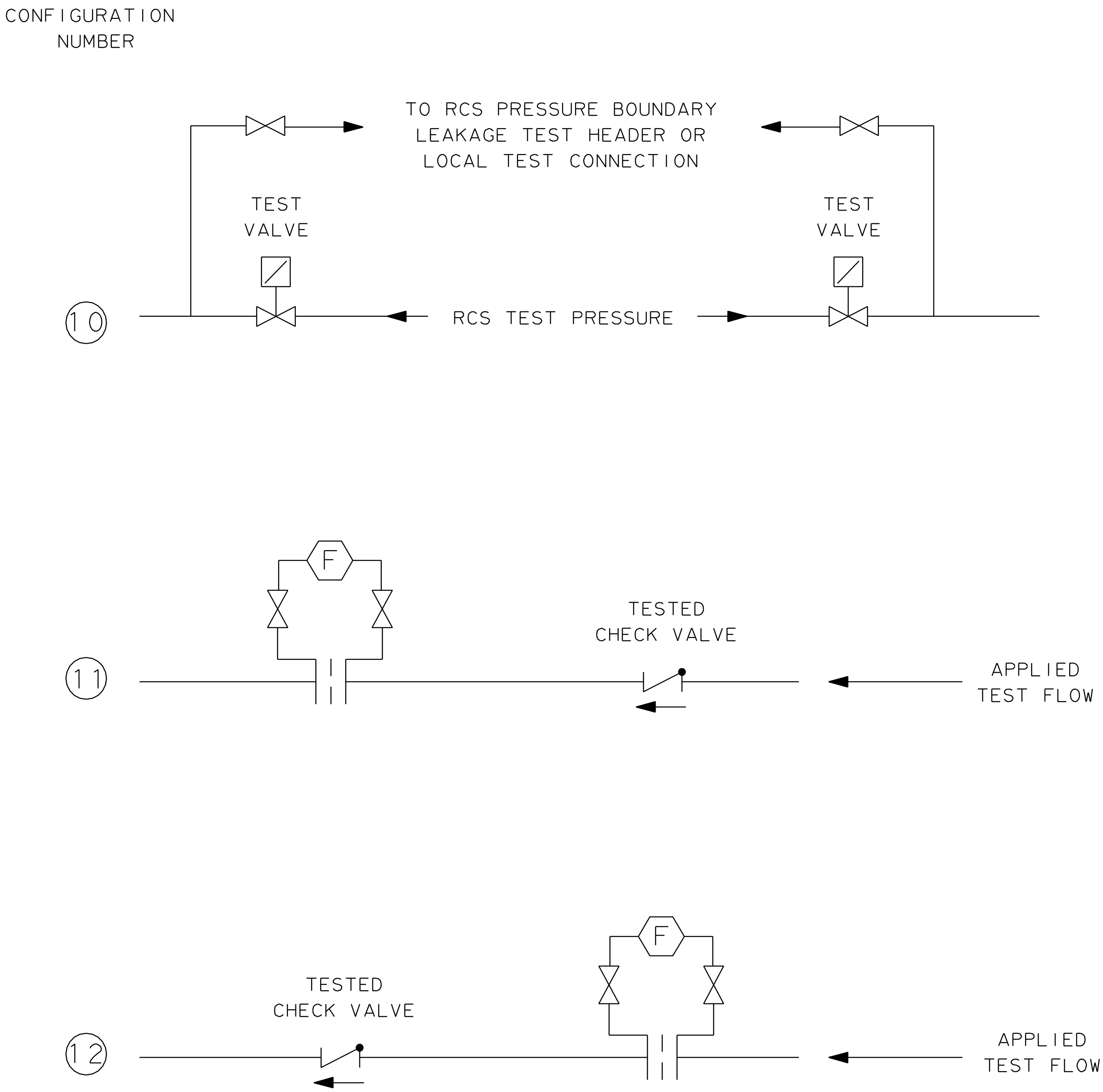


Figure 3.9-15 Typical Inservice Testing Connections (4 of 9)

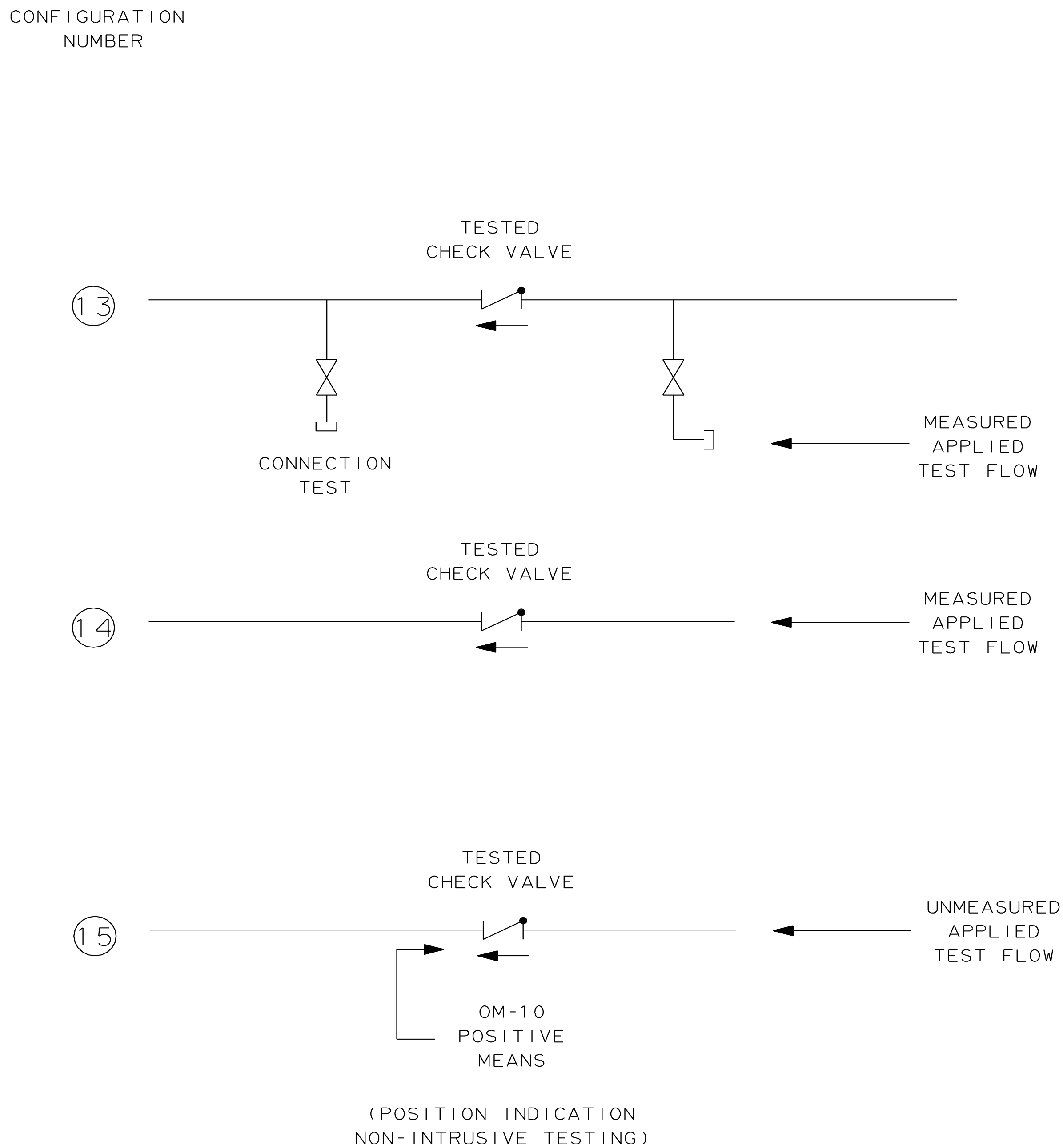
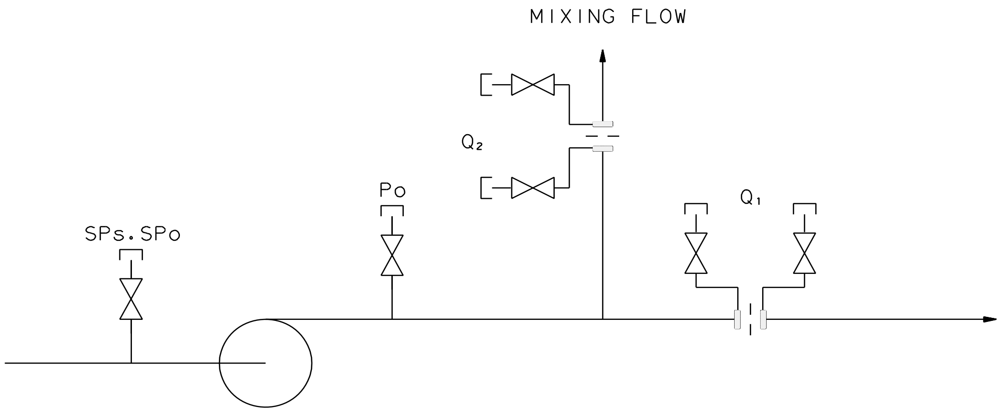


Figure 3.9-15 Typical Inservice Testing Connections (5 of 9)

CONFIGURATION
NUMBER

CCW PUMP TESTING CONFIGURATION

16



VERTICAL WET PIT PUMP TESTING CONFIGURATION

17

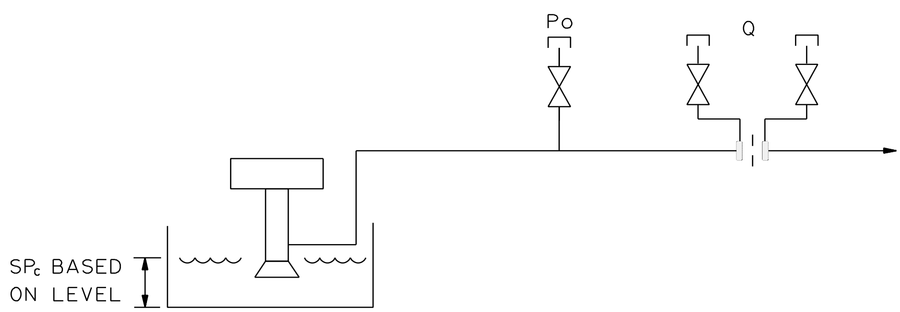


Figure 3.9-15 Typical Inservice Testing Connections (6 of 9)

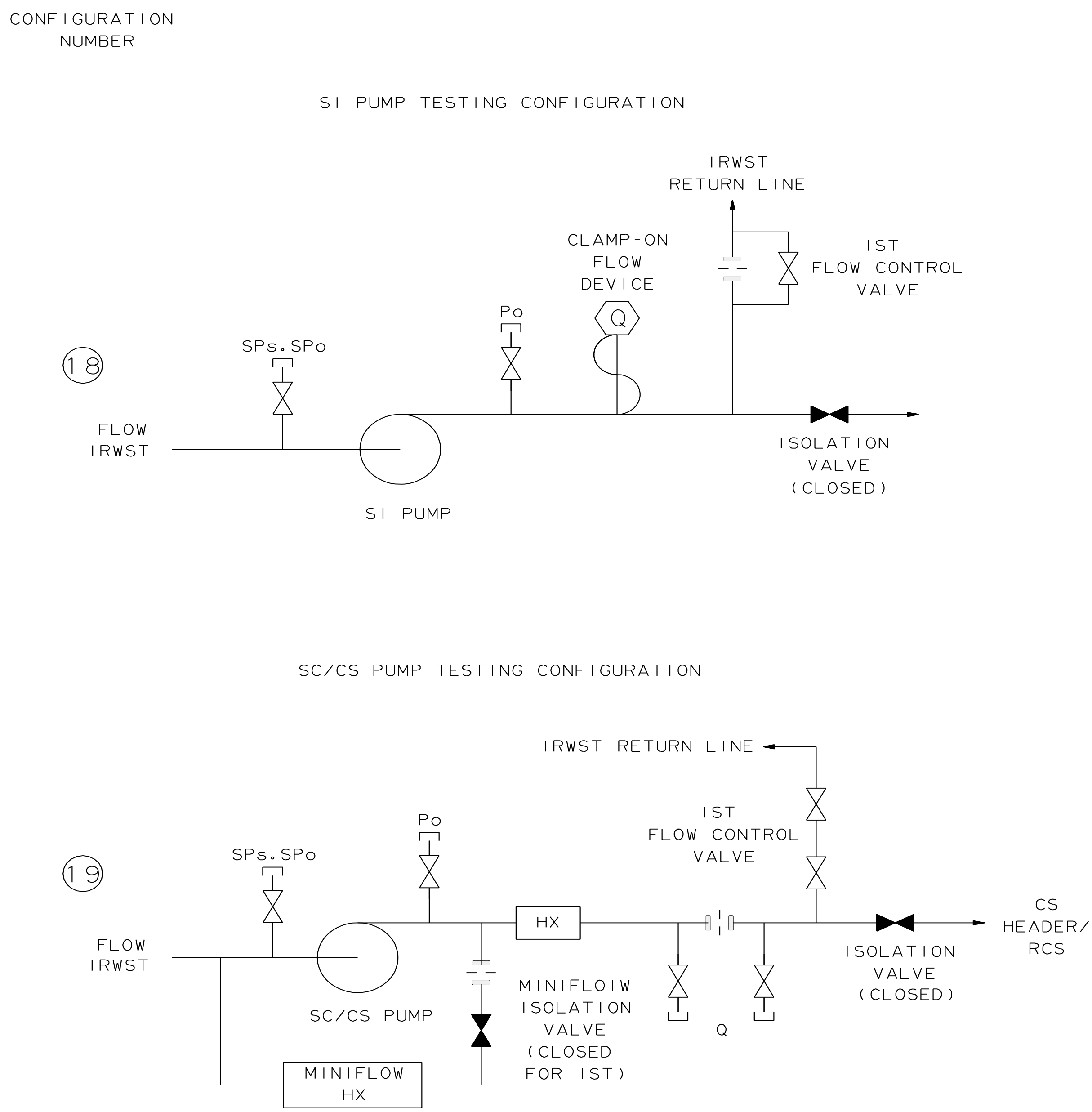
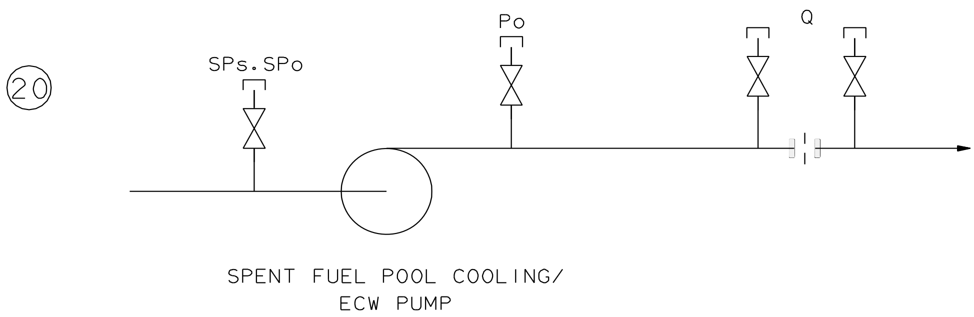


Figure 3.9-15 Typical Inservice Testing Connections (7 of 9)

CONFIGURATION
NUMBER

SPENT FUEL POOL COOLING/ECW PUMP TESTING CONFIGURATION



AFW PUMP TESTING CONFIGURATION

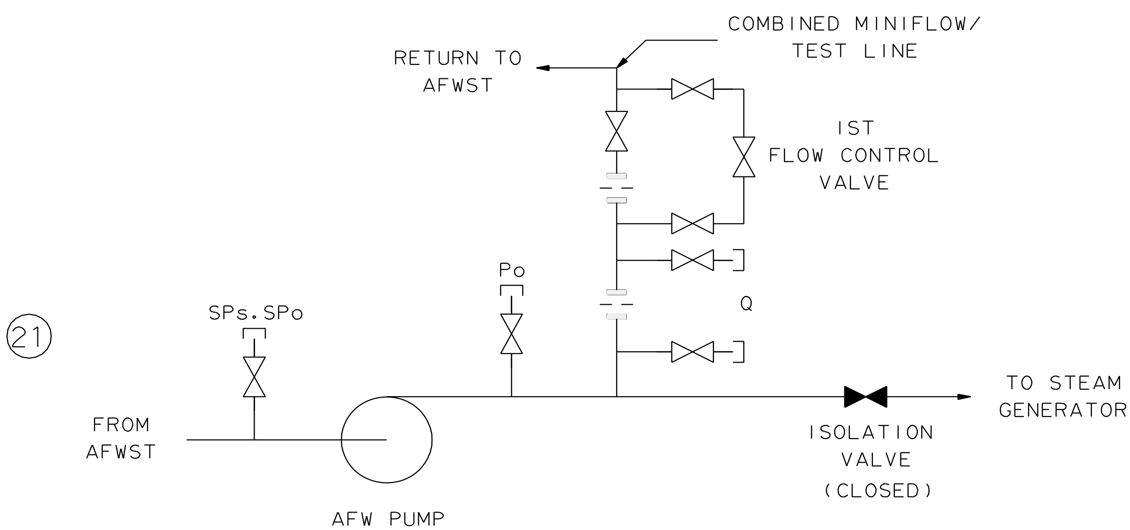


Figure 3.9-15 Typical Inservice Testing Connections (8 of 9)

CONFIGURATION
NUMBER

CCW MAKE-UP PUMP TESTING CONFIGURATION

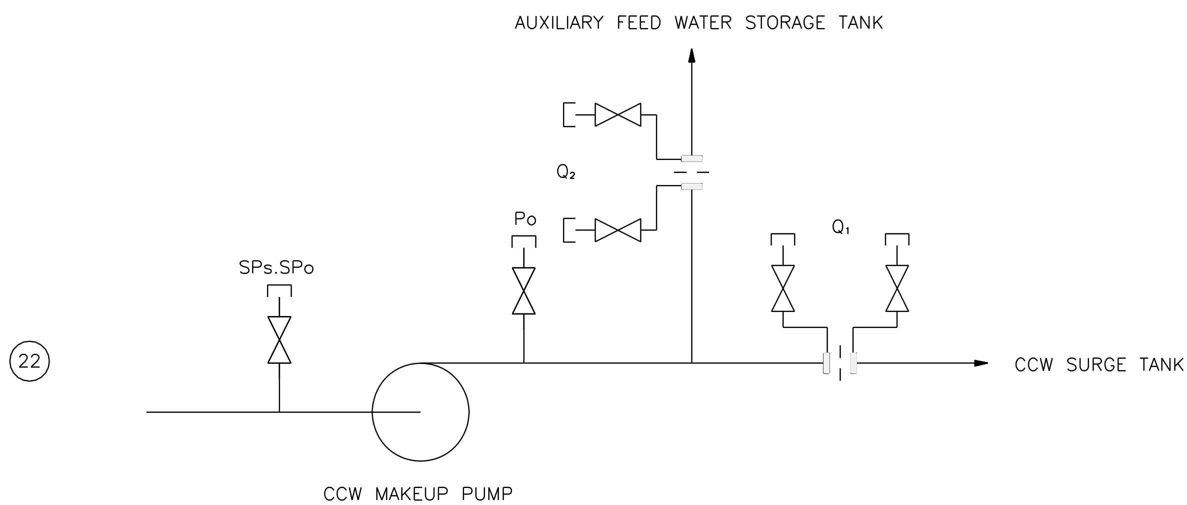


Figure 3.9-15   Typical Inservice Testing Connections (9 of 9)

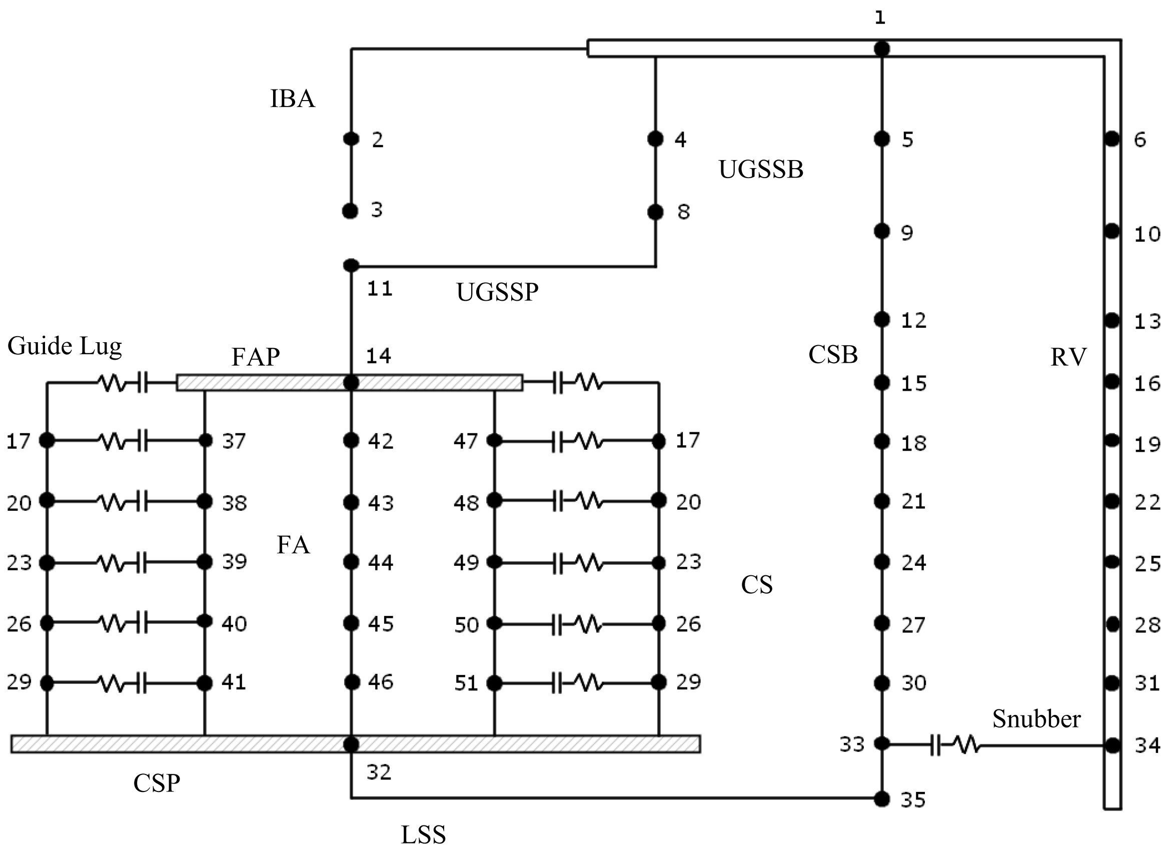


Figure 3.9-16 Reactor Internals Horizontal Seismic Analysis Model

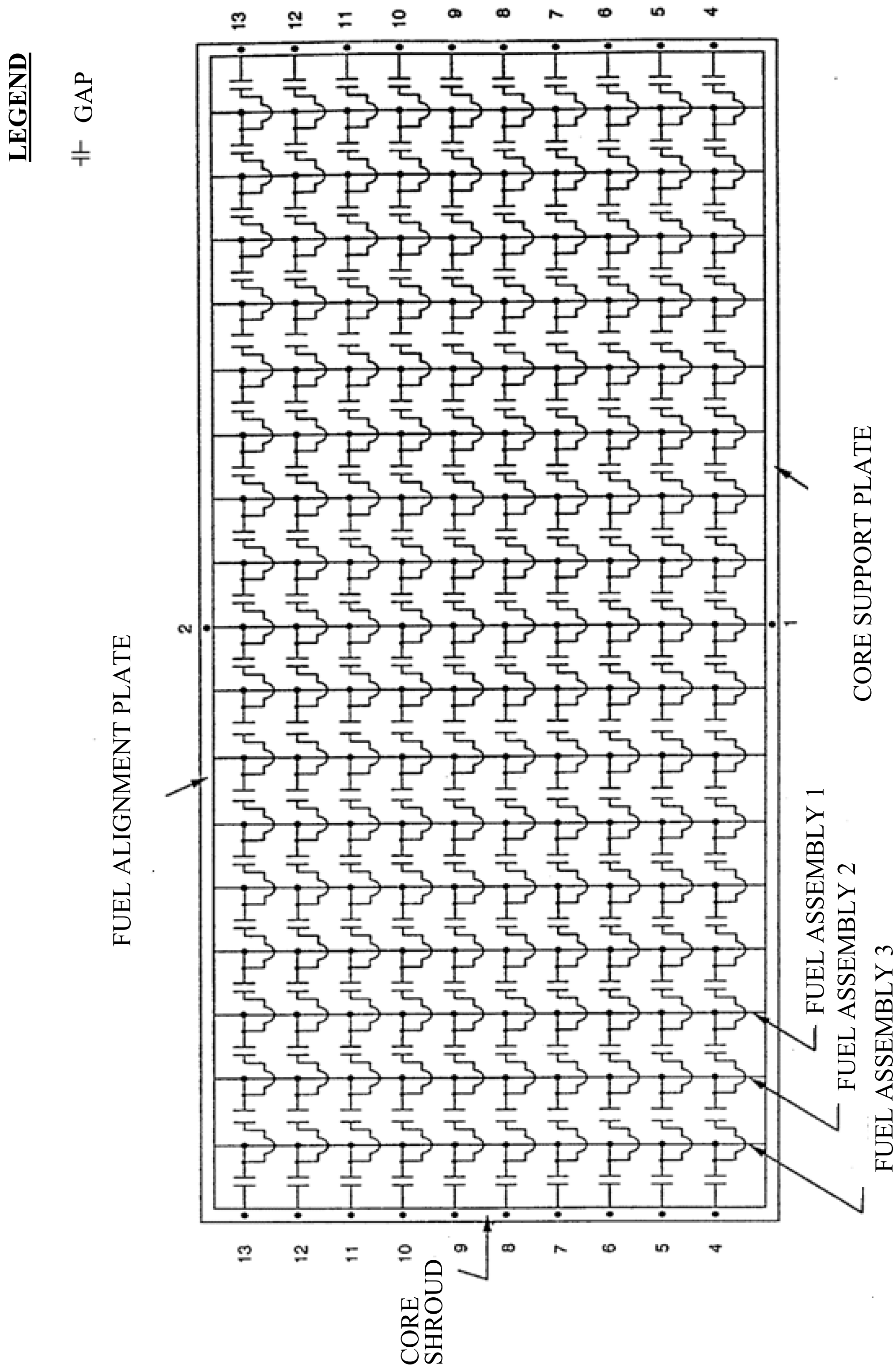


Figure 3.9-17   Core Seismic Analysis Model – One Row of 17 Fuel Assemblies

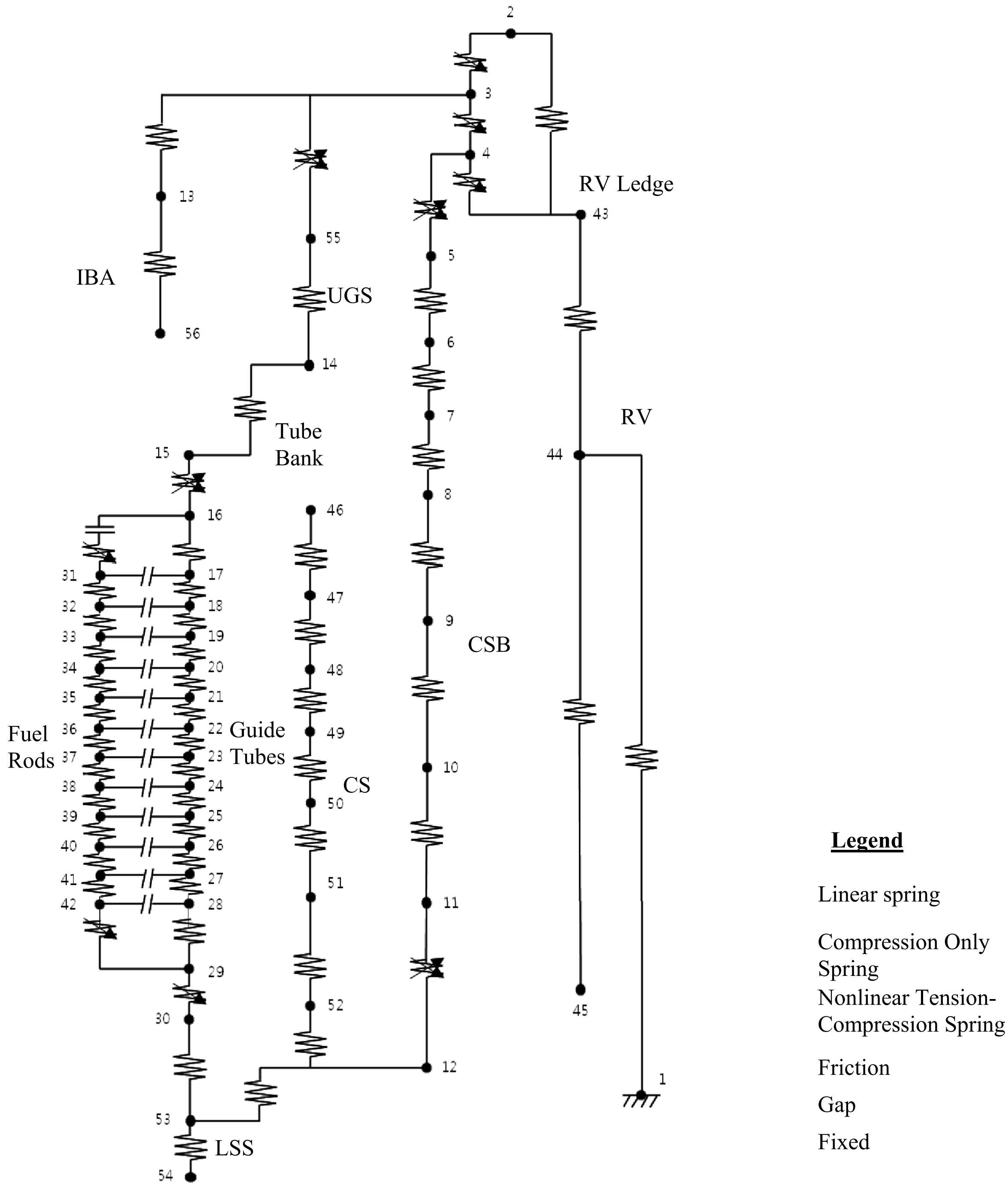


Figure 3.9-18 Reactor Internals Vertical SSE and Pipe Break Analysis Model

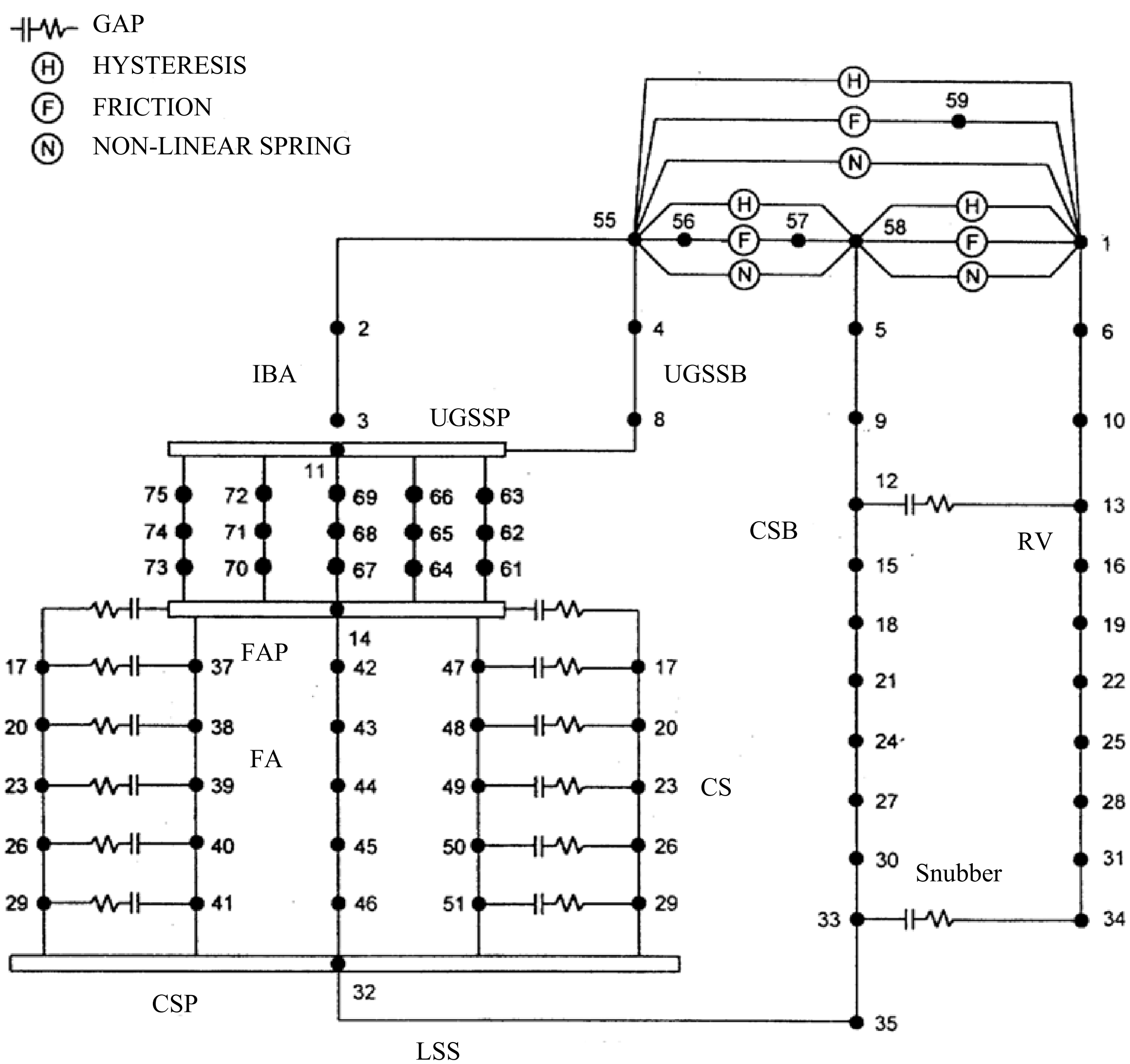


Figure 3.9-19 Reactor Internals Horizontal Pipe Break Analysis Model

APPENDIX 3.9A

SUPPLEMENTAL INFORMATION ON CRITERIA OF THE APR1400 DISTRIBUTION SYSTEMS

APPENDIX 3.9A – SUPPLEMENTAL INFORMATION ON CRITERIA OF THE APR1400 DISTRIBUTION SYSTEMS

TABLE OF CONTENTS

APPENDIX 3.9A – SUPPLEMENTAL INFORMATION ON CRITERIA OF THE APR1400 DISTRIBUTION SYSTEMS

3.9A.1 HVAC Ductwork and Supports

3.9A.1.1 General

Heating, ventilation, and air conditioning (HVAC) ductwork is designed and supported to withstand the loading combinations presented in this section, as applicable. The design and analysis guidelines herein apply to seismic Category I and II HVAC ductwork and supports. Seismic Category II HVAC ductwork and supports, as defined in Subsection 3.2.1, are analyzed to provide reasonable assurance that their failure would not adversely impact safety-related equipment or components.

3.9A.1.2 Design Considerations

3.9A.1.2.1 Internal Pressure Load (P_O, P_A)

Internal pressure loads (P_o, P_A) do not affect the design of HVAC duct supports but should be considered in the design of ductwork. Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) guidelines (References 1 and 2) are used in determining of duct thickness, stiffener, and companion angle requirements.

3.9A.1.2.2 Dead Load (D)

Dead load (D) include the weight of the ductwork itself, in-line components (e.g., dampers, humidifiers, in-duct electric heaters), externally mounted components, and insulation. Self-weight of structural members and dead load of ductwork are considered in the design of HVAC duct supports. An additional 23 kg (50 lb) concentrated load is considered in the design of HVAC duct supports for attachments such as conduits and lighting fixtures.

3.9A.1.2.3 Thermal Load (T_O, T_A)

Stresses resulting from the movement of supports or expansion of ductwork under temperature changes are avoided by using expansion joints in the system design. For ducts with gasket companion angles, thermal loads are negligible.

3.9A.1.2.4 Seismic Load (SSE)

Seismic loads on HVAC ductwork and support systems are considered using the methods described in Subsection 3.9A.1.4.3. Stresses are determined from the seismic excitation in each of the three orthogonal directions by a square root of the sum of squares (SRSS) method.

3.9A.1.2.5 Live Load (L)

HVAC duct supports are designed to withstand the expected live load. The live load considered is a construction/maintenance man-load of 114 kg (250 lb). Live load is not considered in the design of HVAC ductwork.

3.9A.1.2.6 External Pressure Differential (EPD)

HVAC ductwork and supports are designed to withstand dynamic external pressure differential (EPD) loads resulting from postulated pipe breaks. This condition is normally precluded by routing ductwork away from the affected area.

3.9A.1.2.7 Wind Load (W)

Exposed safety-related HVAC ductwork and supports are designed to withstand forces generated by wind.

3.9A.1.3 Load Combinations

The load combinations to be considered for the design of HVAC ductwork and support are as follows:

Service Level	Load Combination
A (Normal)	$D + L + P_o + T_o$
B (Severe)	$D + W + P_o + T_o$
C (Extreme)	$D + SSE + P_A + T_A$
D (Abnormal)	$D + SSE + EPD + P_A + T_A$

Where:

D	=	dead load
L	=	live load
W	=	wind load
P_o	=	internal pressure load in normal operation
P_A	=	internal pressure load in accidental operation
T_O	=	thermal load in normal operation
T_A	=	thermal load in accidental operation
SSE	=	safe shutdown earthquake
EPD	=	external pressure differential

3.9A.1.4 Analysis and Acceptance Criteria

3.9A.1.4.1 General

Structural integrity of HVAC ductwork and supports is demonstrated to provide reasonable assurance that the ductwork and support functions are not impaired. Ductwork stresses are maintained within the allowable limits. Ductwork deflection is limited to the maximum deflection criteria. HVAC duct support stresses are also maintained within the allowable limits.

3.9A.1.4.2 Damping Values

The following damping values are used in the design of HVAC ductwork and supports:

Structures	Damping Values
HVAC ductwork	7%
HVAC duct supports	4%

3.9A.1.4.3 Seismic Analysis

3.9A.1.4.3.1 Equivalent Static Analysis Method

The equivalent static analysis method is used for HVAC ductwork and supports. The system response is assumed to be the peak of the required response spectra. This response is then multiplied by a static coefficient of 1.5. The seismic load in the design of HVAC ductwork and supports is obtained by multiplying the peak acceleration by a static coefficient of 1.5 and the participating mass.

3.9A.1.4.3.2 Dynamic Analysis Method

If a specific dynamic analysis is required, HVAC ductwork and supports are modeled to accurately represent the mass distribution and stiffness characteristics. The response spectrum modal analysis or time history analysis can be applicable.

3.9A.1.5 Allowable Stress Criteria

All HVAC ductwork and supports safely sustain stresses induced by the various loading conditions. The stress values are provided as a basis for evaluating the required structural integrity of the HVAC ductwork and supports.

For HVAC ductwork, the allowable stresses for the various operating conditions are as follows:

Service Level	Allowable Stress
A (Normal)	Basic Allowable Stress
B (Severe)	Basic Allowable Stress
C (Extreme)	1.6 × Basic Allowable Stress ($\leq 0.95\ F_y$)

D (Abnormal)	1.7 × Basic Allowable Stress (≤ 0.95 F_y)

For stiffeners, companion angles, and HVAC duct supports, the allowable stresses for the various operating conditions are as follows:

Service Level	Allowable Stress
A (Normal)	Basic Allowable Stress
B (Severe)	Basic Allowable Stress
C (Extreme)	1.6 × Basic Allowable Stress

Connections are designed so that stress levels in welds and bolts do not exceed the basic allowable stresses under any of the operating conditions.

3.9A.1.5.1 Basic Allowable Stress

3.9A.1.5.1.1 HVAC Ductwork

Basic allowable stresses for HVAC ductwork are per the American Iron and Steel Institute (AISI) "Cold-Formed Steel Design Manual" (Reference 3). The basic allowable stress for the stiffener, companion angle for HVAC ductwork is determined based on *[ANSI/AISC N690,]** "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities" (Reference 4).

3.9A.1.5.1.2 HVAC Duct Supports

Basic allowable stresses for structural steel, welds, and bolts are per *[ANSI/AISC N690]** (Reference 4).

3.9A.1.6 Allowable Deflection Criteria

Maximum deflection is evaluated to provide reasonable assurance that the duct function is not impaired and the required clearances are maintained.

3.9A.1.7 Welding and Weld Acceptance Criteria

Welding activities for HVAC ductwork are accomplished in accordance with American Welding Society (AWS) D1.3 (Reference 5). For HVAC duct supports, welding activities are fulfilled in accordance with AWS D1.1 (Reference 6). The visual acceptance criteria of welding are defined in standard NCIG-01 (Reference 7).

3.9A.2 Cable Tray/Conduit Supports

3.9A.2.1 General

Cable tray and conduit are designed and supported to withstand the loading combinations presented in this section, as applicable. The design and analysis guidelines herein apply to seismic Category I and II cable tray/conduit supports as defined in Subsection 3.2.1.

3.9A.2.2 Design Considerations

3.9A.2.2.1 Dead Load (D)

Dead loads (D) include the weight of the cable tray or conduit, fittings, covers, and any other dead loads applied to the system. An additional 23 kg (50 lb) concentrated load is considered in the design of the cable tray supports for attachments such as conduits and lighting fixtures.

3.9A.2.2.2 Live Load (L)

Cable tray supports are designed to withstand expected live load. The live load considered is a construction/maintenance man load of 114 kg (250 lb). Live load is not considered in the design of conduit supports. However, where live loads such as wind and snow are applicable, they are considered in the design.

3.9A.2.2.3 Seismic Load (SSE)

Cable tray/conduit supports are designed to withstand seismic load. Stresses are determined for the seismic excitation in each of the three orthogonal directions by a square root of the sum of squares (SRSS) method. Seismic load is determined using the equivalent static analysis method or the dynamic analysis method.

3.9A.2.2.3.1 Equivalent Static Analysis Method

The equivalent static analysis method is used for cable tray/conduit supports. The system response is assumed to be the peak of the required response spectra. This response is then multiplied by a static coefficient of 1.5. The seismic load in the design of the cable tray/conduit supports is obtained by multiplying the peak acceleration by a static coefficient of 1.5 and the participating mass.

3.9A.2.2.3.2 Dynamic Analysis Method

If specific dynamic analysis is required, the cable tray/conduit supports are modeled to accurately represent their mass distribution and stiffness characteristics. The response spectrum modal analysis or time history analysis can be used.

3.9A.2.3 Load Combinations and Allowable Stress Criteria

The loading combinations considered for the design of cable tray/conduit supports are as follows:

Service Level	Load Combination
A (Normal)	D + L
C (Extreme)	D + L + SSE

Where:

D = dead load

L = live load (only for cable tray supports)

SSE = safe shutdown earthquake

For cable tray/conduit supports, the allowable stresses for the various operating conditions are as follows:

Service Level	Allowable Stress
A (Normal)	Basic Allowable Stress
C (Extreme)	1.6 × Basic Allowable Stress

3.9A.2.3.1 Basic Allowable Stress

Basic allowable stresses for the structural steel, welds, and bolts of cable tray/conduit supports are determined based on *[ANSI/AISC N690]** (Reference 4).

3.9A.2.4 Damping Value

The following damping values are used in the design of the cable tray/conduit supports:

Structures	Damping Values
Cable tray support	7%
Conduit support	5%

3.9A.2.5 Welding and Weld Acceptance Criteria

Welding activities for cable tray/conduit supports are accomplished in accordance with AWS D1.1 (Reference 6). The visual acceptance criteria of welding are defined in standard NCIG-01 (Reference 7).

3.9A.3 References

1. Sheet Metal and Air Conditioning Contractors' National Association, "Rectangular Industrial Duct Construction Standards," SMACNA 1922, 2004.

2. Sheet Metal and Air Conditioning Contractors' National Association, "Round Industrial Duct Construction Standard," SMACNA 1520, 1999.

3. AISI Manual, "Cold-Formed Steel Design," 2008.

4. *[ANSI/AISC N690 with Supplement No. 2, 2004]**, "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities."

5. AWS D1.3, "American Welding Society, Structural Welding Code – Sheet Steel," 2008.

6. AWS D1.1, "American Welding Society, Structural Welding Code – Steel," 2010.

7. NCIG-01, "Visual Weld Acceptance Criteria for Structural Welding of Nuclear Power Plants," Rev. 2, EPRI NP-5380, 1987.