3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

This subsection addresses information concerning methods of analysis for Seismic Category I components and supports, including those designated as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III (Reference 3.9-1), Division 1 Class 1, 2, 3, subsection NG for core support structures, subsection NF for supports, and those not covered by the ASME BPVC as discussed in NUREG 0800 Standard Review Plan (SRP) 3.9.1. Information also is presented concerning design transients for ASME BPVC Class 1 and core support structure components and supports.

The NuScale Power Plant design meets the relevant requirements of the following General Design Criteria (GDC) of 10 CFR 50, Appendix A:

- GDC 1, as it relates to components being designed, fabricated, erected, constructed, tested, and inspected in accordance with the requirements of applicable codes and standards commensurate with the importance of the safety-related functions to be performed. Compliance with GDC 1 is discussed in Section 3.1.
- GDC 2, as it relates to mechanical components of systems being designed to withstand seismic events without loss of capability to perform their safety-related functions. Pursuant to GDC 2, mechanical components are designed to withstand the loads generated by natural phenomena as discussed Section 3.1.1.
- GDC 14, as it relates to the reactor coolant pressure boundary (RCPB) being designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. As discussed below, the design transients and consequent loads and load combination with appropriate ASME code service limits, provide reasonable assurance that the RCPB is designed to maintain the stresses within acceptable limits to accommodate the system pressures and temperatures expected from normal operation including anticipated operational occurrences (AOOs), infrequent events, and accident loading conditions such as safe shutdown earthquake (SSE).
- GDC 15, as it relates to the mechanical components of the reactor coolant system (RCS) being designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. The overpressure protection features are designed with sufficient capacity to prevent the RCPB from exceeding 110 percent of design pressure during normal operations and AOOs. Safety-related mechanical components are designed to remain functional under postulated combinations of normal operating conditions, AOOs, postulated pipe breaks, and seismic events in compliance with the requirements of GDC 14 and 15.
- 10 CFR 50, Appendix B, Section III, as it relates to quality of design control. Section 17.5 satisfies the requirements of 10 CFR 50, Appendix B, to ensure that SSCs are designed, procured, fabricated, inspected, erected, and tested to standards commensurate with their contribution to plant safety.
- 10 CFR 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics.

This requirement is met by including design-transient seismic events as part of the design basis for withstanding the effects of natural phenomena.

3.9.1.1 Design Transients

The design transients define thermal-hydraulic conditions (i.e., pressure, temperature, and flow) for the NPM. Bounding thermal-hydraulic design transients are defined for components of the RCPB. The number of cycles for each design transient is based on a plant life of 60 years. The transients are defined for safety-related equipment design purposes and are intended to provide a bounding representation of the NPM operation.

The following operating condition categories, as defined in the ASME BPVC, Section III (Reference 3.9-1), apply to Class 1 and 2 components, the containment vessel (CNV), supports, reactor vessel internals (RVI), piping, and valves inside and outside of containment up to the outermost containment isolation valve:

- **ASME Service Level A**

  Service Level A includes conditions associated with events that are planned to occur due to routine operation of the plant. Examples include startup, power maneuvers, and shutdown.

- **ASME Service Level B**

  Service Level B includes conditions associated with transients that occur often enough that the operability of the plant is not affected. These transients will not result in damage requiring repairs.

- **ASME Service Level C**

  Service Level C events may result in permanent deformation and repairs may be required to correct large deformations in areas of structural discontinuity.

- **ASME Service Level D**

  Service Level D events may result in gross deformation and dimensional instability. Repair or replacement of components may be necessary to correct mechanical damage.

- **Test Conditions**

  These conditions include pressure tests required by ASME BPVC, Section III (Reference 3.9-1), and other tests required by the design specifications.

Table 3.9-1, Summary of Design Transients, lists the design transients by ASME service level and includes the number of events over the design life of the plant for each transient. Load combinations and their acceptance criteria are given in Section 3.9.3 for mechanical components and associated supports and in Section 3.12 for piping systems.

The Service Level A and B transients are representative of events that are expected to occur during plant operation. These transients are severe or frequent enough to be evaluated for component cyclic behavior and equipment fatigue life, and the analyzed conditions are based on a conservative estimate of the frequency and magnitude of temperature and pressure changes. When used as a basis for component fatigue evaluation, the bounding transients provide confidence that the component is appropriate for its application over the design life of the plant. Service Level C and D conditions are not typically included in fatigue evaluations in accordance with the ASME BPVC, Section III (Reference 3.9-1). For select component and transient combinations, Service Level C events are evaluated against Level B stress limits. This selection is made either because the event contains significant stress cycles or the transient is considered a normal design operation for that component. The following sections describe the assumptions used in thermal-hydraulic analysis for each Service Level.

3.9.1.1.1 Service Level A Conditions

Service Level A Transient 1 - Reactor Heatup to Hot Shutdown

This transient covers the heatup and pressurization from transition mode to hot shutdown. The event begins with a depressurized reactor vessel filled with water. The CNV also is filled initially with water up to the elevation of the pressurizer baffle plate.

The CNV is pressurized to at-or-above the minimum pressure required to begin the containment drain process. The RCS is pressurized equivalently by adding nitrogen gas to the pressurizer. Once pressurizer heaters are actuated to increase RCS pressure, the nitrogen is removed through the reactor pressure vessel (RPV) high point degasification line and is replaced with steam. After the RCS has reached the hot shutdown temperature and normal operating pressure of 1850 psia, a system leakage test is performed per the requirements of ASME BPVC Section XI (Reference 3.9-2).

Service Level A Transient 2 - Reactor Cooldown from Hot Shutdown

This transient encompasses the cooling from hot shutdown to transition mode and is generally the reverse of the reactor heatup to hot shutdown. The temperature of the RCS is continually reduced by controlling the feedwater flow rate. The steam and feedwater flow rates are controlled to keep the cooling rate below the maximum of 100 degrees F/hr (200 degrees F/hr in the pressurizer). The RCS temperature changes also are limited to maintain subcooling between the pressurizer and RCS hot leg less than 250 degrees Fahrenheit. The chemical and volume control system (CVCS) is used to increase the boron concentration to shutdown levels and to add makeup to compensate for coolant shrinkage. The containment flooding and drain system is used to add pool water to containment to continue cooling the CNV and RPV.

Service Level A Transient 3 - Power Ascent from Hot Shutdown

This transient covers the power ascent from hot zero power conditions in hot shutdown mode to 15 percent of full power at which point the control systems are placed in automatic mode. Automatic mode is expected to cover power levels above 15 percent of full power. Throughout this transient, the steam and feedwater flow rates are controlled to match the demanded load ramp, which is specified to be limited to 0.5 percent of full power per minute. The feedwater temperature remains at the condenser hot well temperature as the feedwater heaters are unavailable.

Service Level A Transient 4 - Power Descent to Hot Shutdown

This transient occurs as the reverse of the power ascent from hot shutdown. The transient covers the reactor conditions that span from 15 percent of full power to hot zero power conditions in hot shutdown mode. The lower limit of the power range where the reactor is under automatic control occurs at 15 percent of full power. Since the turbine is offline, steam produced by cooling the RCS is diverted through the turbine bypass valve. A uniform ramp decrease in power is specified to occur a maximum rate of 0.5 percent of full power per minute. Feedwater heating is not available as the turbine is offline and, therefore, the feedwater temperature is equal to the condenser hot well temperature.

Service Level A Transient 5 - Load Following

The reactor could be required to provide load following capabilities to match the electrical demand of the grid over a 24-hour period. The load begins at full power and ramps down to 20 percent of full power over two hours. The load then remains constant for up to ten hours before ramping back to full power over two hours. The load remains constant at full power for the remainder of a 24-hour cycle.

Service Level A Transient 6 - Load Regulation

Load regulation refers to fluctuations in load due to the plant participating in some form of grid frequency control. The frequency control transient is defined as a 10 percent of full power increase or decrease in load at 2 percent of full power per minute. This load regulation is a plant-wide capacity, thus the change in plant load is the total power change for all operating modules. Load regulation is provided while at a steady power level or while performing the ramp power changes required for load following. Reactor power will lag behind the step change in load demand.

Service Level A Transient 7 - Steady State Fluctuations

While operating at a steady load, there may be small fluctuations in RCS temperature and pressure. These fluctuations could be due to minor control system malfunctions, instrument drifts, small power variations, or other unplanned variations. The full-power, normal operating bands for RCS average temperature and pressurizer pressure are expected to be ±0.5 degrees Fahrenheit and ±5 psia.

Service Level A Transient 8 - Load Ramp Increase

When the reactor is in automatic mode, the reactor will be capable of providing a load increase at a rate of 5 percent of full power per minute over the power range of automatic control, 15 to 100 percent of full power. A rate of 5 percent of full power per minute is an upper bound on the load increase rate for power maneuvers and is consistent with other pressurized water reactor designs. Throughout this transient, the pressure, average RCS temperature, and pressurizer level are under automatic control.

Service Level A Transient 9 - Load Ramp Decrease

When the reactor is in automatic mode, the reactor will be capable of providing a load decrease at a rate of 5 percent of full power per minute over the power range of automatic control, 15 to 100 percent of full power. A rate of 5 percent of full power per minute is an upper bound on the load decrease rate for power maneuvers and is consistent with other pressurized water reactor designs. Automatic control mode is initiated at 15 percent of full power. Feedwater temperature will decrease due to less feedwater heating as the power level decreases.

Service Level A Transient 10 - Step Load Increase

When the reactor is in automatic mode, the nuclear steam supply system components are designed to withstand the cycles associated with a 10 percent of full power step load increase. This transient could occur due to a disruption in the electrical grid. As load is increased, the imbalance between load and core power causes the RCS temperature and pressure to decrease. The pressurizer heaters will respond to a pressure decrease by increasing proportional heater output and energizing the backup heaters.

Service Level A Transient 11 - Step Load Decrease

Nuclear steam supply system components must also be capable of withstanding the cycles associated with a 10 percent of full power step load decrease. This transient could occur due to a disruption in the electrical grid. As load is decreased, the imbalance between load and core power causes the temperature and pressure to increase. The pressurizer will respond to any large pressure increase by reducing heater output and initiating normal spray flow.

Service Level A Transient 12 - Large Step Load Decrease

This transient occurs when there is a large decrease in the demanded load by the grid from full power down to 20 percent of full power. When the load decreases, steam pressure increases and steam flow rate decreases. RCS temperature and pressure increase due to the decrease in secondary heat removal. Some steam will likely need to bypass the turbine to prevent a reactor trip on high pressurizer pressure or level. The bypass load is ramped down at about 5 percent power per minute to give the reactor time to reduce power. There are two control signals to detect a large step load decrease and to regulate the bypass steam flow. An error signal between the demanded and actual turbine load will determine the need for steam bypass, and a signal will provide the expected bypass valve position as a function of the demanded load.

Service Level A Transient 13 - Refueling

During refueling, the containment vessel flange and reactor vessel flange are opened and the upper portion of the reactor module is lifted away from the lower portion, exposing the reactor core for refueling. This operation takes place in the refueling pool. There is a negligible thermal cycle on the RPV as the flanges are unbolted and cold pool water mixes in. For module handling operations to begin, the RCS must be in transition mode, below 200 degrees Fahrenheit. The maximum temperature change would then be 200 degrees Fahrenheit minus the minimum pool water temperature. There will be negligible thermal cycles for cold unbolting and re-bolting of the RPV flanges for reactor startup, because the RPV temperature will be near equilibrium with the surrounding reactor pool water following the duration of a refueling outage. There will also be stress cycles introduced by the module handling operations, such as module lifting, bolting, and unbolting.

Service Level A Transient 14 - Reactor Coolant System Makeup

The RCS makeup transient consists of the normal replenishment of RCS fluid due to minor leakage or for boron concentration adjustment by the CVCS makeup pumps. The CVCS continuously circulates coolant through the demineralizers and filters and back to the RCS. Makeup flow is required to maintain the pressurizer level, change the boron concentration, or adjust the RCS chemistry.

This transient begins when the CVCS makeup pumps are energized to add makeup coolant. The makeup coolant can be demineralized or borated water. CVCS flow is pumped to the RCS through the RCS injection line and to the pressurizer through the spray bypass line. The makeup water begins at low temperature before it is heated by the CVCS regenerative heat exchanger. The coolant returning to the RCS is colder during makeup compared to nonmakeup flows, and piping and components subjected to makeup flows experience thermal cycles.

Service Level A Transient 15 - Steam Generator Inventory Control from Hot Shutdown

This transient occurs while leaving a hold at hot shutdown. Normally, continuous feedwater and steam flows are used for steam generator (SG) inventory control when transitioning to and from hot shutdown. If there is an extended hold at hot shutdown, the decay heat generation may drop below the minimum capability of the secondary heat removal systems, thus securing the continuous feedwater and steam flows. Breaking the hold will initiate the feedwater flow again, which will provide cold feedwater to components that had already reached equilibrium with the hot RCS. The main steam isolation valves (MSIVs) are opened to the desired position and the feedwater and steam flows are operated continuously to achieve the objective. An option is to use the CVCS module heatup system to maintain the primary coolant temperature, which reduces the need to turn off feedwater flow and reduces the thermal cycles on RPV components. During a reactor cooldown due to decay heat removal system (DHRS) actuation, the DHRS cooldown may be interrupted by re-establishing feedwater flow to the SG. The feedwater flow is a continuous flow, providing inventory to the SG. This will be less severe for components, such as the feedwater plenum, since the plenum would already be at a cooler temperature due to the flow of DHRS condensate during the reactor cooldown.

Service Level A Transient 16 - High Point Degasification

There are two transients for the high-point degasification line, consisting of the normal operation venting of the pressurizer and shutdown degasification of the RCS. The normal operation venting transient involves periodically opening the valves in the high-point degasification line to remove non-condensable gases that have collected in the vapor space of the pressurizer. Prior to and during shutdown operations, the high-point degasification line is used to mechanically degas the RCS to remove gases from the pressurizer vapor space and dilute the concentration of hydrogen in the reactor coolant by venting and providing makeup from the CVCS.

Service Level A Transient 17 - Containment Evacuation

The containment evacuation system connects to the containment vessel nozzle with no internal piping and is used to add and remove gases. The containment evacuation transient is made up of three events: startup operation with air or nitrogen addition and removal, shutdown operation with air removal, and normal operation removal of water vapor or non-condensable gases. During startups and shutdowns, service air or nitrogen is added and removed through the containment evacuation system to control containment liquid levels. During normal operation, the line is used for continuous or sporadic removal of water vapor or gases to maintain a vacuum in the containment vessel. This ensures that water vapor leaked into the containment vessel does not condense and collect at the bottom. If there is leakage, then the containment evacuation system is expected to run, continuously or intermittently, until the leak is fixed during the next reactor shutdown.

Service Level A Transient 18 - Containment Flooding and Drain

The containment flooding and drain system connects to a CNV nozzle with piping extending from the top head of the CNV to the bottom for each module. The piping is used to add and remove water to and from the CNV. This transient is split into two events: containment flooding operations after shutdown and containment drain operations prior to startup. After shutdown, the containment flooding and drain containment isolation valves are opened and the pump transfers water from the reactor pool to the CNV. Prior to startup of the module, the containment flooding and drain system containment isolation valves are opened and containment is pressurized through the containment evacuation system penetration to the minimum pressure required or to provide adequate net positive suction head to the pump, which helps drain the CNV of water.

3.9.1.1.2 Service Level B Conditions

Service Level B Transient 1 - Decrease in Feedwater Temperature

A decrease in feedwater temperature could occur due to many different malfunctions in the secondary side system. However, the bounding malfunction is the loss of feedwater heating. Such a failure at full power drops the feedwater temperature significantly, which quickly reduces the RCS temperature and adds reactivity due to the negative moderator temperature coefficient. The colder coolant and reactivity insertion cause core power to increase and can result in a reactor trip on high power. The secondary control system compensates for the lower feedwater temperature by adjusting the feedwater flow rate to reach the load demand setpoint. If a trip setpoint is not reached, reactivity feedback will allow the reactor power to re-adjust to match the demanded load.

Service Level B Transient 2 - Increase in Secondary Flow

An equipment or control system malfunction could cause an increase in secondary flow. A malfunction could be on the steam side, such as opening the turbine throttle valve, or on the feedwater side, such as opening the feedwater regulating valve or increasing the feedwater pump speed. Any of these malfunctions leads to an increase in feedwater flow rate, but the feedwater pressure could increase or decrease. One of the control valves opening leads to a feedwater pressure decrease while an increase in feedwater pump speed increases the feedwater pressure.

The bounding cases are the following: the complete opening of either the feedwater regulating valve, turbine throttle valve, or turbine bypass valve or the feedwater pump speed increasing to 100 percent. The RCS responds to an increase in secondary flow rate with a decrease in temperature and pressure. Reactivity feedback then causes an increase in reactor power.

There will also be a control system response for the secondary side. The steam superheat will fall below the setpoint and the actual SG load will be larger than the setpoint. The feedwater regulating valve and the turbine throttle valve will both close to try to match the superheat and load setpoints. Depending on the magnitude of the feedwater flow or steam flow increase and what caused the malfunction, this transient may lead to a turbine trip due to low superheat or a higher load than demanded. A reactor trip will also occur on low pressurizer level, low steam pressure, or high reactor power, and if the change in steam pressure causes the main steam and feedwater isolation valves to close, then the DHRS will be actuated.

Service Level B Transient 3 - Turbine Trip without Bypass

The turbine trip transient may be caused by any of several equipment or control system malfunctions. This transient covers the scenario where the turbine trip leads to a reactor trip. Once the turbine trips, the turbine stop valve shuts, stopping all steam flow and increasing steam pressure. The turbine bypass is postulated to be unavailable.

The RCS pressure and temperature increase due to the loss of heat removal and the pressurizer level rises due to the expanding RCS fluid. The reactor will trip and actuate both trains of the decay heat removal system (DHRS) to remove decay heat and cool the RCS. The reactor safety valves (RSVs) do not open.

Service Level B Transient 4 - Turbine Trip with Bypass

The turbine trip transient may be caused by any of several equipment or control system malfunctions. This transient covers the scenario when the turbine trips and the turbine bypass flow is available. After switching to bypass flow, the feedwater temperature decreases due to the feedwater heaters being offline. Reactor power stabilizes at its original level. The reactor does not trip. Reactor power is then decreased at a rate consistent with the Service Level A Transient 9-Load Ramp Decrease. Feedwater heating is not available throughout the power decrease.

Service Level B Transient 5 - Loss of Normal AC Power

A loss of normal AC power consists of a loss of AC power with no credit taken for the backup power supply system. Under these circumstances the reactor trips, the containment isolation valves fail closed, and the DHR actuation valves fail open. The module reaches a safe shutdown state by dissipating the heat through the DHR condensers. Batteries supply power to the five emergency core-cooling-system (ECCS) valves (three reactor vent valves (RVVs) and two reactor recirculation valves (RRVs)) that hold the valves closed. Once battery power is supplied to the ECCS valves a 24 hour timer begins. After 24 hours, battery power is removed and the RVVs and RRVs fail open. Actuation of the ECCS establishes a two-phase, natural circulation loop. Steam generated in the RPV exits through the RVVs and condenses on the walls of the CNV. The condensed water returns to the RPV through the RRVs. Coincident losses of the DC power systems, EDS and/or EDNS, as well as delays in MPS actuations, are considered to determine bounding pressure and temperature responses for mechanical design.

Service Level B Transient 6 - Inadvertent Main Steam Isolation Valve Closure

An inadvertent closure of an MSIV will cause a sudden decrease in the secondary-side flow for the affected SG and an increase in flow in the other SG. The closed MSIV causes the SG pressure to increase. The reactor trips on either high-steam pressure or high-pressurizer pressure.

The RSVs do not lift. Both trains of the DHRS are actuated. The DHRS removes heat through the two SGs and rejects the heat to the reactor pool. The components of the DHRS are sized to remove decay heat and cool the RCS.

Service Level B Transient 7 - Inadvertent Operation of the Decay Heat Removal System

The inadvertent operation of the DHRS could occur in two ways. The first is the inadvertent opening of one of the DHRS actuation valves. Opening an actuation valve allows flow between the DHRS condenser and the steam line as the steam and feedwater pressures equalize. The initial pressure equalization in the secondary side causes a disruption in the primary temperature. Both DHRS trains actuate and the reactor trips. The second way to inadvertent DHRS actuation is by the module protection system (MPS) sending a signal to actuate the DHRS by closing the MSIVs and feedwater isolation valves and opening the DHRS actuation valves on both trains of the DHRS. This results in the full-power operation of both trains of the DHRS. The DHRS actuation signal causes a reactor trip. The RSVs do not lift for either occurrence.

Service Level B Transient 8 - Reactor Trip from Full Power

A reactor trip from full power could be caused by multiple spurious sensor signals to the module protection system (MPS), or a spurious trip signal from the MPS, or miscellaneous failures that cause a reactor trip setpoint to be reached and are not already included in other transients. Once the trip begins, the control rods drop into the core to take the core subcritical. This reduces the core thermal power to decay heat and causes the hot- and cold-RCS temperatures to converge close to the average RCS temperature. Cooling is then initiated by one of two methods, either normal feedwater or actuating the DHRS. If the DHRS is actuated, then a containment isolation signal may also be generated. When circulating feedwater through the SGs, the steam produced is directed through the turbine bypass valve to the condenser. The steam and feedwater flow rates are controlled to keep the cooling rate below the maximum of 100 degrees F/hr (200 degrees F/hr in the pressurizer). This transient ends once the reactor reaches approximately steady hot shutdown conditions. Any cooldown from there is accounted for in the cycles of the cooldown from hot shutdown. If the DHRS is actuated for a more severe failure, heat is removed through the DHRS condenser to the pool.

Service Level B Transient 9 - Control Rod Misoperation

This transient includes misoperations of the control rod assemblies (CRAs), such as the drop of a single CRA, the drop of a bank of CRAs, withdrawal of a single CRA, or withdrawal of a CRA bank. The CRA adds significant negative reactivity to the core that quickly reduces reactor power. Such a reduction in power leads to a decrease in RCS temperature and pressure. The decreasing temperature leads to a reactivity insertion due to the negative moderator temperature coefficient. The reactor will likely trip on low pressure or pressurizer level. However, if the rod worth is low enough, the reactor may reach a new steady state at the original power level but with a lower average RCS temperature. Removal of decay heat is by feedwater flow.

Service Level B Transient 10 - Inadvertent Pressurizer Spray

The inadvertent pressurizer spray transient entails, either through equipment failure or operator error, actuation of continuous pressurizer spray. With the spray control valve fully open, spray flow at the maximum design flow and the minimum expected temperature is provided to the pressurizer. The pressurizer heaters energize to counteract the decrease in pressurizer pressure. A reactor trip on low pressurizer pressure will occur. The low pressurizer pressure also triggers containment isolation and actuates both trains of the DHRS. Once the reactor trips, it will take the operators some time to identify the failure that caused depressurization. Removal of decay heat is by the DHRS.

Service Level B Transient 11 - Cold Overpressure Protection

When the RPV is at low temperatures, the metal is more prone to brittle failure. To prevent this type of failure, lower maximum pressure limits are implemented when the RPV is at low temperature. Cold overpressurization could be caused by equipment malfunctions or operator error that cause excessive heat and/or inventory to be added to the RCS. The RVVs are providing protection against low-temperature overpressurization.

If the RCS is at or below the low-temperature overpressure protection enable temperature and the RCS pressure is at or above the low temperature overpressure protection pressure setpoint, the RVVs will open to relieve the pressure by blowing down to the CNV. Interlocks in the control system will prevent this action when the reactor coolant is above the low temperature overpressure protection enable temperature. When the RVVs open, all of the components within the RPV experience a rapid decrease in fluid pressure. The CNV pressure will increase as it receives coolant from the RPV and once the RCS pressure and CNV pressure reach equilibrium, the RRVs will be opened.

Service Level B Transient 12 - CVCS Malfunctions

This transient includes malfunctions of the CVCS that can cause an increase in RCS inventory or addition of cooler water to the RCS. An increase in RCS inventory could result from a spurious makeup pump operation, excessive charging, or a failure in the letdown line to compensate for the increase in inventory. These events could cause the pressurization of the RCS and a CVCS isolation or reactor trip will likely occur. If there is a malfunction of the pressurizer spray and the RCS pressure is high enough to reach the RSV setpoint, the RSVs will lift to release pressure. Another CVCS malfunction transient is possible if recirculation flow is stopped due to the malfunction of the CVCS recirculation pumps. A full or partial valve closure in the letdown line is also specified, which limits the amount of letdown flow. This would allow colder makeup water to be pumped to the RCS using the makeup pumps, with limited heat addition through the regenerative heat exchanger. Depending on the reactor power level and primary flow rate, the addition of colder makeup water could affect the reactivity, which could possibly result in a reactor trip on high reactor power.

3.9.1.1.3 Service Level C Conditions

Service Level C Transient 1 - Spurious ECCS Valve Actuation

The ECCS consists of three RVVs and two RRVs. In the event of an inadvertent actuation of an RVV or RRV, the inadvertent actuation block feature provides mechanical pressure-locking to prevent opening of the valve when the RCS and CNV are at normal operating pressure. However, if there is a failure of at least one of the inadvertent actuation block devices, there are two plausible scenarios involving a valve opening. The first scenario is the opening of a single RVV or RRV due to equipment malfunction. This event causes a decrease of RCS inventory due to the blowdown of RCS fluid to the CNV. The bounding operating condition for the opening of an RVV or RRV is full power operation. Once the trip valve fails open and the inadvertent actuation block fails to keep the main valve closed, the reactor trips likely on either high containment pressure or low pressurizer pressure. The high containment pressure would cause a containment isolation and DHRS actuation signal. The open valve allows reactor coolant to blow down into the CNV. As the hot steam contacts the CNV walls, it condenses to liquid and accumulates in the bottom of the CNV. The CNV wall is cooled by convection to the surrounding reactor pool.

The ECCS is actuated when either the liquid accumulating in the CNV reaches the high CNV water level setpoint or the RCS liquid level decreases to the low RCS level setpoint. All other RRVs and RVVs open, and this configuration establishes a two-phase, natural recirculation loop that provides cooling for the RCS through the RVVs and keeps the core covered by returning liquid to the RPV through the RRVs.

The second possible scenario is that operator error or a failure of the control system causes two ECCS valves to open and failure of the inadvertent actuation block devices on these valves. Since there are two RRVs and three RVVs, each RRV is paired with two RVVs and these pairs are actuated separately by two divisions of the engineered safety features actuation system (one RVV is expected to be shared by the two divisions). This scenario could therefore include the failure of two RVVs and their inadvertent actuation block devices, or the failure of one RVV and one RRV and their inadvertent actuation block devices. A failure of more than two ECCS valves at a time is considered beyond design basis. This transient develops in the same sequence as the first scenario, but the depressurization occurs more quickly.

Service Level C Transient 2 - Inadvertent Opening of a Reactor Safety Valve

The inadvertent opening of one of the RSVs causes the RCS to quickly depressurize as the primary coolant blows down to the CNV. The reactor will trip likely due to high containment pressure or low pressurizer pressure. The high containment pressure causes a containment isolation and DHRS actuation signal. The hot vapor entering the CNV will condense on the walls and fall to the bottom of the CNV. When either the low RCS or high CNV liquid level setpoints are reached, all five ECCS valves will open. The open valves establish the ECCS two-phase, natural recirculation loop. Decay heat is removed by the vapor moving through the RVVs to the CNV and the core is kept covered by the liquid returning to the RPV through the RRVs. Removal of decay heat is expected through the containment wall and peak pressure in the CNV is kept below design pressure.

Service Level C Transient 3 - CVCS Pipe Break

The CVCS Pipe Break is characterized by a rupture of a pipe penetrating the RCPB. The break could occur inside or outside of containment. A break inside containment maximizes the dynamic response of the RPV and RVI and captures a pressure and thermal cycle for the CNV and components inside containment. A break outside of containment could cause stresses on the components just outside of containment. In this transient, the RCS depressurizes through the break and the level in the pressurizer decreases. The reactor trips due to either low pressurizer pressure or level or high containment pressure, and the DHRS is actuated. The ECCS may eventually actuate based on either a low RCS liquid level or a high water level in containment. Removal of decay heat is expected through the containment wall and peak pressure in the CNV is kept below design pressure.

Service Level C Transient 4 - Steam Generator Tube Failure

The steam generator tube failure (SGTF) transient is bounded by the double-ended failure of a SG tube. The term "failure" is used here to include both a tube collapsing due to higher external pressure and a tube bursting due to higher inner pressure. Multiple simultaneous SGTFs are considered beyond design basis. In this transient, the RCS blows down into the SG. A reactor trip would occur quickly due to high steam pressure, low pressurizer pressure, or low pressurizer level. Both trains of the DHRS will be actuated to remove the decay heat as normal cooldown using feedwater flow is not possible with SGTF. A SGTF incapacitates one train of the DHRS, but cooldown is still accomplished with the other train. Components within the RPV will experience a decrease in pressure when the SG tube fails and the RCS blows down to the SG. Once the MSIVs and feedwater isolation valves close and the DHRS actuates, the pressure decrease will slow to be only a function of the RCS cooldown rate. The cooldown rate is determined by the performance of the single DHRS train.

3.9.1.1.4 Service Level D Conditions

Service Level D Transient 1 - Steam Piping Failures

A main steam line break could cover a wide range of break types. A rupture will cause an increase in steam flow rate and will reduce the SG inventory. A break inside containment is not postulated to occur because of leak before break detection on these lines. A break outside of containment could cause stresses on the components just outside of containment. RCS temperature and pressure briefly decrease due to the excess heat removal provided by the steam line blowdown. A break will quickly cause a reactor trip on low steam pressure or high containment pressure. Once the reactor is tripped, both trains of the DHRS will be activated. One train of the DHRS will be ineffective due to the break. A single train of the DHRS will be capable of removing the decay heat from the reactor. The RSVs do not lift and there is no ECCS actuation. Removal of decay heat is by the DHRS and peak pressure in the CNV is kept below design pressure.

Service Level D Transient 2 - Feedwater Piping Failures

A feedwater line break could cover a wide range of break types. Due to the interaction of the DHRS and feedwater system, the spectrum of feedwater piping breaks includes any breaks in the DHRS. A break inside containment is not postulated to occur because of leak before break detection on these lines. A break outside of containment could cause stresses on the components just outside of containment. RCS temperature and pressure briefly decrease due to the excess cooling provided by the feedwater line blowdown. Once the quick blowdown phase is over, the transient results in heating and pressurization of the RCS. A break will quickly cause a reactor trip on low steam pressure or high containment pressure. Once the reactor is tripped, both trains of the DHRS will be activated. One train of the DHRS will be ineffective due to the line break. A single train of the DHRS will be capable of removing the decay heat from the reactor. The RSVs do not lift and there is no ECCS actuation. Peak pressure in the CNV is kept below design pressure.

Service Level D Transient 3 - Control Rod Assembly Ejection

This transient covers a spectrum of possible control rod ejection scenarios in order to find the most limiting case. Scenarios must be considered at different power levels, fuel burnups, and rod configurations. The most reactive control rod for a given scenario is postulated to be ejected from the core. Removing the control rod causes a local reactivity insertion that leads to a pressure increase. Once the rod is ejected, there will be a delay before the module protection system trips the reactor. The trip could be caused by high reactor power or high-rate power change.

Service Level D Transient 4 - Combustible Gas Detonation

This transient covers the CNV and components necessary to maintain safe shutdown. The CNV and components must withstand the environmental conditions created by the burning of hydrogen within the first 72 hours of any design basis event and maintain containment structural integrity and safe shutdown capability.

A typical design basis event where combustible gas control is relevant is any event that results in ECCS actuation. Initiating events that result in ECCS operation include LOCAs, spurious valve openings, and a loss of all DC power. Regardless of the initiating event, the outcome is similar: the ECCS successfully actuates and maintains RPV liquid level above the top of the core. Because heat removal from the containment is very effective, temperatures will usually decrease rapidly. Subatmospheric pressure in the containment is expected within a few hours after event initiation.

Continued operation and long term cooling by the ECCS will result in a stable condition. Aside from temperature gradually approaching the reactor pool temperature, the only other long term change in the containment condition under ECCS operation is the accumulation of radiolytically generated gases. Radiolytic production of gases is capable of creating a flammable atmosphere soon after event initiation. As radiolytic production continues, a higher pressure flammable atmosphere becomes possible. At 72 hours after event initiation sufficient oxygen could be produced through radiolysis to create a flammable atmosphere.

The production of hydrogen and oxygen from radiolysis following a reactor shutdown and activation of ECCS, in combination with the low temperature and initial pressure of the containment, can lead to the formation of a combustible atmosphere. Once sufficient oxygen is produced and an ignition source is available deflagration and detonation could occur as well as a deflagration-to-detonation transition.

A deflagration propagates at subsonic speeds, resulting in a quasi-static pressurization of the CNV and SSCs inside containment. This event is best simulated as a suddenly applied force, which remains on the structure indefinitely.

Pressure reflection is not considered for subsonic events because these do not attain appreciable momentum to cause an amplified reflected pressure pulse. This is bounded by analysis of detonations.

A detonation results in spherically expanding pressure waves travelling at the Chapman-Jouguet (C-J) speed, leading to incident pressure waves that are twice the peak pressure of a deflagration. Reflected C-J pressure waves are further amplified upon impacting a hard surface. This is considered an ASME Level C event.

Deflagration-to-detonation transition (DDT) is a condition resulting when a gaseous mixture burns leading to flame acceleration that reaches a sonic or supersonic condition where the deflagration transitions to a detonation. If the DDT occurs near a reflecting surface, a significant amplification above the peak reflected C-J pressure is possible due to pre-compression of the unburned gases ahead of the shock front. This is considered an ASME Level D event.

The initial conditions are determined from the optimum temperature and pressure that produces the largest pressure pulse. The reactor has tripped, ECCS has been activated and the NPM is cooling to reactor pool temperature.

For design purposes, this transient is specified to occur one time over the 60 year life of the plant. The expected outcome is that the ECCS operation continues.

3.9.1.1.5 Test Conditions

Primary Side Hydrostatic Test

The initial primary side hydrostatic test consists of pressurizing the RPV and the reactor coolant system to a minimum of 125 percent of design pressure and a water temperature of at least 70 degrees Fahrenheit to a maximum of 140 degrees Fahrenheit. The testing complies with ASME BPVC Section III (Reference 3.9-1), Article NB-6000 and Non-mandatory Appendix G for the minimum testing temperature of RT_{NDT} +60 degrees Fahrenheit.

This hydrostatic test takes place in the fabrication shop prior to the first startup, with the RPV filled with water.

The RPV is designed for 10 cycles of this type of hydrostatic test. This hydrostatic test must be performed before the first startup and extra cycles are added in the event that significant repairs require additional tests.

The system leakage tests required by ASME BPVC Section XI are performed at nominal operating pressure and temperature, are not considered hydrostatic tests and, therefore, are not included in the number of occurrences.

Secondary Side Hydrostatic Test

The initial, secondary-side, hydrostatic test consists of pressurizing the secondary side to a minimum of 125 percent of design pressure and a water temperature of at least 70 degrees Fahrenheit to a maximum of 140 degrees Fahrenheit. The testing complies with ASME BPVC Section III (Reference 3.9-1), Article NB-6000 and Non-mandatory Appendix G for the minimum testing temperature of RT_{NDT} +60 degrees Fahrenheit.

This hydrostatic test takes place in the fabrication shop prior to the first startup, with the secondary side filled with water.

The secondary side is designed for 10 cycles of this type of hydrostatic test. This hydrostatic test must be performed before the first startup and extra cycles are added in the event that significant repairs require additional tests.

The system leakage tests required by ASME BPVC Section XI are performed at nominal operating pressure and temperature, are not considered hydrostatic tests and, therefore, are not included in the number of occurrences.

Containment Hydrostatic Test

The initial containment vessel hydrostatic test consists of pressurizing the containment vessel to a minimum of 125 percent of design pressure and a water temperature of at least 70 degrees Fahrenheit to a maximum of 140 degrees Fahrenheit. The testing complies with ASME BPVC Section III (Reference 3.9-1), Article NB-6000 and Non-mandatory Appendix G for the minimum testing temperature of RT_{NDT} +60 degrees Fahrenheit.

This hydrostatic test takes place in the fabrication shop prior to the first startup, with the CNV filled with water. If the CNV is hydrostatically tested with the RPV installed, the RPV (both primary and secondary sides) must be vented to the CNV to preclude a differential pressure external to the RPV.

The containment vessel is designed for 10 cycles of this type of hydrostatic test. This hydrostatic test must be performed before the first startup and extra cycles are added in the event that significant repairs require additional tests.

The system leakage tests required by ASME BPVC Section XI are performed at nominal operating pressure and temperature, are not considered hydrostatic tests and, therefore, are not included in the number of occurrences.

3.9.1.2 Computer Programs Used in Analyses

The computer programs used by NuScale in the dynamic and static analyses of mechanical loads, stresses, and deformations, and in the hydraulic transient load analyses of seismic Category I components and supports, are listed below.

The development, procurement, testing, and maintenance of computer programs used in these analyses are completed in compliance with an established quality-assurance program described in Chapter 17. Computer program acceptability is pre-verified or the results verified with the design analysis for each application. Pre-verified computer programs are controlled using a software configuration management process.

Methods of software design verification include: software design reviews, alternate calculations, and qualification testing.

In establishing its program for design control and verification, NuScale commits to compliance with NQA-1-2008 and NQA-1a-2009 addenda, Requirement 3, Sections 100 through 900, and the standards for computer software in NQA-1-2008 and NQA-1a-2009 addenda, Part II, Subpart 2.7 and Subpart 2.14 for Quality Assurance requirements for commercial grade items and services. Delegated responsibilities may be performed under an approved supplier's or principal contractor's quality assurance program, in which case the supplier is responsible for the control of computer programs used.

The following computer programs are used by NuScale.

ANSYS - The ANSYS Inc. ANSYS software package includes Mechanical, CFX, Fluent, ICEM CFD, Design Modeler, Workbench and Help. ANSYS is a pre-verified and configuration managed finite element analysis program used in the design and analysis of safety related components. The use of this program in structural and seismic analyses is discussed in Sections 3.7.2, 3.8.4, and its use in piping stress analyses is discussed in Section 3.12.4.

AutoPIPE - Bentley AutoPIPE Nuclear is pre-verified and configuration managed, and is used for performing stress analysis on piping systems throughout the NuScale power plant. This includes static and dynamic analyses for fluid and thermal transients and seismic accelerations. The use of this program in piping analysis is discussed in Section 3.12.4.

NRELAP5 - NRELAP5 is NuScale's proprietary system thermal-hydraulics code for use in safety-related design and analysis calculations, and is pre-verified and configuration managed. NRELAP5 is based on RELAP5-3D, a product of Idaho National Lab. The code permits simulation of single-phase or two-phase systems and includes many generic component models, which can be used in transient dynamic analyses. The development, use, verification, validation, and code limitations of this program are discussed in Section 15.0.2.

The following additional programs are used by suppliers:

RspMatch2009 - RspMatch2009 is commercially available from GeoMotions, LLC and is used for response spectral matching and adjusting seismic acceleration time histories.

SAP2000 - SAP2000 is commercially available from Computers and Structures, Inc. and performs finite element analysis for non-seismic load analyses and the design of structures. The use of this program is discussed in Sections 3.7.2 and 3.8.4.

SASSI2010 - SASSI2010 is commercially available and is used for the numerical analysis of the soil-structure interactions. The use of this program in seismic analyses is discussed in Sections 3.7.2 and 3.8.4.

SHAKE2000 - SHAKE2000 is commercially available and is used to calculate the strain-compatible soil properties and in-layer response acceleration time histories for the soil-structure interaction analyses of the NuScale structures.

EMDAC - EMDAC is a finite element analysis code produced by Curtiss-Wright Electro-Mechanical Division, and is used for the seismic structural analysis of the control rod drive mechanisms (CRDM).

Simulink - The Multiphysics Simulink computer program is used to simulate the operating dynamics of the CRDM, and is operated in a MatLab environment.

For computer programs used in Section 3.7 Seismic design, see Section 3.7.5.

3.9.1.3 Experimental Stress Analysis

Experimental stress analysis is not used for the NuScale Power Plant design.

3.9.1.4 Considerations for the Evaluation of Service Level D Condition

The analytical methods used to evaluate stresses for Seismic Category I systems and components subjected to Service Level D condition loading are described in Section 3.9.3.

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

This section presents the criteria, testing, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, and reactor internals and their supports under dynamic and vibratory loading, including those due to fluid flow during normal plant operation, transient conditions and postulated seismic events. Section 14.2 contains test abstracts that describe in general terms the planned tests that will be performed and describes the programmatic controls that will be used to develop the individual tests.

The NuScale Power Plant design complies with the relevant requirements of the following regulations, including the General Design Criteria (GDC) of 10 CFR 50, Appendix A:

- GDC 1 and 10 CFR 50.55a, as they relate to the testing of systems and components to quality standards commensurate with the importance of the safety-related functions to be performed. The Quality Assurance Program Description, in accordance with 10 CFR 50, Appendix B, addresses the quality standards applied to the dynamic testing and analysis of SSC.
- GDC 2 and 10 CFR 50, Appendix S, as they relate to structures, systems, and components (SSC) designed to withstand appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena without losing the ability to perform their safety functions. Pursuant to GDC 2, mechanical components are designed to withstand the loads generated by natural phenomena as discussed in Section 3.1.1.
- GDC 4 as it relates to SSC being appropriately protected against the dynamic effects of discharging fluids. As discussed in FSAR Section 3.6, the NuScale Power Plant design appropriately protects SSC against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, which may result from equipment failures and from events and conditions outside the nuclear power unit .

- GDC 14 as it relates to SSC of the RCPB being designed to have an extremely low probability of rapidly propagating failure or of gross rupture. Section 3.9.2 addresses dynamic testing of components of the reactor coolant pressure boundary to ensure that they will withstand the applicable design-basis seismic and dynamic loads in combination with other environmental and natural phenomena loads without leakage, rapidly propagating failure, or gross rupture.
- GDC 15 as it relates to the reactor coolant system being designed with sufficient margin to ensure that the RCPB is not breached during normal operating conditions, including AOOs. The RCPB is designed to resist seismic, LOCA, and other environmental loads. Dynamic analyses are described to confirm the structural design adequacy of the reactor coolant pressure boundary. Vibration, thermal expansion, and dynamic effects testing are also described to verify the design.
- 10 CFR 50, Appendix B, as it relates to quality assurance in the dynamic testing and analysis of systems, structures, and components. The NRC-approved NuScale Quality Assurance Program Description discussed in Section 17.5 satisfies the requirements of 10 CFR 50, Appendix B, to ensure that SSCs are designed, procured, fabricated, inspected, erected, and tested to standards commensurate with their contribution to plant safety.

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

Piping systems can be damaged by thermal expansion and vibrations due to transient events such as pipe breaks, valve closure, etc. This section addresses the pre-operational testing, and initial startup testing that is performed to verify that the vibrations and thermal expansion and contraction of the as-built piping systems are bounded by the design requirements. The piping systems tested include:

- ASME BPVC, Section III (Reference 3.9-1), Class 1, 2, and 3 piping systems identified in Table 3.2-1.
- other high-energy piping systems inside Seismic Category 1 structures or those whose failure would reduce the functioning of any Seismic Category I plant feature to an unacceptably level. See Section 3.6.1.
- Seismic Category I portions of moderate-energy piping systems located outside of containment identified in Table 3.2-1 and Section 3.6.1.

The test program, as described in Section 14.2, verifies that the Class 1, Class 2, Class 3, and other high-energy and Seismic Category 1 piping systems meet functional design requirements and that piping vibrations and thermal expansions are within acceptable levels and will withstand dynamic effects due to operating transients. Piping systems are validated through a series of checks, inspections, and tests, as follows:

- The first validation step is during the manufacturing process at the manufacturing facility and during the construction. The piping systems and other components are inspected to verify the correct assembly and to record the initial positions under cold conditions.
- The second validation step is plant heat up, whereupon the plant is heated to normal operating temperatures. Expansion and contraction of the systems and

components is monitored and recorded to verify that it is within the assumed conditions identified in the analyses.

- The third validation step is performance tests. The systems are operated to verify the performance of critical SSCs such as valves, controls, and auxiliary equipment. This phase of testing includes transient tests as outlined in Chapter 14 to identify unacceptable expansion and contraction, noise, vibration, and stresses which are not bounded by the design analyses.

The initial test program is described in Section 14.2. The vibration, thermal expansion, and dynamic effect elements of this test program, summarized below, are performed during Phase I pre-operational testing and Phase II initial startup testing.

Phase I - Pre-operational Testing

Preoperational tests are performed to demonstrate that the piping system components meet functional design requirements, and that piping vibrations and thermal expansions and contractions are bounded by the analyses. If the design basis parameters are not bounding compared to the measured values, then corrective actions (i.e. reanalyzing with as-built values) are implemented and the systems are retested.

Phase II - Initial Startup Testing

Initial startup testing is performed after the reactor core is loaded into a reactor module. These Phase II tests establish that the vibration level and piping reactions to transient conditions are acceptable and bounded by the analyses. If the vibration levels are not bounded, the analyses use the vibration level from the testing as input and verify that the design is acceptable.

3.9.2.1.1 Piping Vibration Details

3.9.2.1.1.1 Piping Included in Comprehensive Vibration Assessment Program

ASME Code Class 1, 2, and 3 piping systems that are part of the reactor module are included within the scope of the NuScale Comprehensive Vibration Assessment Program (CVAP) (Reference 3.9-5) Piping systems that meet the screening criteria for applicable flow induced vibration mechanisms are evaluated in the analysis program. If a large margin of safety is not demonstrated, prototype testing is performed in accordance with the CVAP measurement program.

Reactor module components, piping, and supports with a high degree of safety margin are excluded from testing in the prototype measurement program, consistent with the overall measurement program objectives of validating relevant analytical inputs, results, and margins of safety. Therefore, comprehensive vibration testing of all ASME Code Class 1, 2, and 3 piping is not performed.

3.9.2.1.1.2 Piping Not Included in Comprehensive Vibration Assessment Program

For ASME Code Class 3 piping that is not part of the reactor module (there is no Code Class 1 or 2 piping which is not part of the reactor module) and other ASME B31.1 piping outside of containment which requires vibration testing, vibration test specifications are developed in accordance with ASME OM-S/G, Division 2 (OM Standards), Part 3 (Reference 3.9-3). SRP 3.9.2 recommends using this part of the ASME OM Code for developing preoperational vibration test specifications. Piping vibration testing and assessment are performed in accordance with ASME OM-2012, Division 2 (OM Standards), Part 3 (Reference 3.9-3).

The Phase I and II tests demonstrate that the piping systems withstand vibrations resulting from Service Level A loads and Service Level B loads.

Service Level A vibration loads are sustained loads encountered during normal plant startup, operation, refueling, and shutdown. These vibration loads are continuous or steady state over a period of time. If excessive vibration is observed which is outside the bounds of the analyses, a re-analysis to determine the cause and to identify the corrective action is performed.

Service Level B loads are infrequent loads with a high probability of occurrence but which cause no damage or reduction in component function. The vibrations are the result of valve operation, pumps, and other loads from transients. If excessive vibration is observed which is outside the bounds of the analyses, a re-analysis to determine the cause and to identify the corrective action is performed.

The Phase I and Phase II tests do not address vibrations resulting from Service Level C or Service Level D loads.

3.9.2.1.2 Piping Thermal Expansion Details

Thermal expansion testing verifies that the design of the piping systems tested prevents constrained thermal contraction and expansion during service level A and B transient events. The tests also provide verification that the component supports can accommodate the expansion of the piping for the service levels for these modes of operation. Section 14.2 provides descriptions of selected planned piping thermal expansion measurement tests. Test specifications for thermal expansion testing of piping systems during preoperational and start-up testing will be in accordance with ASME OM Standard (Reference 3.9-3), Part 7.

3.9.2.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

This section describes the seismic system analysis and qualification of Seismic Category I SSC identified in Section 3.2, Table 3.2-1, performed to confirm functional integrity and operability during and after a postulated seismic event. Seismic design criteria for the NuScale Power Module (NPM) is addressed in Appendix 3.A.

3.9.2.2.1 Seismic Qualification Testing

The methods and criteria for seismic qualification testing of Seismic Category I mechanical equipment are described in Section 3.10.

3.9.2.2.2 Seismic System Analysis Methods

Methods for seismic analysis of SSCs including piping are addressed in Section 3.7, Section 3.10, Section 3.12 and Appendix 3.A.

3.9.2.2.3 Determination of Number of Earthquake Cycles

See Section 3.7.3.

3.9.2.2.4 Basis for Selection of Frequencies

See Section 3.7.3.

3.9.2.2.5 Three Components of Earthquake Motion

See Section 3.7.2 and Section 3.12.

3.9.2.2.6 Combination of Modal Responses

See Section 3.7.2, Section 3.12 and Appendix 3.A.

3.9.2.2.7 Analytical Procedures for Piping

See Section 3.12.

3.9.2.2.8 Multiple-Supported Equipment Components with Distinct Inputs

See Sections 3.7.3 and Section 3.12.

3.9.2.2.9 Use of Constant Vertical Static Factors

See Section 3.7.3.

3.9.2.2.10 Torsional Effects of Eccentric Masses

See Sections 3.12 and 3.7.3.

3.9.2.2.11 Buried Seismic Category I Piping and Conduits

ASME Code Class 2 and Class 3 Seismic Category I buried piping in the NuScale Power Plant design is analyzed as discussed in Section 3.12.

3.9.2.2.12 Interaction of Other Piping with Seismic Category I Piping

See Section 3.12.

3.9.2.2.13 Analysis Procedure for Damping

See Section 3.7.3.

3.9.2.2.14 Test and Analysis Results

See Section 3.9.2.2.1 and Section 3.9.2.2.2 above.

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

Flow-induced vibration (FIV) behaviors and characteristics are complex and require both analysis and testing to assess the vibrational responses. NuScale has developed a CVAP (Reference 3.9-5) in accordance with Regulatory Guide (RG) 1.20 to verify the structural integrity of the NPM components to FIV. The NuScale CVAP (Reference 3.9-5) documents the analytical evaluation of NPM components determined to be susceptible to FIV and identifies how the analytical results are verified by vibration measurement and inspection during pre-operational and initial startup testing.

The NuScale Power Module represents a first-of-a-kind design in its size, arrangement, and operating conditions, although its technology is based on well-proven light water reactor designs with long operational experience. Accordingly, the first operational NPM is classified as a prototype in accordance with RG 1.20. After the first NPM is qualified as a valid prototype, subsequent NPMs will be classified as non-prototype category I.

Evaluation of flow-induced vibration (FIV) for commercial SGs and pressurized water reactors (PWRs) has been well documented. As such, FIV mechanisms and the relevant structural and fluid characteristics that increase FIV risk are readily identified from open source references. NPM components are screened against the various FIV mechanisms, and analysis is performed to determine component susceptibility. The NPM components that were shown to be susceptible based on the screening criteria are discussed in the CVAP (Reference 3.9-5). Due to the first-of-a-kind NPM design, component screening analysis errs on the side of including potentially susceptible components, even when they could be excluded based on engineering judgment or precedent. This minimizes the risk of failing to analyze a significant component. Compared to the existing PWR and BWR designs, the natural circulation design of the NPM is inherently less susceptible to FIV due to the lower primary coolant velocities. Based on these two factors, FIV analysis results demonstrate that many components have very large margins of safety.

To validate the FIV inputs, analytical results, and the margins of safety determined in the analysis program, a combination of separate effects, pre-operational and initial startup testing are performed. Separate effects testing is performed on a fully-prototypic portion of the design. Pre-operational testing is performed with the NPM components prior to fuel loading, at any time during module construction when the testing can be assured to accomplish the objectives of the measurement program. Initial startup testing is performed under full power normal operating conditions, after fuel loading. The results of all three testing types are used to validate the prototype NPM design. The CVAP (Reference 3.9-5) demonstrates that the NPM components for the NuScale Power Plant integrated pressurized water reactor are not expected to be subject to unacceptable flow-induced vibrations.

3.9.2.4 Flow-Induced Vibration Testing of Reactor Internals Before Unit Operation

A Comprehensive Vibration Assessment Program (CVAP) (Reference 3.9-5) for the NuScale Power Module (NPM) is established in accordance with the NRC Regulatory Guide (RG) 1.20. The CVAP ensures that the structural components of the NPM exposed to fluid flow are precluded from the detrimental effects of flow induced vibration (FIV). Given its prototype classification, the NuScale CVAP addresses the applicable criteria of RG 1.20, Section 2. The CVAP establishes the scope of analyses, testing, and inspections required to ensure that components of the NPM are not subject to unacceptable vibratory degradation.

A vibration test program for the NPM is conducted to validate the analysis program. The prototype testing consists of separate effects, factory, and initial startup tests. The testing results are used to validate the FIV analysis results and non-trivial analysis inputs, and to confirm that unacceptable vibratory response is precluded under steady state and transient operating conditions. The CVAP (Reference 3.9-5) is focused on confirming acceptable performance of the NPM components that are susceptible to FIV for all steady-state and transient operating conditions. This includes three main program components:

- analysis of the susceptible NPM components for applicable FIV mechanisms.
- pre-test predictions of the testing results, including experimental result ranges that account for uncertainties due to operating conditions, manufacturing tolerances, and instrument error. Pre-test predictions demonstrate the range of acceptable experimental results that can be used to validate analysis inputs and results.
- post-test analysis that verifies that the results fall within the pre-test predictions.

During FIV testing, NPM components are subjected to an operating time that results in cyclic loading of greater than one million cycles. This requirement is to address components that are affected by turbulent buffeting FIV mechanisms. To support the validation of analytical results related to this FIV mechanism, testing is performed until one million cycles of vibration are achieved for the most limiting (low structural natural frequency) NPM component. This is expected to take less than two days, depending on the operating conditions during the initial startup testing. The factory and initial startup testing performed to validate analysis results and non-trivial analysis inputs, and to confirm that unacceptable vibratory response is precluded, is documented in the NuScale CVAP (Reference 3.9-5).

Prior to and following initial startup testing, components are inspected for mechanical wear and signs of vibration induced damage. Initial startup testing provides a sufficient duration for the limiting NPM component to experience a minimum of one million cycles of vibration. All components that are evaluated in the analysis program undergo inspection. For the components validated in the measurement program via testing, the inspection provides a secondary confirmation of the FIV integrity of the NPM components. For components that do not require testing due to large safety margins, the inspection confirms that the testing performed on more limiting components sufficiently bounds the performance of the non-tested components.

Based on acceptable completion of the CVAP analysis, measurement and inspection program for the prototype NPM, subsequent NPMs are classified as non-prototype Category I.

COL Item 3.9-1: A COL applicant that references the NuScale Power Plant design certification will submit the results from the comprehensive vibration assessment program for the NuScale Power Module, in accordance with Regulatory Guide 1.20.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Service Level D Conditions

Appendix 3.A includes the dynamic system analysis of the reactor internals under service level D conditions.

Appendix 3.A provides details of the structural and dynamic analysis. The dynamic analysis for Level D service condition events considers safe shutdown earthquake (SSE) events and pipe rupture conditions. Section 3.9.3 defines the loads and loading combinations for components and the RVIs.

The dynamic model used for the blowdown analysis includes the CNV, the RPV, lower RVI, upper RVI, and the control rod drive mechanisms (CRDMs). See Appendix 3.A for a representative diagram of the model and additional information regarding the dynamic loading analysis of this model. Note that certain pipe breaks are not considered due to the application of leak-before-break methodology (see Section 3.6.3).

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

The results of analysis of the reactor vessel internals and other NPM components and supports are compared to the results of the prototype tests to verify the analytical models provide appropriate results. If the predicted responses differ significantly from the measured values during the testing, the calculated vibration responses are re-analyzed (including updates to models as needed) and reconciled with the measured vibration response.

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

Pressure-retaining components, core support structures, and component supports that are safety-related are classified as Class A, B, or C (see subsection 3.2.2) and are constructed according to the rules of the ASME BPVC, Section III, (Reference 3.9-1), Division 1. As noted in subsection 3.2.2, Class A, B, and C mechanical components meet the requirements of ASME Code Classes 1, 2, and 3, respectively. This section discusses the structural integrity of pressure-retaining components, their supports, and core support structures which are designed in accordance with the rules of ASME BPVC, Section III (Reference 3.9-1), Division 1 and GDC 1, 2, 4, 14, and 15.

The NuScale Power Plant design complies with the relevant requirements of the following regulations including General Design Criteria of 10 CFR 50, Appendix A:

- GDC 1 and 10 CFR 50.55a, as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety-related function to be performed. The design is in accordance with the applicable codes required in 10 CFR 50.55a as stated in Section 3.1. Section 3.2 provides quality group classifications of structures and components.
- GDC 2 and 10 CFR 50, Appendix S, as they relate to safety-related structures and components being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions without loss of capability to perform their safety functions. Pursuant to GDC 2, mechanical components are designed to withstand the loads generated by natural phenomena as discussed Section 3.1.1.
- GDC 4 as it relates to structures and components 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 loss-of-coolant accidents.
- 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.
- GDC 15 as it relates to the reactor coolant system (RCS) and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during conditions of normal operation, including anticipated operational occurrences.

ASME BPVC, Section III (Reference 3.9-1), requires that a design specification be prepared for ASME Class 1, Class 2, and Class 3 components. The ASME BPVC also requires design reports for all Class 1, Class 2, and Class 3 components, piping, supports, and core support structures are prepared which document that the as-designed and as-built configurations satisfy the requirements of the respective ASME design specification.

The NuScale Power Plant design is consistent with the 2013 ASME Code, Section III (Reference 3.9-1), Division 1, with the applicable addenda subject to the limitations and modification identified in 10 CFR 50.55a(b)(1). The piping analysis criteria and methods, modeling techniques, and pipe support criteria are described in Section 3.12.

COL Item 3.9-2: A COL applicant that references the NuScale Power Plant design certification will develop design specifications and design reports in accordance with the requirements outlined under American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III (Reference 3.9-1). A COL applicant will address any known issues through the reactor vessel internals reliability programs (i.e. Comprehensive Vibration Assessment Program, steam generator programs, etc.) in regards to known aging degradation mechanisms such as those addressed in Section 4.5.2.1.

3.9.3.1 Loading Combinations, System Operating Transients, and Stress Limits

The integrity of the pressure boundary of safety-related components is provided by the use of the ASME Code. Using the methods and equations in the ASME Code, stress levels in the components and supports are calculated for various load combinations. These load combinations may include the effects of internal pressure, dead weight, thermal expansion, dynamic loads due to seismic motion, and other loads.

This section describes and defines the design, test, and service level loadings and loading combinations used for the design of ASME Class 1, Class 2, and Class 3 components, supports, and core support structures. The loading combinations and corresponding stress limits for ASME Code design are defined for the design condition, service levels A, B, C, and D, and test conditions.

Section 3.9.1 lists the design transients and number of events used in fatigue analyses. Load combinations used to evaluate piping and supports are described in Section 3.12.

Certified design reports, in accordance with NCA-3551.1 and Non-mandatory Appendix C of Reference 3.9-1, shall be prepared which cover all ASME Class 1, Class 2, and Class 3 components, piping, and supports except as provided in NCA-3551.2 and NCA-3551.3. Per Appendix C, the presentation of analysis in the design reports will include the results of thermal, structural, and fatigue evaluations. These results will include tabulated stresses and cumulative usage factors for each area of investigation, and descriptions of the areas with the maximum stress for the design conditions or for any specified transients. The recommendations of RG 1.206 regarding listing all stress values within 10 percent of the allowable value and listing the contributions of each of the individual loads (e.g., deadweight, pressure, seismic, etc.) are not necessarily used unless it is determined to be beneficial to the interpretation of the results.

3.9.3.1.1 Loads for Components, Component Supports, and Core Support Structures

This section describes the loads considered in the design of components, component supports, and core support structures. Loads used for piping analysis are described in Section 3.12. Section 3.9.1 provides the design transients and number of events used in fatigue analyses.

Pressure

Pressure loading is identified as either design pressure or operating pressure. The term operating pressure (P) is associated with service levels A, B, C, and D conditions and is the highest pressure during an applicable transient and may be internal or external. The criteria for incorporating the effects of both internal and external pressures for components are described in the ASME BPVC Code, Section III (Reference 3.9-1).

Deadweight

Deadweight analyses consider the weight of the component, piping, or structure being analyzed, including the weight of the internal fluid, external insulation, and other appurtenances. For piping and components, the deadweight present during hydrostatic test loadings is also considered where such loadings exceed the normal operational deadweights.

Thermal Expansion

The effects on components, piping, and supports from restrained thermal expansion and contraction is considered in the design. Various operating modes are considered in order to determine the most severe thermal loading conditions. The zero thermal load temperature is taken as 70 degrees Fahrenheit.

Seismic

Analyses of seismic loads on ASME Class 1, Class 2, and Class 3 components and supports are described in Appendix 3.A. The number of SSE stress cycles included in the fatigue analysis is identified in Section 3.7.3. The fatigue effects of such cyclic events are also considered in the design of Class 1 components, piping, and supports.

System Operating Transients

Descriptions of system operating transients are included in Section 3.9.1.1. Variations in fluid temperature, pressure, and flow are provided as inputs for the analysis of these transients. Additionally, the number of cycles for each transient is provided to facilitate fatigue evaluations that are performed as required by the ASME BPVC Code (Reference 3.9-1).

Other transient loads that are considered are those due to rapid actuation of control valves and pumps, check valve closure, pump and turbine trips, and relief valve actuation. These events may cause various dynamic fluid loads such as water or steam hammer, thrust forces, dynamic pressures from blowdown, and cavity pressurization. Water and steam hammer loads primarily affect piping and are discussed in more detail in Section 3.12. Thrust forces due to the actuation of relief valves which are located on piping are discussed in more detail in Section 3.12. Thrust forces, blowdown, and cavity pressurization resulting from the actuation of the ECCS and relief valves located on the RPV and pipe breaks are discussed in more detail in Appendix 3.A.

Wind

For the NuScale Power Plant design, as all Class 1, Class 2, and Class 3 components and piping are located within the Reactor Building, which is designed to withstand the effects of natural phenomena; no wind or missile loading due to hurricanes or tornadoes are applicable. The only exception to this is the ultimate heat sink makeup water line which is a non-safety related ASME Class 3 line that is routed outside the building. This line will consider loading due to natural phenomena.

Pipe Break

Loads due to high-energy pipe breaks can take the form of pipe whip, jet impingement, elevated ambient temperatures, thrust forces, dynamic pressure transients associated with blowdown of the system, and cavity pressurization. Pipe whip and jet impingement are mitigated using special restraints which are discussed in detail in Section 3.6. Dynamic pressure transients in piping are discussed in Section 3.12. Loading on components due to thrust forces, dynamic pressure transients associated with blowdown, and cavity pressurization are discussed in Appendix 3.A.

Thermal Stratification, Cycling, and Striping

Thermal stratification, cycling, and striping (including applicable NRC Bulletins 79-13, 88-08, and 88-11) are discussed in Section 3.12.

Friction

Frictional forces induced by the pipe on the support develop when sliding occurs across the surface of a support member in the unrestrained direction(s) due to thermal expansion and contraction. Since friction is due to the gradual movement of the pipe, loads from friction are calculated using the only the deadweight and thermal loads normal to the applicable support member. Friction due to other piping loads is not considered.

Environmentally Assisted Fatigue

A fatigue analysis is performed in accordance with ASME BPVC Section III (Reference 3.9-1) Subsections NB-3200, or NG-3200 considering the effects of the light-water reactor environment in accordance with RG 1.207 and NUREG/CR-6909.

The effects of the environment on fatigue for Class 1 piping and supports are addressed in Section 3.12.

3.9.3.1.2 Load Combinations and Stress Limits

The RPV is a Seismic Category 1, ASME Section III, Class 1 component. The load combinations and stress limits for the RPV and its supports are presented in Table 3.9-3.

The CNV is a Seismic Category 1 component. The ASME classification of the CNV and its supports is described in Section 3.8.2.2. The load combinations and stress limit for CNV and its supports are presented in Table 3.9-4

The RVI are Seismic Category 1 components. Portions of the RVI, which perform a core support function, are classified as Class CS components in accordance with ASME Section III, Subsection NG. The remaining portions of the RVI are designated as internal structures; however, they are designed using NG-3000 as a guide and constructed to ASME Subsection NG. The load combinations and stress limit are presented in Table 3.9-5.

The portions of the CRDM providing a RCPB function are ASME Code Class 1, Seismic Category I components. The CRDM coil heat exchangers, tubes, and connections, which provide cooling water and are external to the RCPB, are ASME Code Class 2, Seismic Category I components. The CRDM pressure housing is a Class 1 appurtenance per ASME BPVC, Section III, NCA-1271. The load combinations and stress limit are presented in Table 3.9-6. The CRDM seismic supports located on both the RPV and CNV head are ASME Code Class 1, Seismic Category I component supports.

The DHRS condensers are Seismic Category I components and are classified as ASME Section III, Class 2 components. The condenser supports are classified as ASME Section III, Subsection NF, Class 2 supports. The load combinations and stress limit are presented in Table 3.9-7.

Load combinations for the ECCS valves, containment isolation valves, RSVs, thermal relief valves and the DHRS actuation valves are presented in Table 3.9-9 through Table 3.9-14.

ASME Class 1 Piping

The loading combinations and corresponding stress design criteria per ASME service level for ASME Class 1 piping are presented in Table 3.12-1 in Section 3.12.

ASME Class 2 and 3 Piping

The loading combinations and corresponding stress design criteria per ASME service level for ASME Class 2 and Class 3 piping are presented in Table 3.12-2 of Section 3.12.

Core Support Structures

The core support structures are designed to ASME BPVC Section III Subsection NG. The SG tube supports are internal supports and, therefore, are designed to the same criteria as the core support structure. The loading combinations and corresponding stress design criteria per ASME service level for ASME core support structures are consistent with the RVI load combinations and acceptance criteria (see Table 3.9-5).

ASME Class 1, 2, and 3 Component Supports

The ASME Class 1, Class 2, and Class 3 components and piping supports are designed in accordance with ASME BPVC Section III, Subsection NF. These supports include the CNV support skirt, the CNV lugs, the top support structure mounting assemblies, the RPV support plate/gusset, the DHRS condenser supports, the top support structure, and the CRDM seismic support structure. The load combinations are included in Table 3.9-3, Table 3.9-4, Table 3.9-7 and Table 3.9-8. The allowable stress criteria are supplemented by RGs 1.124 and 1.130 for Class 1 linear-type and plate-and-shell-type support structures.

The top support structure is mounted to the CNV top head, and it provides support for piping systems and valves attached to penetrations in the CNV top head and for electrical cables and conduit for various equipment in the NPM. It is a Seismic Category 1 component and classified as an ASME III, Subsection NF Class 2 support. The ASME BPVC Code analysis is in accordance with NF-3350 and it is designed to withstand the service loads and loading combinations specified in Table 3.9-8.

ASME Class 1, 2, and 3 Pipe Supports

The loading combinations and corresponding stress design criteria per ASME service level for ASME Class 1, Class 2, and Class 3 pipe supports is provided in Table 3.12-3 in Section 3.12.

3.9.3.2 Design and Installation of Pressure Relief Devices

ASME Class 1 Pressure Relief Valves

The RCS reactor safety valves (RSV) are designed as ASME BPVC Code, Section III, Class 1 pressure-relief, pilot-operated devices. They are part of the RCPB and are located on the RPV head. There are two RSVs, which are not connected to any piping on their discharge sides and vent directly into the CNV. The RSV function is to prevent RCS pressure from exceeding 110 percent of design pressure under normal and abnormal conditions and to prevent exceeding service limits. The two valves, each with sufficient capacity to limit overpressurization of the RPV, are normally closed, low leakage, and are used infrequently. The RCS and pressurizer steam space are sized to avoid an RSV lift for anticipated transients (see Section 5.2.2). RSVs are designed to withstand vertical and lateral loading due to seismic ground accelerations considering the appropriate damping values for pressure boundary valve bodies.

The ECCS valves are located on the RPV and are part of the RCPB. These valves are normally closed during startup, shutdown, and power operation; however, are normally open during refueling. They are remotely actuated by a module protection system (MPS) signal, loss of power, or operator action, to allow flow between the RPV and CNV. The ECCS valves are Seismic Category I components and designed as ASME BPVC Section III Class 1 components. They are also classified as Category A valves per the ASME OM Code. The ECCS valves are discussed in detail in Section 6.3.

ASME Class 2 Pressure Relief Valves

Each NPM contains two thermal relief valves in the FW piping in the (SG) system and one thermal relief valve in the control rod drive system (CRDS) cooling piping. These thermal relief valves are classified as ASME III Class 2 relief valves per Reference 3.9-1. They are Seismic Category 1 Components.

Thermal relief valves are designed to withstand vertical and lateral loading due to seismic ground accelerations considering the appropriate damping values for pressure boundary valve bodies.

The function of thermal relief valves is to prevent system pressure from exceeding 110 percent of design pressure under normal and abnormal conditions and to prevent exceeding service limits specified in the applicable component Design Specifications.

The SG thermal relief valves are installed in the feedwater (FW) piping and provide overpressure protection during water-solid conditions that may occur during NPM shutdown.

The CRDS cooling system thermal relief valve provides overpressure protection for the CRDS cooling piping during a containment isolation event during plant operation.

Pressure Relief Device Discharge System Design and Analysis

The design of the pressure relief valves uses the guidance of ASME BPVC Code Section III, Appendix O, "Rules for the Design of Safety Valve Installations," with respect to calculation of reaction loads. The reaction forces and moments are based on a static analysis with a dynamic load factor of 2.0 unless a justification is provided to use a lower dynamic load factor. A dynamic structural analysis may also be performed to calculate these forces and moments. The safety or relief valves that discharge directly to the atmosphere or containment are considered open-discharge configurations. The analysis requirements for these devices are addressed in Section 3.12.

3.9.3.3 Pump and Valve Operability Assurance

The NuScale Power Plant does not rely on pumps to perform any safety-related functions. A listing of the active safety related valves is provided in Section 3.9.6.

Active valves are subject to factory tests to demonstrate operability prior to installation. These tests are followed by post-installation testing in the plant. The factory- and post-installation tests performed are described in the inservice testing (IST) program. The IST requirements for ASME Class 1, Class 2, and Class 3 components are contained in the ASME Operation and Maintenance (OM) Code (Reference 3.9-3).

A description of the functional and operability design and qualification provisions and IST programs for safety-related valves is provided in Section 3.9.6. Environmental qualification of safety-related valves is discussed in Section 3.11. The seismic qualification of safety-related valves is performed in accordance with ASME QME-1 (Reference 3.9-4) as endorsed by RG 1.100, Revision 3 and as discussed in Section 3.10.

The stress limits are discussed in Section 3.9.3.1.

3.9.3.4 Component Supports

Section 3.9.3.1 provides the load combinations, system operating transients, and stress limits for component supports.

As described in Section 3.9.3.3, the functionality assurance, environmental and seismic qualification programs that are applied to components are also applied to the associated supports.

3.9.4 Control Rod Drive System

The control rod drive system (CRDS) consists of the control rod drive mechanisms (CRDMs), and related mechanical components that provide the means for control rod assembly (CRA) insertion into the core as described in Section 4.6, as well as the rod position indication to the module control system. The CRDM control cabinets, rod position indication cabinets and associated cables, plus the CRDS cooling water piping inside containment are also part of the CRDS. The CRDM is an electro-magnetic device which moves the CRA in and out of the nuclear reactor core and is connected to two independent rod position indication trains. The CRDS provides one of the independent reactivity control systems as discussed in GDC 26 and NuScale Principal Design Criteria (PDC)-27.

The control rods and their drive mechanisms are capable of reliably controlling reactivity under conditions of normal operation, including AOOs, or under postulated accident conditions. The CRDM internals, consisting of the latch mechanism and drive shaft are, therefore, safety related. A positive means of insertion of the control rods is always maintained and, combined with the design of the CRDS, a margin of safety is provided that accommodates postulated malfunctions such as stuck rods.

The CRDM internals that ensure positive CRA insertion consist of the latch mechanism and control rod drive shaft and are classified as safety related and risk significant. Portions of the CRDS are a part of the RCPB (specifically the pressure housings of the CRDMs) and are safety related. The system is designed, fabricated, and tested to quality standards commensurate with the safety-related functions to be performed. The design complies with the codes and standards in accordance with 10 CFR 50.55a. This provides assurance the CRDS is capable of performing its safety-related functions by withstanding the effects of AOOs, postulated accidents, and natural phenomena, such as earthquakes, as discussed in GDC 1, 2, 14, 26, 29 and PDC-27.

The structural materials of construction for the CRDS are discussed in detail in Section 4.5.1. Materials for the pressure boundary portions of the CRDM are discussed in Section 5.2.3.

The NuScale Power Plant design complies with the relevant requirements of the following General Design Criteria (GDC) of 10 CFR 50, Appendix A and NuScale Principal Design Criteria (PDC):

- GDC 1 and 10 CFR 50.55a, as they relate to the CRDS being designed to quality standards commensurate with the importance of the safety functions to be performed. The NuScale quality assurance program satisfies the requirements of 10 CFR 50 Appendix B and ASME NQA-1 "Quality Assurance Requirements for Nuclear Facility Applications." As such the NuScale QA program provides confidence that the SSC, including CRDS that are required to perform safety functions, will perform the functions satisfactorily.
- GDC 2, as it relates to the CRDS being designed to withstand the effects of an earthquake without loss of capability to perform its safety-related functions. See Section 3.2 for the seismic classification of the CRDS in accordance with RG 1.29. The seismic analysis is performed for the CRDM to ensure that the components can withstand the effects of natural phenomena without loss of capability to perform their safety functions. Dynamic analysis of the CRDM is performed for the SSE event to

ensure that pressure integrity is maintained during the SSE and the capability to lower the CRA connect to the CRDM drive shaft is not compromised.

Seismic qualification is performed for the CRDS electrical and instrumentation and controls components to ensure that the CRDM electrical and instrumentation and controls equipment can fully operate after the seismic event. Additional protection against the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and tsunamis, is provided by locating the CRDS components inside the Reactor Building, which is a Seismic Category I building.

- GDC 14, as it relates to the RCPB portion of the CRDS being designed, constructed, and tested for the extremely low probability of leakage or gross rupture. The pressure-retaining components are seismically and environmentally qualified, ensuring components RCBP is maintained.
- GDC 26, as it relates to the CRDS being one of the independent, reactivity-control systems that is designed with appropriate margin to assure its reactivity control function under conditions of normal operation including AOOs. The CRDS facilitates reliable operator control by performing a safe shutdown (i.e., reactor scram) by gravity dropping of the CRA on a reactor trip signal or loss of power. The CRDS is designed such that core reactivity can be safely controlled and that sufficient negative reactivity exists to maintain the core subcritical under cold conditions.
- PDC-27, as it relates to the CRDS being designed with appropriate margin for reliably controlling reactivity under postulated accident conditions. The ECCS does not perform core cooling by adding any fluid mass. Therefore, a poison addition safety function is not required to compensate for the addition of otherwise nonborated fluid. As discussed in Section 3.1.3, the CRDS and the CVCS, along with the boron addition system, have the combined capability to reliably control reactivity changes and maintain the core cooling capability under postulated accident conditions with appropriate margin for a stuck rod.
- GDC 29, as it relates to the CRDS, in conjunction with reactor protection systems, being designed to assure an extremely high probability of accomplishing its safety-related functions in the event of AOOs. The CRDS fulfills its safety-related functions to control the reactor within fuel and plant limits during AOOs despite a single failure of the system. The CRDS accomplishes safe shutdown (i.e., reactor shutdown via gravity-dropping of the control rod assemblies) on a reactor trip signal or loss of power. The CRDS pressure housing is an ASME Class 1 pressure boundary.

3.9.4.1 Descriptive Information of Control Rod Drive System

The CRDS is composed of a pressure-retaining housing enclosing the working mechanism, a control rod drive shaft with a coupling for attaching to the CRA hub, external electromagnetic coils with cooling loop heat exchangers, the power/control system, and the rod position indication system. It provides the rod control, reactor scram, and control rod position indication necessary for operation of the reactor module. The CRDS includes the CRDM, the control and indication cabinets and cables, and supporting SSC as described below and in Section 4.6. Information regarding the CRA and its interface with the fuel system design is in Section 4.2.

The CRDS functional testing program is discussed in Section 3.9.4.4.

3.9.4.1.1 Control Rod Drive Mechanism

The CRDM assembly is a hermetically sealed electro-mechanical device, which moves the CRA in and out of the nuclear reactor core, or may hold the CRA at elevations within the range of CRA travel. If electrical power is interrupted to the CRDM, the CRA (connected to the CRDM drive shaft) is released and inserted by gravity into the core. Figures 4.6-1 through 4.6-6 depict the CRDM assemblies mounted above the pressurizer steam space on the reactor pressure vessel (RPV). The structural materials of construction for the non-pressure boundary portions of the CRDM are discussed in Section 4.5.1. Materials for the pressure boundary portions of the CRDM are discussed in Section 5.2.3. The materials for the CRA are provided in Section 4.2.2.9. Additional characteristics of the CRDMs are provided in Section 4.1.

The reactor core is controlled using 16 CRDMs. One CRDM consists of two pressure housings (including the lower portion called latch housing, and the upper portion called rod travel housing), a latch mechanism assembly internal to the lower pressure housing operated by an outside drive coil assembly, one control rod drive shaft, a rod position indication coil assembly, and the associated wiring and water cooling connections which are described in further detail below. The rods are moved in a controlled manner to maintain control of the power level and power distribution in the core. The CRDM is connected to the CRA at the bottom end of the control rod drive shaft.

The CRDMs insert (scram) the control rod drive shaft and the attached CRA by force of gravity following a power interruption or a reactor trip. The CRDM is capable of a continuous full-height withdrawal and insertion and holding a position during normal operating conditions.

The CRDM components in contact with the primary coolant are designed to operate for a 60 year design life. The CRDM are designed to be replaceable and freely interchangeable without limitations in function and connections.

Control Rod Drive Shaft

The rod drive shaft is the link and the method of transferring force between the CRDM and the CRA. The control rod drive shaft must pass through the upper region of the reactor vessel to allow the CRDM to raise, lower, or hold the CRA. The control rod drive shaft must also interact with the rod position indication sensor coils that communicate the elevation of the control rods. The control rod drive shaft allows for the release of the CRA for refueling purposes.

Drive Coil Assembly

The drive coil assembly has four main coils: the lift coil, the movable gripper coil, the stationary gripper coil, and the remote disconnect coil. The direct current generated by the control cabinets is sent through a coil which generates a magnetic field; this magnetic field engages the flat-face plunger magnet, which moves the latch arm to engage the control rod drive shaft. The rate at which the movable gripper coil, the stationary gripper coil, and the lift coil are energized determines the speed of the control rod drive shaft. The power from the direct current electrical and alternating current distribution system to the CRDM control cabinet can be interrupted if the reactor trip breakers open, causing the control rods to be inserted via gravity. Rod movement logic tracks the speed of the control rods, which utilizes direct rod position indication. The rod movement logic has a latching function for providing extra current to the coil(s) during initial movement (startup) to ensure the latch assembly is engaged positively to the control rod drive shaft. The remote disconnect mechanism coil and latches are capable of remotely connecting and disconnecting the drive shaft from the CRA, as the drive shafts are not accessible during reactor module disassembly, as customary for the current fleet of PWRs.

Pressure Housings

The pressure housings include all components of the CRDM that form the pressure boundary for the reactor coolant. The pressure housings are ASME BPVC Section III, Subsection NB components. The pressure housings consist of the latch housing (welded to the reactor vessel head nozzle) and the rod travel housing. The rod travel housing is threaded into and seal welded to the top of the latch housing.

Latch Mechanism Assembly

The latch mechanism assembly consists of three separate latch assemblies that have the ability to grab and release the drive shaft in order to lift and lower the drive shaft in three-eighths-inch incremental steps and support operation of the remote disconnect mechanism. These motions are produced by electromagnetic forces generated by the drive coils. The latch mechanism assembly releases the control rod drive shaft during loss of power. The latch mechanism assembly is shown in Figure 4.6-5.

The latch assembly attaches to the bottom of the rod travel housing and is inserted into the latch housing.

Sensor Coil Assembly

The sensor coil assembly contains the rod position indication coils and is attached to, and supported by the rod travel housing. The sensor coil assembly is shown in Figure 4.6-4.

3.9.4.1.2 Operation of the Control Rod Drive Mechanisms

The basic CRDM mechanical and operational requirements are discussed in Section 4.6.

During operation, the CRA in each control bank are held in place by the control rod drive shafts when the drive coils are energized. When the signal is given to lift the control rod drive shafts, the CRDM drive coils are energized in the sequence to provide lifting of the control rods step-by-step starting from the rest position. Sequential rod control is necessary to control reactivity addition rates automatically and to control rod programming for the desired flux level. Rod control includes manual mode, automatic mode, and insertion-only automatic mode. Rod selection while in sequential rod control is consistent with rod programming requirements.

When a reactor trip signal occurs, the operating coils are de-energized. This causes the latch mechanism assembly magnets to drop, retracting the latches from the drive shaft grooves and allowing the drive shaft and the CRA to drop into the reactor core under gravity.

3.9.4.2 Applicable Control Rod Drive System Design Specifications

The design, fabrication, examination, testing, inspection, and documentation of the pressure boundary parts of the CRDS are in accordance with the requirements of ASME BPVC Code, Section III (Reference 3.9-1), Division I, Subsection NB. Classification of the pressure retaining portions of the CRDS is addressed in Section 3.2.2.

The pressure boundary materials are in accordance with the requirements of ASME BPVC, Section II. These pressure boundary materials are described in Section 5.2.3. The non-pressure boundary materials of the CRDS are described in Section 4.5.1.

The CRDM, which is considered part of the reactor coolant pressure boundary (RCPB), is designed in accordance with 10 CFR 50.55a. The pressure boundary components are designed to meet the stress limits and design and transient conditions specified in Table 3.9-6. The preservice and inservice inspection requirements of ASME Code, Section XI (Reference 3.9-2) are applicable to the CRDM. Welding is performed in accordance with the ASME BPVC Code, Section III, Division I, Subsection NB. The requirements to prevent brittle fracture presented in ASME BPVC Code, Section III, Division I, Subsection NB are also applicable to the CRDM. The CRDM bolting is designed in accordance with the ASME BPVC Code, Section III, as addressed in Section 3.13. Additional information on compliance with codes and code cases for the RCPB is provided in Section 5.2.1.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

The CRDM internal design and normal operating conditions are listed below:

- design pressure (RCS) - 2,100 psia
- normal operating pressure (RCS) - 1,850 psia
- design temperature (RCS) - 650 degrees Fahrenheit
- normal operating temperature (RCS) - 625 degrees Fahrenheit

The CRDMs are designed for the loading combinations and loading values specified in Section 3.9.3.

The worth of the 16 CRA in conjunction with the CRDS trip function is sufficient to overcome a stuck rod event. In addition, design requirements have been established for clearances during seismic, thermal expansion and dynamic events.

3.9.4.4 Control Rod Drive System Operability Assurance Program

The ability of the CRDS pressure housing components within the CRDMs to perform throughout the operating design life of 60 years is confirmed by the design report required by the ASME BPVC, Section III (Reference 3.9-1).

Although the NuScale CRDS is similar in design to CRDS of the currently operating fleet of PWRs, it has some unique features that include a longer control rod drive shaft (due to the presence of an integral SG and a pressurizer volume between the top of the core and the top of the RPV), and a remote disconnect mechanism. A prototype testing program was created that integrates the CRDM, the control rod drive shaft, the CRA, and the fuel assembly to demonstrate the acceptable mechanical functioning of a prototype CRDS. Rod drops under various conditions are tested and measured.

The testing of the prototype includes performance testing, stability testing, endurance testing and production testing.

The performance testing verifies the performance of the CRDS components under a broad range of conditions of temperature, pressure, and flow. The system behavior provides information for optimizing the coil activation sequence for a more reliable and accurate stepping operation. The performance tests also demonstrate the acceptability of the as -built design to meet the seismic and dynamic conditions that are expected based on the seismic and dynamic analyses.

The stability tests are conducted to demonstrate acceptable mechanical operation of the CRDM over the operation lifetime of the plant (60 years). These tests repeat the stepping sequencing motions under nominal conditions as well as rod drop testing from the full height withdrawn position.

The endurance testing involves testing the coils for the following number of operations with no appreciable damage. Actual CRDM performance may require up to one coil replacement every 60 years.

- 3.5×10^6 instances of travel (insertion or withdrawal)
- 12×10^6 total number of CRDM steps
- 2,000 operational or test scrams
- one safe shutdown during an earthquake

A series of production tests are performed on each CRDM that verifies the integrity of the pressure housing and the function of the CRDM. These tests include a hydrostatic test in accordance with the ASME BPVC Code, Section III, Division I, Subsection NB.

The as-built CRDMs are subject to pre-operational testing that verify the sequencing of the operating coils and verify the design requirements are met for insertion, withdrawal, and drop times. A description of the initial startup test program is provided in Section 14.2.

In accordance with the technical specifications, the CRDMs are subjected periodically to partial-movement checks to demonstrate the operation of the CRDM and acceptable core power distribution. In addition, drop tests of the CRA are performed at each refueling to verify the ability to meet trip time requirements.

3.9.5 Reactor Vessel Internals

The RVI assembly is comprised of several sub-assemblies which are located inside the RPV. The RVI support and align the reactor core system, which includes the control rod assemblies (CRAs), support and align the control rod drive rods, and include the guide tubes that support and house the in-core instrumentation (ICI). In addition to performing these support and alignment functions, the RVI channels the reactor coolant from the reactor core to the steam generator (SG) and back to the reactor core.

The RVI primary functions are to:

- provide structures to support, properly orient, position, and seat the fuel assemblies to maintain the fuel in an analyzed geometry to ensure core cooling capability and physics parameters are met under all modes of operational and accident conditions
- provide support and properly align the CRDS without precluding full insertion of control rods under all modes of operational and accident conditions
- provide the flow envelope to promote natural circulation of the RCS fluid with consideration given to minimizing pressure losses and bypass leakage associated with the RVI, and to the flow of coolant to the core during refueling operations

The RVI assembly is comprised of the following sub assemblies/items:

- core support assembly (CSA)
- lower riser assembly
- upper riser assembly
- flow diverter
- PZR spray nozzles

The design and construction of both the core support structures and the internal structures that comprise the RVI comply with the requirements of ASME BPVC Section III, Division 1, Subsection NG. Safety-related structures and components are constructed and tested to quality standards commensurate with the importance of the safety-related functions to be performed, and designed with appropriate margins to withstand effects of normal operation, AOOs, natural phenomena such as earthquakes, and postulated accidents including LOCA, as discussed in GDC 1, 2, 4 and 10 and 10 CFR 50.55a.

The NuScale Power Plant design complies with the relevant requirements of the following General Design Criteria of 10 CFR 50, Appendix A:

- GDC 1 and 10 CFR 50.55a, as they relate to reactor internals; the reactor internals are designed to quality standards commensurate with the importance of the safety-related functions to be performed. RVI components are Seismic Category I and designed to meet ASME BPVC Section III Division 1, Subsection NG Code requirements.

- GDC 2, as it relates to reactor internals; the reactor internals are designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety-related functions for core cooling and control rod insertion. Pursuant to GDC 2, mechanical components are designed to withstand the loads generated by natural phenomena as discussed Section 3.1.1.
- GDC 4, as it relates to reactor internals; reactor internals are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated pipe ruptures, including LOCA. Dynamic effects associated with postulated pipe ruptures such as guillotine breaks of primary piping that cause asymmetric loading effects are excluded from the design basis when analyses demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. The only RCS structures and components that require protection against the effects of pipe whipping and discharge fluids are those that are in the proximity of high and moderate energy piping between the RPV and the CNV. Additionally, the leak-before-break methodology is applied as described in Section 3.6
- GDC 10, as it relates to reactor internals; reactor internals are designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. For further details on compliance, see Section 3.1.2

3.9.5.1 Design Arrangements

Figure 3.9-1 through Figure 3.9-4 show the RVI subassemblies with components that comprise the RVI.

The overall RVI assembly is depicted in Figure 3.9-1. (Note the SG tube bundles which reside in the annulus between the upper riser assembly and the RPV upper shell are not depicted in this figure). The CSA is located near the bottom of the RPV, below the RPV flange. Above the CSA are the lower riser assembly and upper riser assembly. During disassembly, the CSA and lower riser assembly stay with the lower NPM and the upper riser assembly stays attached to the upper NPM. Each of the RVI sub-assemblies is described in more detail below.

The CSA (Figure 3.9-4) includes the core barrel, upper support blocks, lower core plate, lower fuel pins and nuts, reflector blocks, lock plate assembly, lower core support lock inserts, and the RPV surveillance specimen capsule holder and capsules.

The core barrel is a continuous ring with no welds. The upper support blocks, which are welded to the core barrel, serve to center the core barrel in the lower RPV. In addition, one of the upper support blocks engages a core barrel guide feature on the lower RPV to provide circumferential positioning of the core barrel as it is lowered into the lower RPV. The lower core plate, which is welded to the bottom of the core barrel serves to support and align the bottom end of the fuel assemblies. Locking devices align and secure the lower core plate to the core support blocks located on the RPV bottom head.

The reflector blocks contain no welds. The reflector blocks are aligned by reflector block alignment pins and stacked on the lower core plate inside the core barrel. The shape of the reflector block assembly closely conforms to the shape of the peripheral fuel assemblies and thereby constrains lateral movement of the fuel assemblies and minimizes the reactor coolant flow that bypasses the fuel assemblies.

Surveillance specimen capsule holders are welded to the outer surface of the core barrel at about the mid height of the CSA.

A flow diverter is attached to the RPV bottom head, under the CSA, as shown in Figure 3.9-1. This flow diverter smoothes the turning of the reactor coolant flow from the downward flow outside the core barrel to upward flow through the fuel assemblies. The flow diverter reduces flow turbulence and recirculation and minimizes flow related pressure loss in this region.

The lower riser assembly includes the lower riser, the upper core plate, CRA guide tubes, CRA guide tube support plate, and ICI guide tube support structure (see Figure 3.9-3). The lower riser assembly is located immediately above the CSA and is aligned with and supported on the CSA by the four upper support blocks.

The lower riser channels the reactor coolant flow leaving the reactor core upward toward the central upper riser, and separates this flow from the flow outside the lower riser which is returning from the SGs.

The upper core plate which is attached to the bottom of the lower riser by lock plate assemblies, serves to support and align the top end of the fuel assemblies. Sixteen CRA guide tubes are attached to the upper core plate and extend upward to the CRA guide tube support plate. These guide tubes house the portion of the CRAs that extend above the top of the reactor core.

An ICI guide tube support structure is located inside the lower riser to support and align ICI guide tubes with their respective fuel assemblies.

The upper riser assembly is located immediately above the lower riser assembly and extends upward to the PZR baffle plate. It channels the reactor coolant leaving the core upward through the central riser and permits the reactor coolant to turn in the space above the top of the riser and below the PZR baffle plate, and then flow downward through the annular space outside of the riser and inside of the RPV where the SG helical tube bundles are located.

The upper riser assembly includes the upper riser, a series of CRA shaft and ICI guide tube supports referred to as upper CRDS supports, and the upper riser hanger assembly. The upper riser assembly also accepts and positions the RCS injection piping. The ICI guide tubes, which are supported by the upper riser assembly, extend from their respective penetrations in the RPV top head downward through the PZR space, the upper riser, and the lower riser to their respective fuel assemblies. The portion of the ICI guide tubes extending from the RPV upper head penetrations to the bottom of the upper riser assembly is depicted in Figure 3.9-2. The upper riser assembly hangs from the pressurizer baffle plate. A small vertical clearance is provided between the upper riser and the lower riser to accommodate thermal growth in the vertical direction. In addition, there is a bellows assembly in the lower portion of the upper riser (see Figure 3.9-2) to provide added flexibility in the vertical direction to accommodate circumstances that involve sufficient thermal growth to close the vertical gap between the upper and lower riser assemblies. The RVI materials including base materials and weld filler materials are discussed in Section 4.5.2 and are designed to minimize the number of welds and bolted interfaces within the high neutron flux regions.

During refueling and maintenance outages the upper riser assembly stays attached to the upper section of the NPM (upper CNV, upper RPV and SG) while providing physical access for potential inspection of the feedwater plenums, SG, RPV and control rod drive shaft supports. The lower riser assembly and CSA remain with the lower NPM (lower CNV, lower RPV, core barrel, and core plates) when the module is parted for refueling and maintenance.

The RVI upper riser assembly is supported from the RPV integral steam plenum (e.g., below the bottom of the PZR).

Under normal operation, the reactor core is supported by the core support structures of the CSA (core support blocks, core barrel, lower core plate and upper core plate) that surround the fuel assemblies. The deadweight and other mechanical and hydraulic loads from the fuel are transferred to the upper and lower core support plates. The motion of the upper and lower core support plates is coupled through the core barrel. Under seismic and other accident conditions, the core barrel transfers lateral loads to the RPV shell through the core support blocks at the bottom of the RPV and the upper support blocks that are attached to the upper portion of the core barrel. The vertical loads are transferred from the core barrel to the RPV head through the core support blocks.

The fuel is surrounded by a heavy neutron reflector made of reflector blocks stacked on top of each other. The heavy reflector reflects neutrons back into the core to improve fuel performance. The heavy reflector provides the core envelope and directs the flow through the core. Under normal operation the heavy reflector does not provide support to the core and performs as an internal structure. During seismic and other accident events the heavy reflector limits the lateral movement of the fuel assemblies and transfers those loads to the core barrel.

A set of upper CRDM supports in the upper riser assembly, in conjunction with the CRA guide tube support plate, CRA guide tubes, and upper core plate in the lower riser assembly properly align and provide lateral support for the CRAs. The clearances provided at all these supporting members are intended to ensure adequate alignment of the CRDS with the fuel assemblies and permit full insertion of control rods under all design basis events (DBEs).

3.9.5.2 Loading Conditions

Design, construction, and testing of the RVI core support structures and internal structures are in accordance with ASME BPVC Section III, Division 1, Subsection NG.

Section 3.6.2 provides determination and evaluation of pipe rupture locations and loads, and includes dynamic effects of postulated rupture of piping. Section 3.9.1 provides acceptable analytical methods for Seismic Category I components and supports designated ASME BPVC, Section III, Division 1, Class CS, which include RVI. The plant and system operating transient conditions including postulated seismic events and DBE that provide the basis for the design of the RVI are provided in Section 3.9.3. Section 3.9.2 addresses the results of the comprehensive vibration assessment program including the preoperational vibration test program plan for the RVI that is consistent with the guidelines of RG 1.20.

3.9.5.3 Design Bases

Pursuant to GDC 10, the RVI are designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The RVI core support structures and internal structures are designed for the service loadings and load combinations shown in Table 3.9-5. The method of combining loads for ASME service level A, B, C, D, and test conditions is addressed in Section 3.9.3.

Section 3.9.3.1 describes allowable design or service loads to be applied to the RVI and the effects of service environments, deflection, cycling, and fatigue limits.

Section 3.9.2 provides the dynamic analyses of the RVI design under steady-state and operational transient conditions, and the proposed program for pre-operational and startup testing of flow-induced vibration and acoustic resonance.

Structural integrity evaluation for the structural design adequacy and ability, with no loss of safety function, of the reactor vessel internals (RVI) to withstand the loads from breaches in high energy pressure boundaries in combination with the safe shutdown earthquake is provided in Section 3.9.3.

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

This section describes the functional design, qualification provisions and inservice testing (IST) program for ASME BPVC Code, Section III Class 1, Class 2, Class 3, and non-safety-related and non-ASME valves that have been added to the IST program as having an important function and augmented quality requirements. The NuScale Power Plant standard design does not have any pumps or dynamic restraints which perform a specific function identified in the ASME OM Code (OM-2012) Subsection ISTA-1100 (Reference 3.9-3).

The provisions and programs described here verify that components in the IST program are in a state of operational readiness to perform their intended functions throughout the life of the plant.

The IST of valves is performed in accordance with the ASME OM Code, as required by 10 CFR 50.55a(f). In addition, the program also considers the guidance provided in RG 1.192 and NUREG-1482 ASME OM Code, Subsection ISTC defines the functional testing requirements for valves.

In addition to the valves that meet the criteria of ISTA-1100, valves identified by the Design Reliability Assurance Program (DRAP) as augmented quality are included in an augmented IST program, also described in this section.

The following GDC apply to this section

- GDC 1 requires, in part, that structures, systems, and components (SSC), which include pumps, valves, and dynamic restraints be designed, fabricated, erected, constructed, and inspected to quality standards commensurate with the importance of the safety functions they perform.
- GDC 2 requires, in part, that components be designed to withstand the effects of severe natural phenomena, combined with appropriate effects of normal and accident conditions, without a loss of capability to perform their safety functions
- GDC 4 requires, in part, that components be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents as described in Section 3.11. In addition, the NuScale Power Plant design applies the leak-before-break methodology to eliminate the dynamic effects of pipe rupture, as described in Section 3.6.3.
- GDC 14 requires that the reactor coolant pressure boundary (RCPB) be designed with an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.
- GDC 15 requires that the reactor coolant system (RCS) be designed with sufficient margin of safety so that the design conditions of the RCPB are not exceeded during conditions of normal operation, including AOOs.
- GDC 37 requires that the emergency core cooling system be designed to permit periodic functional testing to ensure leak-tight integrity and performance of the active components. The tests verify the operability and performance of the active components in accordance with the ASME OM Code.
- GDC 43 requires the containment atmospheric cleanup system to have functional testing to verify leak tightness. The NuScale Power Plant design does not have a containment atmospheric cleanup system.
- GDC 46 requires that the cooling water system be designed to permit periodic functional testing to ensure leak tight integrity and performance of the active components. The tests verify the operability and performance of the active components in accordance with the ASME OM Code.
- GDC 54 requires that piping systems penetrating the primary reactor containment be provided with the capability to periodically test the operability of the isolation valves and determine valve leakage acceptability. The IST program ensures the active components operability and performance are in accordance with the ASME OM Code.

COL Item 3.9-3: A COL applicant that references the NuScale Power Plant design certification will submit a Preservice Testing program for valves as required by 10 CFR 50.55a.

COL Item 3.9-3: A COL applicant that references the NuScale Power Plant design certification will establish an Inservice Testing program in accordance with ASME OM Code and 10 CFR 50.55a.

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

The NuScale Power Plant standard design does not have any safety-related pumps, dynamic restraints, or motor operated valves.

The functional design and qualification of safety-related valves is performed in accordance with ASME QME-1 (QME-1-2007), as endorsed in RG 1.100, Revision 3 with clarifications as described in Section 3.10.2. Qualification for the electrical components of valves is described in FSAR Section 3.

In accordance with 10 CFR 50.55a(f)(3), Class 1, 2 and 3 valves are designed and provided with access to enable the performance of inservice testing to assess operational readiness in accord with the ASME OM Code and as defined in the inservice testing program. Working platforms are provided in areas requiring inspection and servicing of valves.

A COL applicant that references the NuScale Power Plant design certification will incorporate all IST access requirements into the design and construction, as specified by 10 CFR 50.55a(f)(3). The quality assurance requirements for the design, fabrication, construction, and testing safety-related pumps, valves, and dynamic restraints is controlled by the plant Quality Assurance program as described in Chapter 17. The requirements are in accordance with 10 CFR 50 Appendix B.

3.9.6.2 Inservice Testing Program for Pumps

The NuScale Power Plant design does not have any pumps which perform a specific function identified in ASME OM Code Subsection ISTA-1100.

3.9.6.3 Inservice Testing Program for Valves

The NuScale Power Plant IST program applies to valves classified as ASME Code Class 1, Class 2, or Class 3 valves and non-ASME valves that meet the criteria of ISTA-1100. The IST valve program is summarized in Table 3.9-15 through Table 3.9-23. Table 3.9-15 and Table 3.9-16 include information regarding scope of the valve program, valve functions, valve categories, and test frequencies.

Valves are exercised at the frequency identified in Table 3.9-17 through Table 3.9-23 to affirm their continued availability for service. If a valve fails its surveillance test or exceeds degradation criteria, corrective actions are taken. Periodic Verification of power operated valves will be performed in accordance with the ASME OM Code and the requirements of 10CFR50.55a.

Grouping of valves for analysis or testing in accord with the ASME OM Code is done by valve type, model, and size. The population of each group is made up of valves from all installed NPM. The NuScale IST plan consists of 564 total valves (for 12 NPMs) divided into 15 valve groups. This results in 47 valves per NPM for a NuScale 12-module facility.

COL Item 3.9-4: A COL applicant that references the NuScale Power Plant design certification will identify any site-specific valves and provide inservice testing in accordance with the latest endorsed ASME Code with addenda of the ASME OM Code incorporated by reference by 10 CFR 50.55a 18 months prior to the date for initial fuel load.

3.9.6.3.1 Inservice Testing for Motor-Operated Valves

The NuScale Power Plant design does not have any motor-operated valves that perform a specific function identified in ASME OM code Subsection ISTA-1100.

3.9.6.3.2 Inservice Testing Program for Power-Operated Valves Other Than MOVs

Power-operated valves (POVs) included in the IST program and their testing requirements are summarized in Table 3.9-17 through Table 3.9-22.

Testing and assessment of active pneumatically operated valves is in accordance with ASME OM Code (Reference 3.9-3) Subsection ISTC. There are four groups of active pneumatically operated valves as follows:

- chemical and volume control system (CVCS) boron dilution isolation valves consisting of two valves per NPM (24 valves total)
- FW regulating valves consisting of two valves per NPM (24 valves total)
- backup main steam isolation valves (MSIV) consisting of two valves per NPM (24 valves total)
- backup MS isolation bypass valves consisting of two valves per NPM (24 valves total).

Testing and assessment of active hydraulic-operated valves (HOVs) is in accordance with ASME OM Code (Reference 3.9-3) Subsection ISTC. There are five groups of HOVs as follows:

- 2-inch containment isolation valves consisting of 16 valves per NPM (192 valves total)
- feedwater isolation valves, consisting of two valves per NPM (24 valves total)
- MSIVs consisting of two valves per NPM (24 valves total)
- MS isolation bypass valves consisting of two valves per NPM (24 valves total)
- DHRS actuation valves consisting of four valves per NPM (48 valves total)

3.9.6.3.3 Inservice Testing Program for Check Valves

Check valves included in the IST program and their testing requirements are summarized in Table 3.9-19. Testing and assessment of check valves is in accordance with ISTC-3522 and Check Valve Condition Monitoring (OM Mandatory Appendix II). There are two groups of check valves in the program, the feedwater (FW) isolation check valves and the FW backup isolation check valves, consisting of four valves per NPM (48 valves total).

3.9.6.3.4 Pressure Isolation Valve Leak Testing

The NuScale Power Plant design does not contain any pressure isolation valves which perform a specific function identified in ASME OM code Subsection ISTA-1100.

3.9.6.3.5 Containment Isolation Valve Leak Testing

Containment isolation valves subject to leak testing are shown in Table 3.9-19. Testing requirements for these valves are also shown in the table. ASME Class boundaries relative to containment isolation valves are shown in Figure 3.6-1.

3.9.6.3.6 Inservice Testing Program for Safety and Relief Valves

Safety and relief valves included in the IST program and their testing requirements are presented in Table 3.9-23. Pressure Relief Devices (OM Mandatory Appendix I) have four groups of valves, the reactor safety valves, the reactor vent valves (RVV), reactor recirculation valves (RRV), and steam generator system (SGS) thermal relief valves. The reactor safety valves consist of two valves per NPM (24 valves total). The RVVs are power-actuated relief valves that meet elements of Subsection ISTC of the ASME OM Code. The RVVs consist of three valves per NPM (36 valves total). These valves are tested in place each refueling outage. The RRVs meet the elements of Subsection ISTC of the ASME OM Code. The RRVs consist of two valves per NPM (24 valves total), and these valves are tested in place each refueling outage. The SGS thermal relief valves consist of two valves per NPM (24 valves total). All of these valves are grouped per NPM to meet the intent of Mandatory Appendix I. Any problems identified during testing are evaluated for generic applicability to the entire NuScale Power Plant.

3.9.6.3.7 Inservice Testing Program for Manually Operated Valves

There are no manually-operated safety-related valves in the NuScale Power Plant design which perform a specific function identified in ASME OM code Subsection ISTA-1100.

3.9.6.3.8 Inservice Testing Program for Explosively Activated Valves

The NuScale Power Plant design does not utilize explosive valves.

3.9.6.4 Inservice Testing Program for Dynamic Restraints

The NuScale Power Plant design does not utilize dynamic restraints.

3.9.6.5 Relief Requests and Alternative Authorizations to the OM Code

In the event that compliance with ASME OM Code is impractical, a relief request from the code will be submitted in accordance with 10 CFR 50.55a. The relief request will identify the applicable code requirements, describe alternative testing methods and explain why compliance is impractical. The request will provide a specific schedule for implementation of the relief request and justify the request for relief from the ASME OM Code.

No relief requests to the ASME OM Code are anticipated for the NuScale Power Plant design. For the purpose of the ISI Program, a Plant or Unit is what is defined by a "single" license issued by the governing regulatory authority. A plant or unit may consist of multiple "reactors" as long as the reactors are defined in a single license. The NuScale Power Plant consists of up to 12 NuScale Power Modules (NPMs) licensed under a single operating License. Therefore, a single IST program is used and is adjusted as each new NPM train is constructed and exposed to nuclear heat. This approach may be submitted as an Alternative to the Code upon development of the IST Program.

COL Item 3.9-5: Where the NuScale definition of Modes of Operation differ from those defined in the ASME OM Code, an Alternative may be provided to reconcile the terminology. A COL applicant that reference the NuScale Power Plant design certification will generate any relief request(s) needed as part of the Inservice Testing Program Document.

3.9.6.6 Augmented Valve Testing Program

Components not required by ASME OM Code, Subsection ISTA-1100, but with augmented quality requirements similar to ISTA-1100 are included in an augmented inservice testing program. These components were identified by the DRAP.

The DRAP process identifies functions requiring augmented quality requirements to provide greater assurance that the supporting components will perform their intended function when called upon. The DRAP considered the GDC and other select 10 CFR 50 regulations to identify safety significant functions. The augmented quality components identified as part of the DRAP process were reviewed for inservice test applicability. Those components meeting the definition of ISTA-1100 were included in the IST program. Components not meeting ISTA-1100 but having augmented requirements for the nonsafety-related functions are included in the augmented IST program. These components will be tested to the intent of the OM Code commensurate with their augmented requirements. The augmented IST plan is presented in Table 3.9-24 through Table 3.9-26 and includes valves in the following systems:

- chemical and volume control system
- condensate and feedwater system
- reactor coolant system

Testing and assessment of valves within the augmented IST program meet the intent of Subsection ISTC. The NuScale augmented IST Plan includes 96 total valves (8 valves per NPM) divided into four valve groups as follows:

- two CVCS Class 3 boundary active pneumatically operated valves per NPM (24 valves total)
- two CVCS Class 3 boundary nozzle check valves per NPM (24 valves total)

- one RCS nozzle check valve inside containment (excess flow valve installed in reverse) per NPM (12 valves total)
- three RCS excess flow check valves inside containment per NPM (36 valves total)

3.9.7 References

3.9-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition No Addenda, Section III, "Rules for Construction of Nuclear Facility Components" and applicable addenda, New York, NY.

3.9-2 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition No Addenda, Section XI, "Rules for Inservice Inspection of Nuclear Facility Components," New York, NY.

3.9-3 American Society of Mechanical Engineers, OM-2012 "Standards and Guides for Operation and Maintenance of Nuclear Power Plants," New York, NY, 2012.

3.9-4 American Society of Mechanical Engineers, QME-1-2007 Edition, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," 2007 Edition, New York, NY.

3.9-5 NuScale Power, LLC, "Comprehensive Vibration Assessment Program (CVAP) Technical Report," TR-0716-50439.

Table 3.9-1: Summary of Design Transients

Name	ASME Service Level	Events for 60 Year Design Life
Reactor heatup to hot shutdown	Level A	200
Reactor cooldown from hot shutdown	Level A	200
Power ascent from hot shutdown	Level A	700
Power descent to hot shutdown	Level A	300
Load following	Level A	19,750
Load regulation	Level A	767,100
Steady-state fluctuations	Level A	5,000,000
Load ramp increase	Level A	2000
Load ramp decrease	Level A	2000
Step load increase	Level A	3000
Step load decrease	Level A	3000
Large step load decrease	Level A	200
Refueling	Level A	60
Reactor coolant system makeup	Level A	175,200
Steam generator inventory control from hot shutdown	Level A	600
High point degasification	Level A	440
Containment evacuation	Level A	66,000
Containment flooding and drain	Level A	120
Decrease in feedwater temperature	Level B	180
Increase in secondary flow	Level B	30
Turbine trip without bypass	Level B	90
Turbine trip with bypass	Level B	180
Loss of normal AC power	Level B	60
Inadvertent main steam isolation valve (MSIV) closure	Level B	30
Inadvertent operation of the decay heat removal system (DHRS)	Level B	15
Reactor trip from full power	Level B	125
Control rod misoperation	Level B	60
Inadvertent pressurizer spray	Level B	15
Cold overpressure protection	Level B	30
CVCS malfunctions	Level B	30
Spurious emergency core cooling system valve actuation	Level C	5
Inadvertent opening of a reactor safety valve	Level C	5
CVCS Pipe Break	Level C	5
Steam generator tube failure	Level C	5
Hydrogen Detonation	Level C	1
Steam piping failures	Level D	1
Feedwater piping failures	Level D	1
Control rod assembly ejection	Level D	1
Hydrogen Detonation with DDT	Level D	1
Primary hydrostatic test	Test	10
Secondary hydrostatic test	Test	10
Containment hydrostatic test	Test	10

Table 3.9-2: Pressure, Mechanical, and Thermal Loads

Load	Description
P	Operating pressure(1)
P_{des}	Design pressure(2)
PD	Operating pressure difference(3)
PD_{des}	Design pressure difference
DW	Deadweight
B	Buoyancy
TH	Transient loads(4)
R	Steam generator tube rupture
REA	Rod ejection accident
EXT	Mechanical loads other than piping such as RPV and CNV support reactions, RVI and CNV interface loads, scram loads, fuel assembly weights, and nozzle loads
M	Piping mechanical and thermal loads
MSPB(6)	Main steam pipe break
FWPB(6)	Feedwater pipe break
DBPB(5)	Design basis pipe break other than FWPB and MSPB
RSV	Reactor safety valve actuation
ECCS	Emergency core cooling system actuation
SSE	Safe shutdown earthquake
OBE	Operating basis earthquake
L	Lifting and handling
LL	Live load
LT	Load test
TR	Transportation
H	Hydrostatic test
CILRT	CNV Appendix J Type A integrated leak rate test pressure
P_{g1}	Hydrogen detonation
P_{g2}	Hydrogen detonation with deflagration-to-detonation transition

Notes:

1. Operating pressure, "P," is the highest pressure during an applicable transient.
2. As used for ASME Code stress analysis, design pressure is specified as a gage pressure in accordance with NB-3112.1(b) giving consideration for operation of the RPV with a vacuum on the CNV or pressure testing of the CNV conservatively assuming a vacuum internal to the RPV.
3. Operating pressure difference, "DP," is the highest pressure difference during an applicable transient and may be internal or external.
4. Transient loads include transient thermal loads, as well as other transient loads, such as rapid pressure fluctuations.
5. DBPB includes CVCS pipe break and spurious valve actuation of the RVV, RRV and RSV CVCS pipe break includes DBPB for RPV high point degasification, PRZ spray, RCS discharge and RCS injection piping inside of containment.
6. FWPB and MSPB are breaks outside of the CNV. No FWPB or MSPB are considered inside of the CNV because leak before break is applied to these lines.

Table 3.9-3: Required Load Combinations for Reactor Pressure Vessel American Society of Mechanical Engineers Stress Analysis

Plant Event	Service Level	Load Combination[1]	Allowable Limit[2]
Design	Design	P_{des} + DW + B + EXT + M	Design
RPV hydrostatic test	Test	H + DW + B + EXT + M	Test
Normal operations	A	P + DW + B + EXT + M + TH	Level A
Transients	B	P + DW + B + EXT + M + TH	Level B
Transients + OBE[3]	B	P + DW + B + EXT + M + TH ± OBE	Level B
Design basis pipe break[4]	C	P + DW + B + EXT + M + DBPB	Level C
SG tube rupture[5]	C	P + DW + B + EXT + M + R	Level C
Rod ejection accident	D	P + DW + B + EXT + M + REA	Level C[6]
Main steam and feedwater pipe breaks	D	P + DW + B + EXT + M + MSPB/FWPB	Level D
SSE + DBPB/MSPB/FWPB	D	P + DW + B + EXT ± SRSS(SSE + DBPB/ MSPB/FWPB)[7]	Level D

Notes:
1. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
2. Support service limits also meet the regulatory positions of RG 1.124 and RG 1.130 as applicable.
3. Fatigue analysis is evaluated in accordance with ASME Section III Subsection NB-3200 and NG 3200 considering the effects of the PWR environment in accordance with RG 1.207 and NUREG/CR 6909.
4. Load combinations including thrust loads (RSV or ECCS blowdown) are design condition for connected nozzles.
5. Dynamic load due to SG tube failure is negligible.
6. In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.
7. Dynamic loads are combined considering the time phasing of the events in accordance with References RG 1.92 and NUREG-0484.

Table 3.9-4: Required Load Combinations for Containment Vessel American Society of Mechanical Engineers Stress Analysis

Plant Event	Service Level	Load Combination$^{(2)}$	Allowable Limit$^{(4)(5)}$
Design	Design	P_{des} + DW + B + EXT + M	Design
CNV hydrostatic test	Test	H + DW + B + EXT + M	Test
Normal operations	A	P + DW + B + EXT + M + TH	Level A
Transients$^{(1)}$	B	P + DW + B + EXT + M + TH	Level B
Transients + OBE$^{(1)}$	B	P + DW + B + EXT + M + TH ± OBE	-
Design basis pipe breaks	C	P + DW + B + EXT + M + DBPB$^{(3)}$	Level C
Hydrogen deflagration	C	P_{g1} + DW + B	Level C
SG tube rupture$^{(6)}$	C	P + DW + B + EXT + M + R	Level C
Rod ejection accident	D	P + DW + B + EXT + M + REA	Level C$^{(7)}$
Main steam and feedwater pipe breaks	D	P + DW + B + EXT + M + MSPB/FWPB$^{(4)}$	Level D
SSE + DBPB/MSPB/FWPB	D	P + DW + B + EXT + M ± SRSS(SSE + DBPB/ MSPB/FWPB)$^{(4)}$	Level D
Hydrogen DDT	D	P_{g2} + DW + B	Level D

Notes:

1. Fatigue analysis of all applicable items is evaluated in accordance with the ASME BPVC Section III considering the effects of the PWR environment in accordance with RG 1.207 and NUREG/CR-6909. OBE loading is only applicable to the fatigue analyses.
2. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
3. Dynamic loads are combined considering the time phasing of the events in accordance with RG 1.92 and NUREG- 0484.
4. For supports, service limits meet the regulatory positions of RG 1.124 and RG 1.130, as applicable.
5. Stress limits are as defined in the applicable subsection of ASME BPVC Section III for the specified level.
6. Dynamic load due to SG tube failure is negligible.
7. In accordance with NUREG-0800 SRP Section 15.4.8, Acceptance Criterion 2.

Table 3.9-5: Required Load Combinations for Reactor Vessel Internals American Society of Mechanical Engineers Stress Analysis

Plant Event	Service Level	Load Combination	Allowable Limit
Design	Design	PD_{des} + DW + B + EXT	Design
Normal operations	A	PD + DW + B + EXT + TH	Level A
Transients	B	PD + DW + B + EXT + TH	Level B
Transients + OBE(1)	B	PD + DW + B + EXT + TH ± OBE	Level B
Design basis pipe break	C	PD + DW + B + EXT + DBPB	Level C
SG tube rupture	C	PD + DW + B + EXT + R	Level C
Rod ejection accident	D	PD + DW + B + EXT + REA	Level C(2)
Main steam and feedwater pipe breaks	D	PD + DW + B + EXT + MSPB/FWPB	Level D
SSE + DBPB/MSPB/FWPB	D	PD + DW + B + EXT ± SRSS(SSE + DBPB/MSPB/FWPB)(3)	Level D

Notes:

1. Fatigue analysis of all applicable items is evaluated in accordance with the ASME BPVC Section III considering the effects of the PWR environment in accordance with RG 1.207 and NUREG/CR-6909. OBE loading is only applicable to the fatigue analyses.
2. In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.
3. Dynamic loads shall be combined considering the time phasing of the events in accordance with RG 1.92 and NUREG-0484.

Table 3.9-6: Required Load Combinations for Control Rod Drive Mechanism American Society of Mechanical Engineers Stress Analysis

Plant Event (1)	Service Level	Load Combination(2)(4)	Allowable Limit(6)
Design	Design	P_{des} + DW + EXT	Design
Hydrotest	Test	H + DW	Test
Appendix J CILRT	Test	CILRT + DW	Test
Normal operations	A	P + DW+ TH + EXT	Level A
Transients	B	P + DW + EXT + TH	Level B
Transients + OBE(3)	B	P + DW + EXT + TH ± OBE	Level B
Inadvertent RSV opening	C	P + DW + RSV	Level C
Spurious ECCS valve opening	C	P + DW + ECCS	Level C
Design basis pipe break	C	P + DW + EXT + DBPB	Level C
SG tube rupture (8)	C	P + DW + EXT + R	Level C
Rod ejection accident	D	P + DW + EXT + REA	Level C(7)
Main steam and feedwater pipe breaks	D	P + DW + EXT + MSPB/FWPB	Level D
MSPB/FWPB/DBPB + SSE	D	P + DW + EXT ± SRSS(SSE + MSPB/FWPB/ DBPB)(5)	Level D

Notes:

1. Fatigue analysis is evaluated in accordance with the ASME BPVC Section III and considering the effects of the PWR environment in accordance with RG 1.207 and NUREG/CR-6909.
2. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2. Handling, upending, lifting and transportation loads for all applicable components is evaluated in accordance with the design specification.
3. OBE loading is only applicable to the fatigue analyses.
4. P is the greatest or conservative pressure value during the applicable transient.
5. Dynamic loads are combined considering the time phasing of the events in accordance with RG 1.92 and NUREG-0484.
6. Stress limits are as defined in the applicable subsection of ASME BPVC Section III for the specified level.
7. In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.
8. Dynamic load due to SG tube failure is negligible.

Table 3.9-7: Load Combinations for Decay Heat Removal System Condenser

Plant Event	Service Level	Load Combination	Allowable Limit
Design	Design	P_{des} + DW + EXT + B	Design
Testing	Testing	P + DW + B + EXT	Testing
Normal operating (standby mode)	A	P + DW + B + EXT + TH	Level A
Transients	B	P + DW + B + EXT + TH	Level B
Transients + OBE(1)	B	P + DW + B + EXT + TH ± OBE(1)	Level B
Design basis pipe break	C	P + DW + B + EXT + DBPB	Level C
SG tube rupture(2)	C	P + DW + B + EXT + R	Level C
Rod ejection accident	D	P + DW + B + EXT + REA	Level C(3)
Main steam and feedwater pipe breaks	D	P + DW + B + EXT + MSPB/FWPB	Level D
SSE + DBPB/MSPB/FWPB	D	P + DW + B + EXT ± SRSS(SSE + DBPB/MSPB/FWPB)(4)	Level D

Notes:
1. OBE loading is only applicable to the fatigue analysis, if required.
2. This event imposes a thermal transient, but the dynamic response is negligible.
3. In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.
4. Dynamic loads are combined considering the time phasing of the events in accordance with References RG 1.92 and NUREG-0484.

Table 3.9-8: Load Combinations for NuScale Power Module Top Support Structure

Plant Event	Service Level	Load Combination	Allowable Limit
Design	Design	DW + EXT + M	Design
Test	Testing	LT	Test
Normal operating	A	DW + EXT + M + TH	Level A
Shutdown maintenance	A	DW + LL + EXT	Level A
Transient	B	DW + EXT + M + TH	Level B
Design bases pipe break	C	DW + EXT + M + DBPB(1)	Level C
Steam generator tube rupture(3)	C	DW + EXT + M + R	Level C
Rod ejection accident(3)	D	DW + EXT + M + REA	Level C(4)
Main steam and feedwater pipe break	D	DW + EXT + M + MSPB/FWPB(1)	Level D
SSE + MSPB/FWPB/DBPB	D	DW + EXT + M ± SRSS(SSE + MSPB/FWPB/DBPB)(1)(2)	Level D

Notes:

1. Dynamic loads such as jet impingement loads and pipe whip loads have been eliminated on high energy lines exposed to the NPM Top Support Structure. MS and FW piping meets break exclusion criteria in accordance with BTP 3-4 section B.A.(ii). High-energy CVCS lines use Integral Shield Restraint (ISR) or other protective devices at postulated breaks in accordance with Section 3.6.
2. Dynamic loads are combined considering the time phasing of the events in accordance with References RG 1.92 and NUREG-0484.
3. This event imposes a thermal transient, but the dynamic response is negligible.
4. In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.

Table 3.9-9: Loading Combinations for Decay Heat Removal System Actuation Valves

Plant Event	Service Level	Load Combination(1)	Allowable Limit
Design	Design	P_{des} + DW + B + EXT	Design
Testing	Test	H + DW + B + EXT	Test
Transients	A / B	P + DW + B + EXT + TH	Level B
Transients + OBE(2)	B	P + DW + B + EXT + TH ± OBE	Level B
Design basis pipe breaks	C	P + DW + B + EXT + DBPB	Level C
SG tube rupture(4)	C	P + DW + B + EXT + R	Level C
Rod ejection accident(4)	C	P + DW + B + EXT + REA	Level C
Main steam and feedwater pipe breaks	D	P + DW + B + EXT + M + MSPB/FWPB	Level D
DBPB/MSPB/FWPB + SSE		P + DW + B + EXT ± SRSS(SSE + MSPB/FWPB/DBPB)(3)	

Notes:

1. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
2. Fatigue analysis is evaluated in accordance with the applicable ASME Code. OBE is included in the fatigue analyses as required.
3. Dynamic loads are combined considering the time phasing of the events in accordance with RG 1.92 and NUREG-0484.
4. Dynamic load due to SG tube failure or rod ejection accident is negligible.

Table 3.9-10: Loads and Load Combinations for Reactor Safety Valves

Plant Event[3]	Service Level	Load Combination[1]	Allowable Limit
Design	Design	P + DW + EXT	Design
Testing	Testing	P + DW + EXT	Testing
Normal operation	A	P + DW + EXT + TH	Level A
Transients	B	P + DW + EXT + TH	Level B
Transients + OBE[2][3]	B	P + DW + EXT + TH ± OBE	Level B
Design basis pipe breaks	C	P + DW + EXT + DBPB	Level C
SG tube rupture[4]	C	P + DW + EXT + R	Level C
Main Steam and feedwater pipe breaks	D	P + DW + EXT + MSPB/FWPB	Level D
SSE + DBPB/MSPB/FWPB		P + DW + EXT ± SRSS(SSE + MSPB/FWPB/DBPB)[5]	
Rod ejection accident		P + DW + EXT + REA	Level C[6]

Notes:

1. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
2. Fatigue analysis is evaluated in accordance with the ASME BPVC Section III considering the effects of the PWR environment in accordance with RG 1.207 and NUREG/CR-6909.
3. OBE loading is only applicable to the fatigue analysis.
4. Dynamic load due to SG tube failure is negligible.
5. Dynamic loads are combined considering the time phasing of the events in accordance with RG 1.92 and NUREG 0484.
6. In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.

Table 3.9-11: Load Combinations for Emergency Core Cooling System Valves

Plant Event	Service Level	Load Combination(1)	Allowable Limit
Design	Design	P + DW + B + EXT	Design
Testing	Testing	P + DW + B + EXT	Testing
Normal operation	A	P + DW + B + EXT + TH	Level A
Transients	B	P + DW + B + EXT + TH	Level B
Transients + OBE(2)(3)	B	P + DW + B + EXT + TH ± OBE	Level B
Design basis pipe breaks	C	P + DW + B + EXT + DBPB	Level C
Hydrogen detonation	C	P_{g1} + DW + B	Level C
SG tube rupture(4)	C	P + DW + B + EXT + R	Level C
Rod ejection accident	D	P + DW + B + EXT + REA	Level C(5)
Main steam and feedwater pipe breaks	D	P + DW + B + EXT + MSPB/FWPB	Level D
SSE + DBPB/MSPB/FWPB	D	P + DW + B + EXT ± SRSS(SSE + DBPB/ MSPB/FWPB)(6)	Level D
Hydrogen DDT	D	P_{g2} + DW + B	Level D

Notes:

1. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
2. Fatigue analysis of all applicable components is evaluated in accordance with the applicable ASME Code considering the effects of the PWR environment in accordance with RG 1.207.
3. OBE is included in fatigue analyses.
4. Dynamic load due to SG tube failure is negligible.
5. In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.
6. Dynamic loads are combined considering the time phasing of the events in accordance with RG 1.92 and NUREG-0484.

Table 3.9-12: Required Loads and Load Combinations for Secondary System Containment Isolation Valves

Plant Event	Service Level	Load Combination(1)	Allowable Limit
Design	Design	P + DW + B + EXT + M	Design
Testing	Testing	P + DW + B + EXT + M	Testing
Normal operation	A	P + DW + B + EXT + M + TH	Level A
Transients	B	P + DW + B + EXT + M + TH	Level B
Transients + OBE(2)(3)	B	P + DW + B + EXT + M + TH ± OBE	Level B
Design basis pipe breaks	C	P + DW + B + EXT + M + DBPB	Level C
SG tube rupture(4)	C	P + DW + B + EXT + M + R	Level C
Main steam and feedwater pipe breaks	D	P + DW + B + EXT + M + MSPB/FWPB	Level D
Pipe Breaks + SSE	D	P + DW + B + EXT + M ± SRSS (SSE + MSPB/FWPB/ DBPB)(5)	Level D
Rod ejection accident	D	P + DW + B + EXT + M + REA	Level C(6)

Notes:

1. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
2. Fatigue analysis of all applicable components is evaluated in accordance with the applicable ASME Code.
3. OBE is included in fatigue analyses.
4. Dynamic load due to SG tube failure is negligible.
5. Dynamic loads are combined considering the time phasing of the events in accordance with RG 1.92 and NUREG-0484.
6. In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.

Table 3.9-13: Required American Society of Mechanical Engineers Code Loads and Load Combinations for Primary System Containment Isolation Valves

Plant Event	Service Level	Load Combination(1)	Allowable Limit
Design	Design	P + DW + B + EXT + M	Design
Testing	Testing	P + DW + B + EXT + M	Testing
Normal operation	A	P + DW + B + EXT + M + TH	Level A
Transients	B	P + DW + B + EXT + M + TH	Level B
Transients + OBE(2)(3)	B	P + DW + B + EXT + M + TH ± OBE	Level B
Design basis pipe breaks	C	P + DW + B + EXT + M + DBPB	Level C
Hydrogen detonation	C	P_{g1} + DW + B	Level C
SG tube rupture(4)	C	P + DW + B + EXT + M + R	Level C
Main steam and feedwater pipe breaks	D	P + DW + B + EXT + M + MSPB/FWPB	Level D
Pipe Breaks + SSE	D	P + DW + B + EXT + M ± SRSS (SSE + MSPB/FWPB/ DBPB)(5)	Level D
Hydrogen DDT	D	P_{g2} + DW + B	Level D
Rod ejection accident	D	P + DW + B + EXT + M + REA	Level C(6)

Notes:

1. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
2. Fatigue analysis of all applicable components is evaluated in accordance with the applicable ASME Code.
3. OBE is included in fatigue analyses.
4. Dynamic load due to SG tube failure is negligible.
5. Dynamic loads are combined considering the time phasing of the events in accordance with References RG1.92 and NUREG-0484.
6. In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.

Table 3.9-14: Loads and Load Combinations for Thermal Relief Valves

Plant Event(3)	Service Level	Load Combination(1)	Allowable Limit
Design	Design	P + DW + EXT + M	Design
Testing	Test	P + DW + EXT + M	Test
Normal operation	A	P + DW + EXT + M + TH	Level A
Transients	B	DW + EXT + M + TH	Level B
Transients + OBE(2)(3)	B	P + DW + B + EXT + M + TH ± OBE	Level B
SG tube rupture(5)	C	P + DW + EXT + M + R	Level C
Rod ejection accident(5)	C	P + DW + EXT + M + REA	Level C
Design basis pipe breaks	C	P + DW + EXT + M + DBPB	Level C
Main steam and feedwater pipe breaks	D	P + DW + EXT + M + MSPB/FWPB	Level D
DBPB/MSPB/FWPB ± SSE		P + DW + EXT + M ± SRSS(SSE + MSPB/FWPB/ DBPB)(4)	

Notes:
1. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
2. Fatigue analysis of all applicable components is evaluated in accordance with the applicable ASME Code.
3. OBE is included in fatigue analyses.
4. Dynamic loads are combined considering the time phasing of the events in accordance with RG 1.92 and NUREG-0484.
5. Dynamic load due to SG tube failure or rod ejection accident is negligible.

Table 3.9-15: NuScale Power Plant Inservice Testing Plan (Example Plan to be used in development of COL IST Plan)

1. General Information 1.1 Introduction Inservice testing plan, hereafter referred to as the IST plan, summarizes the test program for certain components pursuant to the requirements of the Code of Federal Regulations, 10 CFR 50.55a(f)(4). This testing plan is applicable to NuScale Power Plant Modules 1 through 12. 1.2 Code edition This example IST plan meets the requirements of the ASME OM Code 2012 as endorsed by 10 CFR 50.55a. In specifically identified instances where an alternative to the Code requirements is proposed or where it has been determined that conformance with certain Code requirements is impractical; an example request for relief from the Code requirement(s), including proposed alternatives to the requirement(s), has been prepared for future Nuclear Regulatory Commission review and approval pursuant to 10 CFR 50.55a(a)(3) or (f)(5). 1.3 Dates of test interval The preservice test and preservice test period will be defined by each individual NPM as it is placed into service. Some preservice testing may be completed in the factory prior to shipping the reactor module to the site. These tests will be described in the Preservice Testing Plan. The Preservice Testing Plan and IST program plans will be coordinated to eliminate overlap, redundancy and excessive testing. The initial 10-year examination and test interval will commence at generation of nuclear heat for the first reactor module. The initial ten year test interval may be less than ten years for reactor modules 02-12. The licensee may consider submitting a request for extension of the test interval from the NRC if there is considerable time between installation and start-up of NPMs.
2. Scope Valve IST is consistent with ASME OM, Subsection ISTC, Appendix I and Appendix II. The valves selected for inclusion in this testing plan are those active or passive ASME Class 1, Class 2, and Class 3 valves and pressure relief devices (and their actuating and position indicating systems) which are required to perform a specific function: a. in shutting down a reactor to the safe shutdown condition, or b. in maintaining the safe shutdown condition, or c. in mitigating the consequences of an accident Excluded from this testing plan are: a. valves used only for operating convenience such as vent, drain, instrument and test valves, or b. valves used only for system control, such as pressure regulating valves, or c. valves used only for system or component maintenance d. skid-mounted valves which are tested as part of the major component e. Category A and Category B safety and relief valves are excluded from the requirements of ISTC 3700 and ISTC 3500, valve testing requirements. Further, the valve actuating system test scope does not include external control and protection systems responsible for sensing plant conditions and providing signals for valve operation.

Table 3.9-15: NuScale Power Plant Inservice Testing Plan (Example Plan to be used in development of COL IST Plan) (Continued)

3. Valve testing table format

Detailed information and testing requirements for the valves included in this IST plan are summarized in tables, with a separate table for each plant system which contains valves within the scope of the plan. These systems are:

- chemical and volume control system
- condensate and feedwater system
- containment system
- decay heat removal system
- emergency core cooling system
- main steam system
- safety and relief valves

The information presented in the valve test tables includes:

a. Valve groups - Valves are grouped by system, safety significance, valve type, actuator type, manufacturer, model number. Each group has a unique group number to facilitate the implementation of the inservice test program.

b. Valve identification - Valve identification includes the valve number field and a brief description of the valve safety function. In each table, the valves are arranged in numerical order by the four digit location number which forms the root of each valve number.
All valves listed in this IST plan are common for each nuclear power module (NPM). Valve numbering, except for the prefix, is the same for all NPMs. The prefix for all valves will start with "01" through "12," depending on the NPM.

c. Risk ranking - A valve will either be ranked as high- or low-safety significant. This was determined by expert panel review and included the NuScale Probability Risk Assessment Group.

d. Size - The size field indicates the nominal valve size in inches.

e. Code Class - The code class field indicates the ASME Boiler and Pressure Vessel Code, Section III classification.

f. Category - The category field indicates the classification of the valve according to characteristics described in ASME OM Code, 2012 Edition, Subsection ISTC-1300. See Valve Table ISTC-3500-1 for a listing of valve categories and their meanings.

g. Function - The function field indicates the manner in which a valve accomplishes its required safety function(s). "A" denotes an active valve and "P" denotes a passive valve with the terms defined as follows:
Active valves - Valves which are required to change obturator position to accomplish their required safety function(s).
Passive valves - Valves which maintain obturator position and are not required to change obturator position to accomplish their required safety function(s).
Obturator - Valve closure member (disk, gate, plug, ball, etc.).

h. Safety function position - The safety function position field indicates the position (open or closed) to which a valve must move or remain in to accomplish its required safety function(s). The open and closed positions are indicated by "O" and "C," respectively.

Table 3.9-15: NuScale Power Plant Inservice Testing Plan (Example Plan to be used in development of COL IST Plan) (Continued)

i.	Test Parameters/Schedule - The test parameters/schedule field denotes the ASME OM code test requirements and test frequencies for valves in the IST plan. The test parameters include leak test, exercise test, fail-safe test, and position verification test. Not all test parameters are applicable to all valves. Rather, the parameters to be tested for any valve are dependent on the valve and actuator type, category, and function. Valves that have both an open and a closed safety function position, and for which the test requirements or frequencies are different in the two positions, have their open and closed test requirements identified separately. Test parameters that are not applicable to a particular valve are indicated "N/A." Check valves are exercise tested in both the open and closed direction regardless of safety function position or valve safety significance. Nonsafety-function exercise tests for high-safety-significant check valves shall be performed at least once every two years. In cases where the performance of a valve full-stroke exercise test is limited to transitions or refueling outages, a table footnote is provided which justifies this determination. See the Valve Table Index at the end of this section for a listing of test parameters and schedule acronyms and their meanings.
j.	Footnotes - Footnotes containing additional valve testing information are located at the back of each system valve table and are referenced in the tables by the footnote number in parentheses.

Table 3.9-16: Example Inservice Testing Valve Table Index

VALVE FUNCTIONS	
A - Active	
P - Passive	
SAFETY FUNCTION POSITIONS	
O - Open	
C - Closed	

VALVE CATEGORIES

Category A -	Valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required safety function(s).
Category B -	Valves for which seat leakage in the closed position is inconsequential for fulfillment of their required safety function(s).
Category C -	Valves which are self-actuating in response to some system characteristic, such as pressure (relief valves) or flow direction (check valves), for fulfillment of their required safety function(s).
Category D -	Valves which are actuated by an energy source capable of only one operation, such as rupture disks or explosively actuated valves, for fulfillment of their required safety function(s). (There are no Category D valves or pressure relief devices in the NuScale IST plan).
NOTE:	Seat tightness determination is performed as part of the performance test for Category C pressure relief devices (SRV) and may be performed as a method of close exercise test for check valves (CV). However, pressure relief devices and check valves are further classified as Category A only if there is a safety analysis criteria existing for valve seat leakage such as for pressure relief devices or check valves performing containment isolation functions or reactor coolant system pressure isolation functions.

Table 3.9-16: Example Inservice Testing Valve Table Index (Continued)

Leak Test	
LT -	Leak test Category A valve (other than containment isolation valves) per the requirements of ISTC-3630.
LTJ -	Leak test Category A containment isolation valve per the requirements of ISTC 3620.
Exercise Test	
MT -	Exercise power operated Category A or B valve full-stroke to its safety function position(s) and measure stroke time per the requirements of ISTC-3510.
ET -	Exercise Category A or B valve full-stroke to its safety function position(s) per the requirements of ISTC-3510.
CV -	Exercise Category C check valve full-stroke to its safety function position(s) per the requirements of ISTC-3510.
CVD -	Disassemble Category C check valve to verify operability per the requirements of ISTC-5220.
SRV -	Performance test Category C safety or relief valve per the requirements of ISTC-5230 and Appendix I.
DD -	Dual direction test/verification shall be performed in accordance with ISTC-5220. Frequency is determined by Appendix II.
Fail Safe Test	
FO -	Fail safe test Category A or B valve in the open direction per the requirements of ISTC-3560.
FC -	Fail safe test Category A or B valve in the closed direction per the requirements of ISTC-3560.
Position Verification Test	
PIT -	Test Category A, B, C or D valve position verification per ISTC-3700.
	Test Frequency
3MO -	Perform exercise test (and fail safe test, if applicable) nominally every three months.
CS -	Perform exercise test (and fail safe test, if applicable) during each transition. Such exercise is not required if the time period since the previous full-stroke exercise is less than three months. Valve exercising during transition shall commence within 48 hours of achieving transition, and continue until all testing is complete or the plant is ready to return to power. For extended outages, testing need not be commenced in 48 hours provided all valves required to be tested during transition will be tested prior to plant startup.
RF -	Perform exercise test (and fail safe test, if applicable) during each refueling outage.
TS -	Perform test at the applicable technical specification frequency.
24MO -	Perform test at least once every refueling outage.
NYR -	Perform test at least once every N years. For position verification tests (PIT), N equals two years. For pressure relief device performance tests (SRV), N nominally equals five years or ten years for Class 1 or Class 2 and Class 3 devices, respectively. However, other test frequencies may apply for pressure relief devices. See ASME OM Code 2012 Edition, Appendix I. (The only pressure relief devices in the NuScale IST plan are Class 1).

Table 3.9-17: Chemical and Volume Control System Valves in the Example Inservice Testing Program

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
GROUP 1: Ball valve / air operated											
CVC-AOV-0101	Low	2	3	B	A	C	N/A	MT/3MO	FC/3MO	PIT/2YR	Boron dilution prevention
CVC-AOV-0119	Low	2	3	B	A	C	N/A	MT/3MO	FC/3MO	PIT/2YR	Boron dilution prevention

Table 3.9-18: Condensate and Feedwater System Valves in the Example Inservice Testing Program

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
GROUP 2: Flow control valve / air operated											
FW-FCV-1006	LOW	6	NC (1)	A	A	C	TS	MT/24MO (2)	FC/24MO (3)	PIT/ 2YR	Feedwater isolation, backup containment isolation, backup DHRS boundary
FW-FCV-2006	LOW	6	NC (1)	A	A	C	TS	MT/24MO (2)	FC/24MO (3)	PIT/ 2YR	Feedwater isolation, backup containment isolation, backup DHRS boundary
GROUP 3: Nozzle check valve											
FW-CKV-1007	LOW	6	NC (1)	C	A	C	N/A	CV/24MO (4) DD/24MO (5)	N/A	N/A	Feedwater isolation, backup DHRS boundary
FW-CKV-2007	LOW	6	NC (1)	C	A	C	N/A	CV/24MO (4) DD/24MO (5)	N/A	N/A	Feedwater isolation, backup DHRS boundary

Notes:

1. These valves are non-ASME Code Class (RG 1.26 Quality Group D) and provide a defense-in-depth function to single isolation valves.
2. FW-FCV-1006/2006, feedwater regulating valves are full-stroke exercised under transition conditions. These valves cannot be full-stroke exercised during plant operation because closing the valves interrupts feedwater flow, resulting in severe steam generator level transients and may initiate a turbine or reactor module trip.
3. ISTC-3560 Fail-safe valves - Valves with fail-safe actuators shall be tested in accordance with the exercising frequency in ISTC-3510.
4. FW-CKV-1007/2007, Feedwater check valves are full-stroke exercised under transition conditions. These valves cannot be full-stroke exercised during plant operation because closing the valves to perform the test will interrupt feedwater flow with a potential for severe steam generator level transients and a potential turbine and reactor trip.
5. ISTC-3550 Valve in regular use - this valve operates in the course of reactor module operation at a frequency that satisfies the requirements of the IST plan.

Table 3.9-19: Containment System Valves in the Example Inservice Testing Program

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
GROUP 4: Ball valve / hydraulic operated to open / nitrogen gas to close											
CVC-ISV-0323	HIGH	2	1	A	A	C	LT LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Reactor coolant pressure boundary, containment isolation
CVC-ISV-0325	HIGH	2	1	A	A	C	LT LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Reactor coolant pressure boundary, containment isolation
CVC-ISV-0329	HIGH	2	1	A	A	C	LT LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Reactor coolant pressure boundary, containment isolation
CVC-ISV-0331	HIGH	2	1	A	A	C	LT LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Reactor coolant pressure boundary, containment isolation
CVC-ISV-0334	HIGH	2	1	A	A	C	LT LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Reactor coolant pressure boundary, containment isolation
CVC-ISV-0336	HIGH	2	1	A	A	C	LT LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Reactor coolant pressure boundary, containment isolation
CVC-ISV-0401	HIGH	2	1	A	A	C	LT LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Reactor coolant pressure boundary, containment isolation
CVC-ISV-0403	HIGH	2	1	A	A	C	LT LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Reactor coolant pressure boundary, containment isolation
CE-ISV-0101	LOW	2	2	A	A	C	LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Containment isolation
CE-ISV-0102	LOW	2	2	A	A	C	LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Containment isolation
CFD-ISV-0129	LOW	2	2	A	A	C	LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Containment isolation
CFD-ISV-0130	LOW	2	2	A	A	C	LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Containment isolation

Table 3.9-19: Containment System Valves in the Example Inservice Testing Program (Continued)

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
RCCW-ISV-0184	LOW	2	2	A	A	C	LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Containment isolation
RCCW-ISV-0185	LOW	2	2	A	A	C	LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Containment isolation
RCCW-ISV-0190	LOW	2	2	A	A	C	LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Containment isolation
RCCW-ISV-0191	LOW	2	2	A	A	C	LTJ	MT/24MO (1)	FC/24MO (6)	PIT/ 2YR	Containment isolation
GROUP 5: Ball valve / hydraulic operated to open / nitrogen gas to close											
FW-ISV-1003	LOW	4	2	A	A	C	LT	MT/24MO (2)	FC/24MO (6)	PIT/ 2YR	Feedwater isolation, containment isolation, DHRS boundary
FW-ISV-2003	LOW	4	2	A	A	C	LT	MT/24MO (2)	FC/24MO (6)	PIT/ 2YR	Feedwater isolation, containment isolation, DHRS boundary
GROUP 6: Nozzle check valve											
FW-CKV-1002	LOW	4	2	B/C	A	C	LT	CV/24MO (3) DD/24MO (5)	N/A	N/A	Feedwater isolation, DHRS boundary
FW-CKV-2002	LOW	4	2	B/C	A	C	LT	CV/24MO (3) DD/24MO (5)	N/A	N/A	Feedwater isolation, DHRS boundary
GROUP 7: Ball valve / hydraulic operated to open / nitrogen gas to close											
MS-ISV-1005	LOW	12	2	A	A	C	LT	MT/24MO (4)	FC/24MO (6)	PIT/ 2YR	Steam line isolation, containment isolation, DHRS boundary
MS-ISV-2005	LOW	12	2	A	A	C	LT	MT/24MO (4)	FC/24MO (6)	PIT/ 2YR	Steam line isolation, containment isolation, DHRS boundary

Table 3.9-19: Containment System Valves in the Example Inservice Testing Program (Continued)

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
GROUP 8: Ball valve / hydraulic operated to open / nitrogen gas to close											
MS-ISV-1006	LOW	2	2	A	P	C	LT	N/A	N/A	PIT/ 2YR	Steam line isolation, containment isolation, DHRS boundary
MS-ISV-2006	LOW	2	2	A	P	C	LT	N/A	N/A	PIT/ 2YR	Steam line isolation, containment isolation, DHRS boundary

Notes:
1. Containment isolation valves (CVC-ISV-0323, 0325, 0329, 0334, 0336, 0401, 0403, CE-ISV-0101, 0102, CFD-ISV-0129, 0130, RCCW-ISV-0184, 0185, 0190, 0191) are full-stroke exercised at transition. These valves cannot be full-stroke exercised during plant operation because closing a containment isolation valve causes a reactor module trip.
2. FW-ISV-1003/2003, Feedwater isolation valves are full-stroke exercised at transition. These valves cannot be full-stroke exercised during plant operation because closing the valves interrupts feedwater flow resulting in severe steam generator level transients and potentially initiating a turbine and reactor trip.
3. FW-CKV-1002/2002, Feedwater isolation check valves are full-stroke exercised at transition. These valves cannot be full-stroke exercised during plant operation because closing the valves interrupts feedwater flow, resulting in severe steam generator level transients and potentially initiating a turbine and reactor trip.
4. MS-ISV-1005/2005, Main steam isolation valves (MSIV) are full-stroke exercised at transition. These valves cannot be full-stroke exercised during plant operation because closing a MSIV causes steam generator pressure and level transients and potentially initiating a turbine and reactor trip.
5. ISTC-3550 Valve in regular use - This valve operates in the course of reactor module operation at a frequency that satisfies the requirements of this IST plan.
6. ISTC-3560 Fail-safe valves - Valves with fail-safe actuators shall be tested in accordance with the exercising frequency in ISTC-3510.

Table 3.9-20: Decay Heat Removal System Valves in the Example Inservice Testing Program

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
GROUP 9: Ball valve / hydraulic operated to open / nitrogen gas to close											
DHR-HOV-1002A	LOW	6	2	B	A	O	N/A	MT/24MO (1) SRV/5YR	FC/24MO (2)	PIT/ 2YR	Decay heat removal
DHR-HOV-1002B	LOW	6	2	B	A	O	N/A	MT/24MO (1) SRV/5YR	FC/24MO (2)	PIT/ 2YR	Decay heat removal
DHR-HOV-2002A	LOW	6	2	B	A	O	N/A	MT/24MO (1) SRV/5YR	FC/24MO (2)	PIT/ 2YR	Decay heat removal
DHR-HOV-2002B	LOW	6	2	B	A	O	N/A	MT/24MO (1) SRV/5YR	FC/24MO (2)	PIT/ 2YR	Decay heat removal

Notes:
1. DHR-HOV-1002A/B and DHR-HOV-2002A/B, Decay heat removal actuation valves are full-stroke exercised at transition. These valves cannot be full-stroke exercised during plant operation because such testing would unnecessarily subject the steam generator nozzles to thermal transients from decay heat condenser condensate flow.
2. ISTC-3560 Fail-safe valves - Valves with fail-safe actuators shall be tested in accordance with the exercising frequency in ISTC-3510.

Table 3.9-21: Emergency Core Cooling System Valves in the Example Inservice Testing Program

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
GROUP 10: Globe valve / hydraulic operated / remote actuated											
ECC-HOV-0101A	HIGH	6	1	B/C	A	O/C	TS (1)	MT/24MO (2) SRV/5YR (3)	FC/24MO (4)	PIT/ 2YR (5)	Core cooling recirculation path, reactor coolant pressure boundary
ECC-HOV-0101B	HIGH	6	1	B/C	A	O/C	TS (1)	MT/24MO (2) SRV/5YR (3)	FC/24MO (4)	PIT/ 2YR (5)	Core cooling recirculation path, reactor coolant pressure boundary
ECC-HOV-0101C	HIGH	6	1	B/C	A	O/C	TS (1)	MT/24MO (2) SRV/5YR (3)	FC/24MO (4)	PIT/ 2YR (5)	Core cooling recirculation path, reactor coolant pressure boundary
GROUP 11: Globe valve / hydraulic operated / remote actuated											
ECC-HOV-0104A	HIGH	4	1	B/C	A	O/C	TS (1)	MT/24MO (2) SRV/5YR (3)	FC/24MO (4)	PIT/ 2YR (5)	Core cooling recirculation path, reactor coolant pressure boundary
ECC-HOV-0104B	HIGH	4	1	B/C	A	O/C	TS (1)	MT/24MO (2) SRV/5YR (3)	FC/24MO (4)	PIT/ 2YR (5)	Core cooling recirculation path, reactor coolant pressure boundary

Table 3.9-21: Emergency Core Cooling System Valves in the Example Inservice Testing Program (Continued)

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	

Notes:

1. The Reactor vent valves and reactor recirculation valves do not have specific leakage criteria. The associated pilot valve bodies form part of the reactor coolant and containment boundaries and are subject to technical specification leakage requirements and Appendix J Type B testing, respectively.

REACTOR VENT VALVE	TRIP VALVE	RESET VALVE
ECC-HOV-0101A	ECC-SV-102A	ECC-SV-103A
ECC-HOV-0101B	ECC-SV-102B	ECC-SV-103B
ECC-HOV-0101C	ECC-SV-102C ECC-SV-107	ECC-SV-103C

REACTOR RECIRCULATION VALVE	TRIP VALVE	RESET VALVE
ECC-HOV-0104A	ECC-SV-105A	ECC-SV-106A
ECC-HOV-0104B	ECC-SV-105B	ECC-SV-106B

2. ISTC-3510 Exercising Test Frequency - Reactor vent valves and reactor recirculation valves shall be tested once per fuel cycle. The associated pilot valves are considered part of the main valve. ECC-HOV-0101A/B/C, reactor vent valves and ECC-HOV 0104A/B, reactor recirculation valves are full-stroke exercised at refueling in the open direction only. These valves cannot be full-stroke or partial-stroke exercised during plant operation because cycling the valves opens an RCS vent path. These valves open and remain open when actuated. The closed safety function is passive.
3. SRV testing includes a functional test of the inadvertent actuation block valve at normal RCS pressure. Testing also includes an augmented functional test to demonstrate that the valves not SRV tested will open on low RCS pressure even though the trip valves remain energized (closed).
4. ISTC-3560 Fail-safe valves - Valves with fail-safe actuators shall be tested in accordance with the exercising frequency in ISTC-3510. The fail-safe test also tests the associated trip valve.
5. ISTC-3700 Position verification testing - Reactor vent valves, reactor recirculation valves and all associated trip and reset valves have remote position indicators and shall be tested for position verification.

Table 3.9-22: Main Steam System Valves in the Example Inservice Testing Program

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
GROUP 12: Gate valve / air operated											
MS-AOV-1003	LOW	12	NC (1)	A	A	C	TS	MT/24MO (2)	FC/24MO (3)	PIT/ 2YR	Steam line isolation, backup containment isolation, backup DHRS boundary
MS-AOV-2003	LOW	12	NC (1)	A	A	C	TS	MT/24MO (2)	FC/24MO (3)	PIT/ 2YR	Steam line isolation, backup containment isolation, backup DHRS boundary
GROUP 13: Gate valve / air operated											
MS-AOV-1004	LOW	4	NC (1)	A	P	C	TS	N/A	N/A	PIT/ 2YR	Steam line isolation, backup containment isolation, backup DHRS boundary
MS-AOV-2004	LOW	4	NC (1)	A	P	C	TS	N/A	N/A	PIT/ 2YR	Steam line isolation, backup containment isolation, backup DHRS boundary

Notes:
1. These valves are non-ASME Code Class (RG 1.26 Quality Group D) and provide a defense-in-depth function to single containment isolation valves.
2. MS-AOV-1003/2003, Secondary main steam isolation valves are full-stroke exercised at transition. These valves cannot be full-stroke exercised during plant operation because closing a secondary MSIV causes steam generator pressure and level transients and, most likely, a turbine and reactor trip.
3. ISTC-3560 Fail-safe valves - Valves with fail-safe actuators shall be tested in accordance with the exercising frequency in ISTC-3510.

Table 3.9-23: Safety and Relief Valves in the Example Inservice Testing Program

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
GROUP 14: Safety valve / pilot operated / self-actuating											
RCS-PSV-0003A	HIGH	4	1	C	A	O/C	N/A	SRV/5YR	N/A	SRV/5YR	Overpressure protection, reactor coolant pressure boundary
RCS-PSV-0003B	HIGH	4	1	C	A	O/C	N/A	SRV/5YR	N/A	SRV/5YR	Overpressure protection, reactor coolant pressure boundary
Group 15: Relief Valve / Self-actuating											
SGS-PSV-1002	LOW	1 1/2	2	C	A	O/C (1)	N/A	SRV/10YR	N/A	N/A	Thermal Overpressure Protection, SGS Pressure
SGS-PSV-2002	LOW	1 1/2	2	C	A	O/C (1)	N/A	SRV/10YR	N/A	N/A	Thermal Overpressure Protection, SGS Pressure Boundary

NOTES

1. SGS-PSV-1002/2002 provide thermal overpressure protection to the SGS and DHRS when both of these systems are inoperable only, and all control rods are on the bottom. Unintended containment isolation during flushing evolutions may result in overpressure conditions caused by changes in fluid temperature. These thermal relief valves provide SGS and DHRS protection. These devices are not credited for overpressure protection when the SGS or DHRS are operable and are not credited for protecting any system during a design basis event. Overpressure protection during SGS and DHRS operability is provided by system design pressure and the reactor safety valves. These valves are included here due to the significance of the systems protected.

Table 3.9-24: Example - NuScale Power Plant Augmented Inservice Testing Plan

1. General Information

1.1 Introduction

The example augmented inservice testing plan, hereafter referred to as the augmented IST plan, has been prepared to summarize the test program for certain components identified as part of NuScale's DRAP. This augmented testing plan is applicable to NuScale Power Plant modules 1 through 12.

1.2 Code Edition

This example augmented IST plan meets specific aspects of the ASME OM Code 2012 as endorsed by 10 CFR 50.55a, based on the augmented quality requirements of the component as defined by NuScale's DRAP.

1.3 Dates of Test Interval

The preservice test and preservice test period will be defined by each individual reactor module as it is placed into service. Preservice testing is performed to verify the augmented quality requirement.

The 10-year examination and test interval for the augmented IST plan will follow that of the IST plan.

2. Scope

The scope of the augmented IST plan is derived from the requirements for important functions that are met by components having augmented quality requirements that shall meet specific aspects of the ASME OM Code 2012 Edition as endorsed by 10 CFR 50.55a(f)(4).

Valve augmented IST is consistent with ASME OM, Subsection ISTC, Appendix I and Appendix II as modified by the expert panel review. The valves selected for inclusion in this augmented testing plan are those active or passive valves and pressure relief devices (and their actuating and position indicating systems) that are required to perform a specific augmented function during a design basis or beyond-design-basis event:

a. in shutting down a reactor to the safe shutdown condition, or
b. in maintaining the safe shutdown condition, or
c. in mitigating the consequences of an accident

Excluded from this testing plan are

a. valves used only for operating convenience such as vent, drain, instrument and test valves
b. valves used only for system control, such as pressure regulating valves
c. valves used only for system or component maintenance
d. skid-mounted valves that are tested as part of the major component
e. Category A and Category B Safety and relief valves, which are excluded from the requirements of ISTC 3700 and ISTC 3500 valve testing requirements.

Further, the valve actuating system test scope does not include external control and protection systems responsible for sensing plant conditions and providing signals for valve operation.

The nonsafety-related valves that have augmented quality requirements in the scope of this testing plan were identified by expert panel review.

3. Valve Testing Table Format

Detailed information and testing requirements for the valves included in the augmented IST plan are summarized in tables. A separate table has been prepared for each plant system which contains valves within the scope of the plan. These systems are:

- the chemical and volume control system
- the condensate and feedwater system
- the reactor coolant system

4. Abbreviations

Abbreviations used in the following tables are defined in Table 3.9-16.

Table 3.9-25: Example Augmented Inservice Testing Valve Program - Chemical and Volume Control System

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
GROUP 1: Ball valve / air-operated											
CVC-AOV-0339	LOW	2	3	A	A	C	LTJ (1)	MT/3MO	FC/3MO	PIT/ 2YR	Containment isolation
GROUP 2: Globe valve / solenoid operated											
CVC-AOV-0406	LOW	1/2	3	A	A	C	LTJ (1)	MT/3MO	FC/3MO	PIT/ 2YR	Containment isolation
GROUP 3: Nozzle check valve											
CVC-CKV-0352	LOW	2 1/2	3	A/C	A	C	LTJ (1)	CV/24MO DD/24MO (2)	N/A	N/A	Containment isolation
CVC-CKV-0353	LOW	2	3	A/C	A	C	LTJ (1)	CV/24MO DD/24MO (2)	N/A	N/A	Containment isolation

Notes:
1. These valves provide a defense-in-depth function to containment isolation valves in accordance with Regulatory Guide 1.26 for a Quality Group C/D class break. These valves have specific leakage criteria similar to containment isolation valves, but are not considered part of the containment isolation boundary.
2. ISTC-3550 Valve in regular use - This valve operates in the course of reactor module operation at a frequency that satisfies the requirements of the IST plan.

Table 3.9-26: Example Augmented Inservice Testing Valve Program - Reactor Coolant System

Valve Number	Risk Ranking	Size	Code Class	Category	Function	Safety Function Position	Test Parameters and Schedule				Remarks
							Leak Test	Exercise Test	Fail Safe Test	Position Verification Test	
GROUP 4: Nozzle check valve											
RCS-CKV-0332	LOW	2	1	C	A	C	N/A	CV/8YR (1)	N/A	N/A	Reactor coolant pressure boundary, containment isolation (beyond-design-basis)
GROUP 5: Excess flow nozzle check valve											
RCS-CKV-0323	LOW	2	1	C	A	C	N/A	CVD/8YR (1)	N/A	N/A	Reactor coolant pressure boundary, containment isolation (beyond-design-basis)
RCS-CKV-0333	LOW	2	1	C	A	C	N/A	CVD/8YR (1)	N/A	N/A	Reactor coolant pressure boundary, containment isolation (beyond-design-basis)
RCS-CKV-0400	LOW	2	1	C	A	C	N/A	CVD/8YR (1)	N/A	N/A	Reactor coolant pressure boundary, containment isolation (beyond-design-basis)

Notes:

1. These valves are tested once every eight years on a staggered test basis. The excess flow closure function cannot be performed. Flow required is in excess of system capability as it is required to simulate pressurized pipe rupture outside containment with failure of both containment isolation valves; therefore, a valve disassembly examination is performed pursuant to ISTC-5221(c)(4).

Figure 3.9-1: Reactor Module Showing Reactor Vessel Internals Component Assemblies

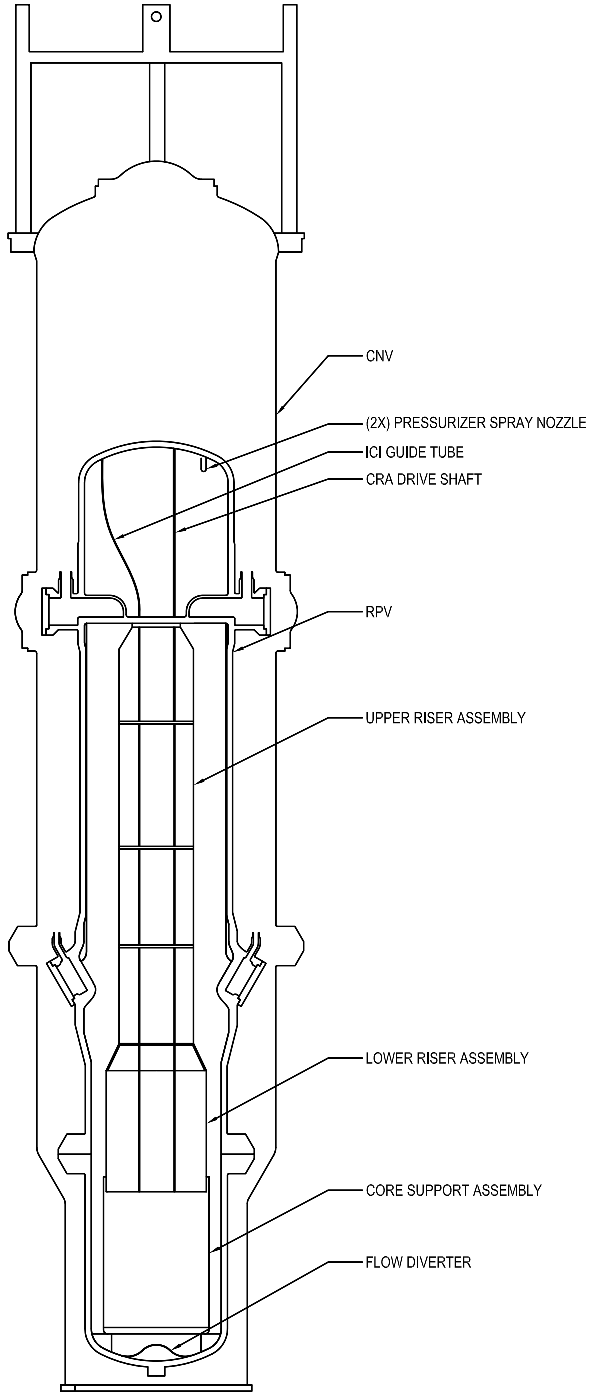


Figure 3.9-2: Upper Riser Assembly

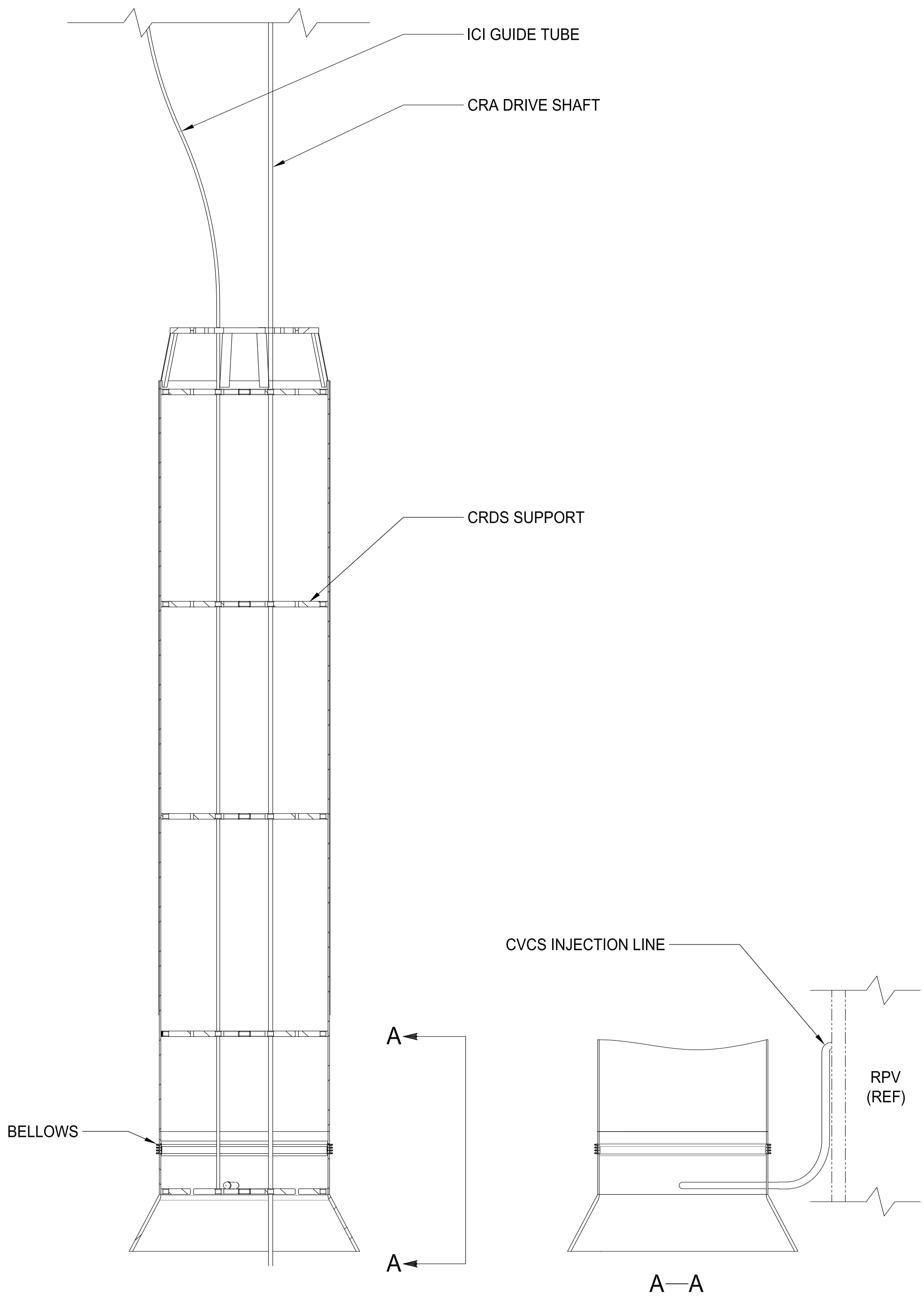


Figure 3.9-3: Lower Riser Assembly

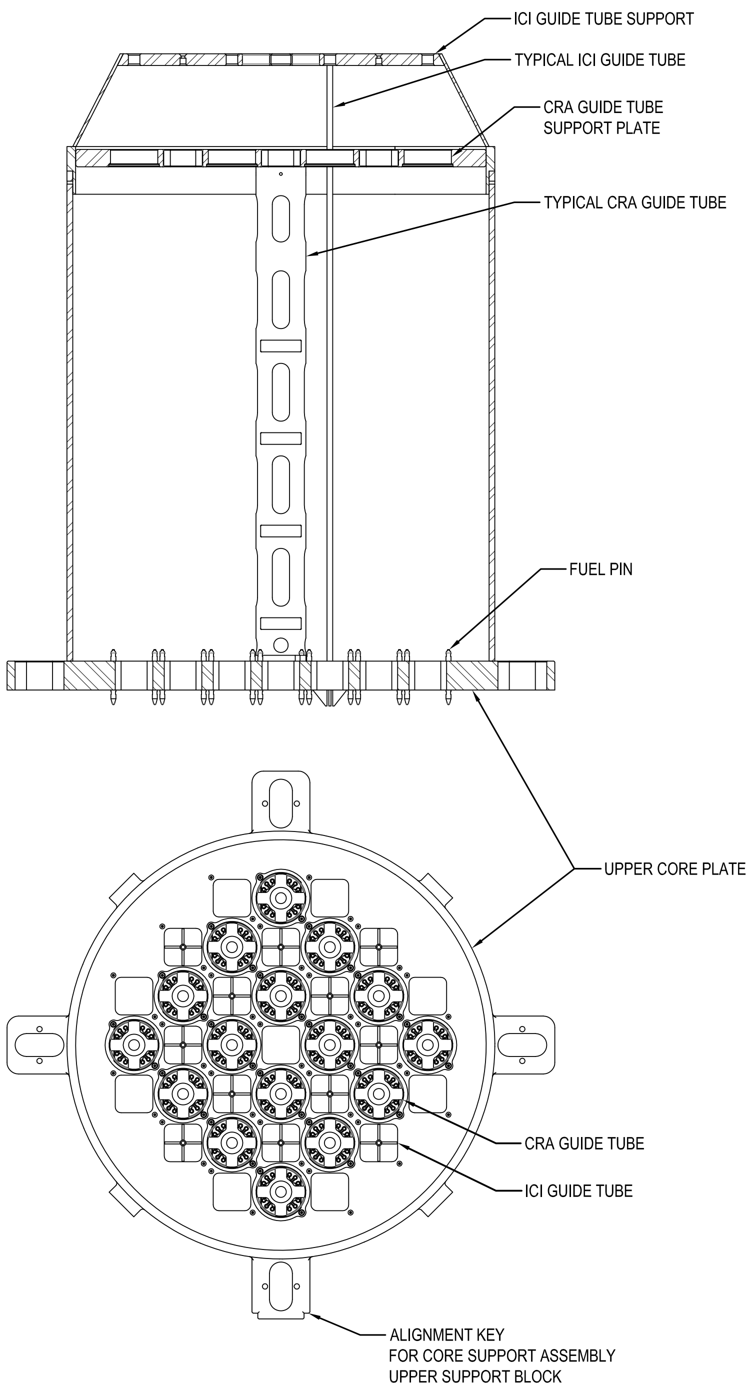
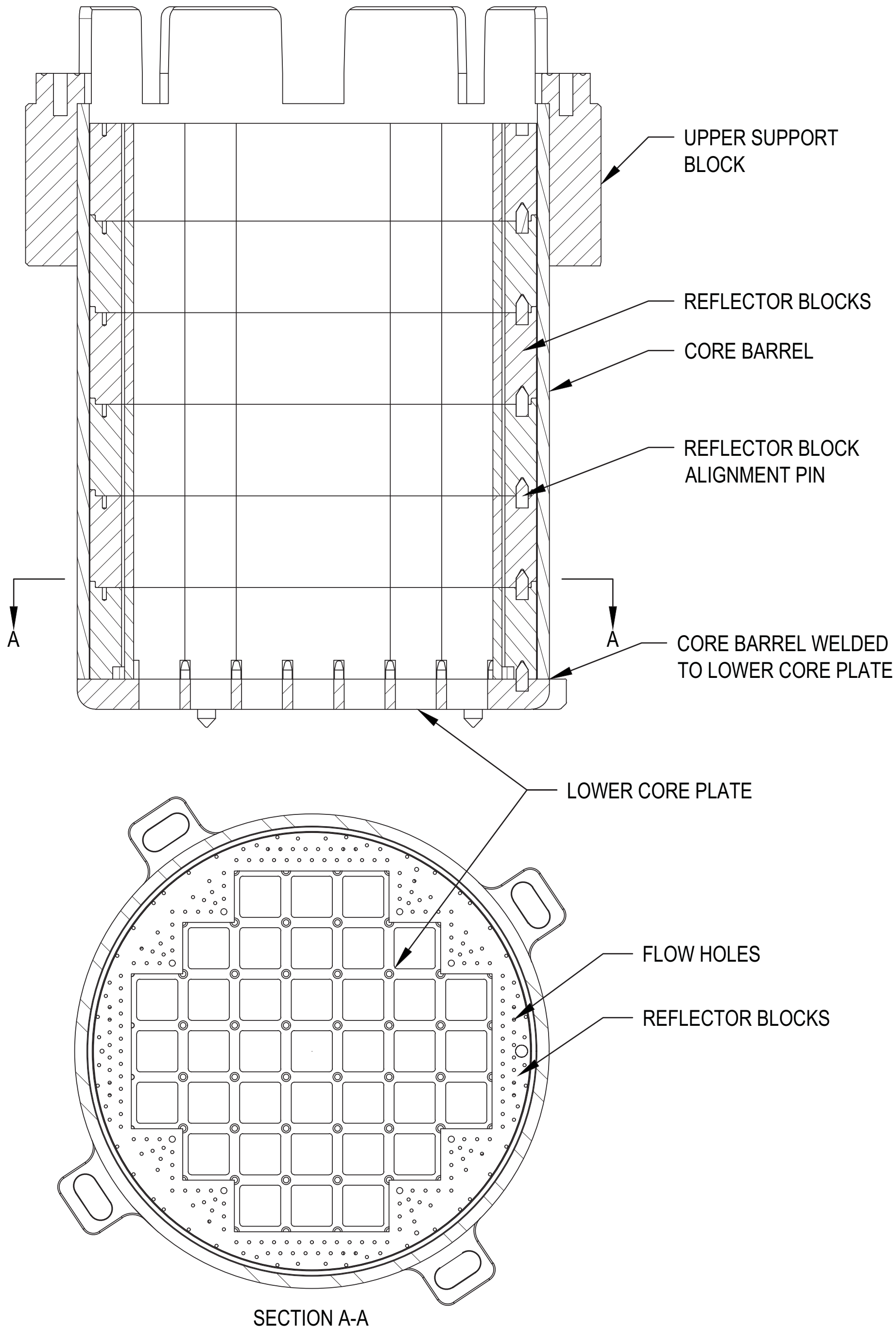


Figure 3.9-4: Core Support Assembly



3.10 Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment

Electrical and mechanical equipment including instrumentation (with exception of piping) and their associated supports classified as Seismic Category I, are demonstrated through qualification to withstand the full range of normal and accident loadings. The equipment to be seismically and dynamically qualified includes the following:

- electrical equipment, including instrumentation and some post-accident monitoring equipment
- active, safety-related mechanical equipment, such as control rod drive mechanisms and some valves, that perform a mechanical motion to accomplish their safety function and other nonactive mechanical components, the structural integrity of which is maintained to perform their safety function

Seismic Category II structures, systems, and components are designed so that the safe shutdown earthquake (SSE) does not cause unacceptable structural failure of or interaction with Seismic Category I items.

The equipment to be qualified includes equipment necessary for safe shutdown, emergency core cooling, containment heat removal, containment isolation, or for mitigating the consequences of accidents or preventing a significant release of radioactive material to the environment. Also included is equipment in the reactor protection system, the engineered safety features, and highly reliable electrical equipment.

The structures, systems, and components qualified as Seismic Category I or Category II are listed in Table 3.2-1. Seismic qualification of the containment vessel, reactor pressure vessel, upper reactor vessel internals, lower reactor vessel internals and reactor core, and control rod drive mechanisms is addressed in Appendix 3A. Seismic design and analysis of the Seismic Category I buildings are addressed in Section 3.7 and Section 3.8. Seismic qualification of the Reactor Building crane and the bioshield are addressed in Section 3.7.3, and seismic qualification of the spent and new fuel racks is addressed in Section 9.1.

The information presented or referenced in this section includes the following:

- identification of the Seismic Category I equipment and supports
- criteria used for seismic qualification of the various types of equipment
- list of the safety-related functional requirements of equipment to be qualified
- definition of the seismic load inputs
- definition of other relevant dynamic load inputs and load combinations
- documentation of the qualification process

This section demonstrates that subject equipment conforms to the requirements of General Design Criteria 1, 2, 4, 14, and 30 as well as Appendix B and Appendix S to 10 CFR 50.

COL Item 3.10-1: A COL applicant that references the NuScale Power Plant design certification will develop and maintain a site-specific seismic and dynamic qualification program.

3.10.1 Seismic Qualification Criteria

3.10.1.1 Qualification Standards

The methodologies for seismic and dynamic qualification of mechanical and electrical equipment are described in Section 3.10.2. These methods are in compliance with the requirements of General Design Criteria 1, 2, 4, 14, 30, and 10 CFR 50 Appendix S. The methods used to implement the requirements of 10 CFR 50, Appendix B are described in Chapter 17.

The NuScale Power Plant implements the requirements of the IEEE 344-2004 standard (Reference 3.10-1) endorsed by RG 1.100 Revision 3. Seismic Category I pressure boundary components are designed in accordance with the requirements of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (Reference 3.10-2) to ensure their structural integrity. Other Seismic Category I equipment is qualified in accordance with IEEE 344-2004.

Qualification by analysis is performed when any of the following conditions are met:

- The only safety-related function of the equipment is to maintain its structural integrity.
- The equipment is too large to test at existing test facilities.
- The interfaces, such as interconnecting cables in a cable cabinet, cannot be regarded as conservatively modeled during testing because of the complexity of the linkage to the equipment subject to testing.
- The equipment has a linear or very simple nonlinear response that can be conservatively calculated by analysis.

The methods and requirements of ASME QME-1-2007 (Reference 3.10-3) as described in RG 1.100 are also used for the seismic qualification of active mechanical equipment.

The qualification of the electrical and mechanical equipment is based on the certified seismic design response spectra and the certified seismic design response spectra - high frequency defined in Section 3.7.1. The certified seismic design response spectra (including the certified seismic design response spectra - high frequency) is the site-independent SSE.

The operating basis earthquake (OBE) is defined as one third of the SSE. Therefore an explicit analysis or design is not required per Appendix S of 10 CFR 50. As a result, the low-level seismic effects (fatigue) required by Institute of Electrical and Electronics Engineers (IEEE) 344-2004 (Reference 3.10-1) to qualify electrical and mechanical equipment are considered using two SSE events, with 10 maximum stress-cycles each, for a total of 20 full cycles. This is considered equivalent to the cyclic load basis of one SSE and five OBEs. The determination of number of earthquake cycles is outlined in Section 3.7.3.

The equipment qualification program is described in Section 3.11. The methodology for seismic analysis of systems is provided in Section 3.7.3. A list of safety-related active valves, in accordance with the guidance of RG 1.100, is provided in Section 3.9.6.

3.10.1.2 Performance Requirements for Seismic Qualification

An equipment qualification record file (EQRF) is developed for each piece of electrical equipment and instrumentation classified as Seismic Category I. Section 3.11 and Appendix 3C provide the environmental conditions of the mechanical and electrical equipment, including the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including seismic events. The performance requirements for the electrical equipment and instrumentation are defined in the EQRF. The test response spectrum (TRS) and required response spectrum (RRS) for the seismic qualification are also identified in the EQRF. The RRS is bounded by the TRS to demonstrate the conservative qualification of equipment.

For Seismic Category I active mechanical equipment, the performance requirements are defined in the corresponding equipment requirements specification. Requirements for active valves and dampers are addressed in EQRFs. Non-active Seismic Category I mechanical equipment have a single performance requirement - to maintain their structural integrity.

3.10.1.3 Performance Criteria

The qualification of Seismic Category I mechanical and electrical equipment demonstrates that the equipment is capable of performing its safety-related function under applicable plant loading conditions, including the SSE as defined in Section 3.7.1, in concert with other concurrent loadings.

3.10.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation

The guidance and requirements of RG 1.100 Revision 3 and IEEE 344-2004 (Reference 3.10-1) are the source of the methods and procedures used for seismic and dynamic qualification of mechanical and electrical equipment. ASME QME-1-2007 (Reference 3.10-3) is used with the exceptions noted in RG 1.100 Revision 3 for the qualification of active mechanical equipment.

The Seismic Category I equipment is qualified to withstand the SSE in combination with other relevant static and dynamic loads with no adverse impacts to the safety functions. The acceptable load combinations for mechanical equipment are defined in Section 3.9.3.

Seismic Category I instrumentation and electrical equipment are qualified by type testing or by a combination of testing and analysis. The choice of qualification method is a function of factors such as expense, viability, equipment complexity, and previous seismic qualification test data. The qualification method for a particular instrument or piece of electrical equipment is identified in the EQRF.

The structural integrity and operability of active valves and dampers is qualified by a combination of analyses and tests. Other mechanical components are qualified by analysis.

3.10.2.1 Qualification by Testing

Seismic qualification of mechanical and electrical equipment by testing is performed in accordance with the requirements of IEEE 344-2004 (Reference 3.10-1). For equipment qualified by testing, the test simulates normal loadings, such as thermal and flow-induced loads, concurrently with the seismic and other dynamic loadings. The loads include forces imposed by piping onto the equipment. The survival and operability of the equipment is verified during and after the testing.

The seismic testing consists of subjecting the equipment to vibratory motion that simulates the vibratory motion postulated to occur at the equipment mounting location. The testing conservatively considers the multi-dimensional effects of the postulated earthquake.

Single-frequency and multi-frequency tests are used for seismic qualification. The in-structure floor response spectra damping values provide the seismic and dynamic test inputs. The purpose of multi-frequency testing is to provide a broadband test motion that can produce a simultaneous response from multiple modes of a multi-degree-of-freedom system, the malfunction of which can be caused by modal interactions. It is preferable to perform multi-frequency testing rather than single-frequency testing because of the usually broad frequency content of the seismic and dynamic load excitation.

However, single-frequency testing, such as sine beats, may be used in the following situations:

- when seismic ground motion is filtered due to a single predominant structural mode
- when it can be shown that the anticipated response of the equipment is sufficiently represented by a single mode
- when the input has enough duration and intensity to cause the excitation of the applicable modes to the required magnitude, causing the TRS to bound the corresponding spectra
- when the resultant floor motion consists of a single predominant frequency

The test input motions are applied to two perpendicular horizontal axes or a vertical and a horizontal axis for the seismic and dynamic part of the load unless it can be shown that the sensitivity of the equipment response to vibratory motion in the horizontal direction is insignificant. To avoid an exclusively rectilinear input motion, the time phasing of the inputs in each direction are chosen carefully. Alternatively, the test may be conducted with the horizontal and vertical inputs in-phase and then the test is repeated, after rotating the equipment 90 degrees horizontally with the horizontal and vertical inputs 180 degrees out-of-phase.

The equipment mounting in the test setup simulates the equipment mounting in service and does not cause nonrepresentative dynamic coupling of the equipment to its mounting fixture. The test simulates the dynamic coupling effects of cable, conduit, instrument lines, electrical connects, and other interfaces, unless adequate justification is provided. The testing also simulates the effects of aging, such as the fatigue effects of five OBEs plus the loadings associated with normal operation for the design life of the equipment prior to simulating the effects of an SSE, which is equivalent to two SSEs, with 10 stress cycles each, per Section 3.10.1.1.

3.10.2.2 Qualification by Analysis

Qualification by analysis is performed on equipment that is only required to maintain its structural integrity to perform its safety function. IEEE 344-2004 (Reference 3.10-1) describes a methodology for calculating the fatigue associated with aging and OBEs. The methods of qualification by analysis are dynamic analysis and static coefficient analysis. The analysis accounts for the complexity of the equipment and accurately represents the response of the equipment to seismic excitation. The two methods of analysis are described below. The analysis shows that the fatigue-inducing effects of the OBEs in combination with other normal, fatigue-inducing operational loads followed by an SSE do not cause the failure of the analyzed equipment to perform its safety function.

For analyses in which multi-module and multi-directional responses are combined, the analyses use the guidance of RG 1.92 Revision 3 “Combining Modal Responses and Spatial Components in Seismic Response Analysis.”

Dynamic Analysis

The mass distribution and stiffness characteristics of the equipment and equipment supports are represented by an appropriate model. To determine whether the equipment is rigid or flexible, a modal analysis is performed. If the model has no resonances in the frequency range below the cutoff frequency of the RRS, the equipment is considered rigid and may be analyzed statically. For flexible equipment, a response spectrum analysis or a time history analysis is used to analyze the model.

Static Coefficient Analysis

The static coefficient analysis method is an alternative to dynamic analysis and includes more conservatism. Natural frequencies do not need to be determined to perform static coefficient analysis. The equipment's acceleration response is assumed to be the maximum acceleration in the amplified region peak of the RRS at a conservative and justifiable value of damping. The effects of multi-frequency excitation and multi-mode response for linear frame-type structures that can be represented by a simple model, such as members like beams and columns, are approximated by a static coefficient of 1.5. A lower static coefficient may be used if the result can be shown to maintain conservatism.

To perform a static coefficient analysis the seismic forces acting on equipment or components are calculated by multiplying the equipment or component's mass by the maximum peak RRS and the static coefficient. The resulting force is distributed over the component proportionally to the mass distribution. The stress is calculated by combining the stress in each direction at the point of interest due to the seismic forces using the square root of the sum of the squares method.

The static analysis method is not sufficient for qualification of active equipment because this analysis is only used for structural integrity.

The following are typical analyses that are used for qualification:

- to determine the input response of sub-assemblies or sub-components of equipment subject to testing
- to determine whether the natural frequency of the pump shaft or rotor is within the frequency range of the vibratory excitations
- to determine the differential pressure acting on a valve disc that considers system arrangement and valve closing dynamics, including the differential pressure and impact energy effects of a loss-of-coolant accident
- to verify the resultant maximum calculated stress in the valve body is within the limits defined in ASME Section III

3.10.2.3 Qualification by Testing and Analysis

When testing or analysis alone are not practical to sufficiently qualify equipment, combined testing and analysis methods are used. The requirements of IEEE 344-2004 (Reference 3.10-1) are used to perform equipment qualification by combined testing and analysis. Operability and structural integrity of components are demonstrated by calculating component deflections and stresses under various loads. These results are then compared to the allowable levels, per the applicable codes.

3.10.3 Methods and Procedures for Qualifying Supports of Mechanical and Electrical Equipment and Instrumentation.

Testing or analysis is used to qualify Seismic Category I mechanical and electrical equipment to demonstrate their structural integrity, including the structural integrity of their anchorage, and their ability to withstand seismic excitation corresponding to the RRS for the equipment's mounting configuration.

The qualification of supports for electrical equipment and instrumentation, which includes electrical cabinets, control consoles, electrical panels, and instrument racks, uses the installed equipment or a dummy weight to simulate the inertial effects and dynamic coupling to the support. The stresses and deflections are compared to the applicable codes and regulations. When testing is not practical, equipment may be analyzed to confirm their structural integrity. The analysis accounts for the complexity of the supports and accurately represent the response to seismic excitation and vibratory motions.

The RRS includes a 1.5 performance-based factor for the critical equipment during severe accident scenarios. This conservatism provides for the effects of a combined multi-mode response. Choosing safety factor depends on the shape of the RRS with the largest value, 1.5, applicable to a broadband RRS. Therefore, the RRS does not necessarily need to be fully enveloped by the TRS. If the equipment's resonances can be determined by testing, the single-frequency TRS needs to envelop the RRS at the resonances of the equipment with one single-frequency input.

The mounting location determines the input motion the equipment is subjected to for the qualification test. Equipment supports are tested using the same methodology employed to qualify equipment. For equipment installed in a non-operational configuration for the support test, the support's response during the test at the location of the equipment's mounting is monitored and described by a TRS used for separate functional qualification of the equipment. The TRSs resemble and envelop the RRS to seismically qualify the support.

The seismic qualification of equipment requires consideration of actual or installed equipment mounting. The mounting conditions and methods for the tested or analyzed equipment simulate the expected or installed conditions. The mountings are designed to avoid extraneous dynamic coupling. The equipment mounting considered in the analysis or testing is identified in the EQRF.

3.10.4 Test and Analysis Results and Experience Database

The results of seismic qualification testing and analysis, per the criteria in Section 3.10.1, Section 3.10.2 and Section 3.10.3 are included in the corresponding EQRFs. The EQRF files are created and maintained during the equipment selection and procurement phase for the equipment requiring qualification. A detailed description of the equipment and their support structures, qualification methodology, test and analysis results are described in the EQRF. The EQRFs are updated and modified as new tests and analyses are performed. The experience database containing plant EQRF data is maintained for the life of the plant. Information to be included in the EQRFs include the following:

- detailed equipment information to include location in building, supplier or vendor, make and model, serial number
- components of the reactor coolant pressure boundary are identified
- the type of support used to mount the equipment
- the weight, dimensions, and physical characteristics of the equipment
- the function of the equipment
- the loads and load intensities for which the equipment is qualified
- for equipment qualified by testing, the test procedures and methods, a description of the test, parameters of the test, and the results of the test
- for equipment qualified by analysis, the analytical methods, assumptions, and results
- the equipment's natural frequencies
- the methods used to qualify equipment for vibration-induced fatigue cycle effects if applicable
- suitability for inspection
- identification of whether or not equipment is installed
- the associated RRS or time-history and the applicable damping for normal loadings and other dynamic loadings in conjunction with the specified seismic load

COL Item 3.10-2: A COL applicant that references the NuScale Power Plant design certification will develop the equipment qualification database and ensure equipment qualification record files are created for the structures, systems, and components that require seismic qualification.

COL Item 3.10-3: A COL applicant that references the NuScale Power Plant design certification will submit an implementation program for Nuclear Regulatory Commission approval prior to the installation of the equipment that requires seismic qualification.

3.10.5 References

3.10-1 Institute of Electrical and Electronics Engineers,"IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," IEEE Standard 344-2004, June 2005.

3.10-2 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section III, Rules for Construction of Nuclear Facility Components, New York, NY.

3.10-3 American Society of Mechanical Engineers, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," ASME QME-1-2007 Edition, November 2007.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

This section provides the methodology for Environmental Qualification (EQ) of equipment and identifies the equipment that is within the scope of 10 CFR 50.49 including instrumentation and control (I&C) and certain post-accident monitoring equipment specified in Regulatory Guide (RG) 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.” The EQ program described in this section also includes the environmental qualification of active mechanical equipment that performs a design function related to safety. The EQ program complies with the requirements of 10 CFR 50, Appendix A, General Design Criteria (GDC) 1, 2, 4, and 23, and 10 CFR 50, Appendix B, Quality Assurance Criteria III, XI, and XVII.

This section addresses equipment that is capable of performing design functions related to safety under normal environmental conditions, anticipated operational occurrences, accident, and post-accident environmental conditions.

Mechanical, electrical, and I&C equipment associated with systems that are essential for emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal, or equipment otherwise essential in preventing significant release of radioactive material to the environment is reviewed to determine whether they are required to be environmentally qualified to meet their intended design function related to safety.

Included in this equipment scope is:

- equipment that performs these functions automatically
- equipment that is used by the operators to perform these functions manually
- equipment that may mislead an operator
- equipment whose failure can prevent the satisfactory accomplishment of one or more of the above design functions related to safety
- electrical equipment (including I&C) as described in 10 CFR 50.49 (b)(1) and (b)(2)
- post-accident monitoring (PAM) equipment as described in 10 CFR 50.49(b)(3)

The equipment qualification program also includes dynamic effects on and seismic qualification of safety-related electrical and mechanical equipment, which are addressed in Section 3.10.

The portions of post-accident monitoring equipment required to be environmentally qualified are discussed in Section 3.11.2.1.

Compliance with the regulatory requirements cited above as they apply to the EQ program is discussed below.

- GDC 1 requires that components are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Components in the scope of this section that are subject to environmental design and qualification are required to have auditable records to document that environmental design and qualification requirements have been met.
- GDC 2 requires that components are designed to withstand the effects of natural phenomena without loss of capability to perform their safety function. Components in the

scope of this section that are subject to environmental design and qualification are designed with consideration of the environmental conditions or effects resulting from natural phenomena as part of the environmental conditions evaluated, including their location within safety designed structures. Additional information is provided in Section 3.2.

- GDC 4 requires that components are designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents. Components in the scope of this section are protected against dynamic effects, including those of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. Components in the scope of this section are also designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.
- GDC 23 requires that protection systems are designed to fail in a safe state, or in a state demonstrated to be acceptable on some other defined basis, if conditions such as postulated adverse environments (e.g., extreme heat or cold, pressure, steam, water, or radiation) are experienced. Components in the scope of this section that are subject to environmental design and qualification requirements are designed with consideration of the failure mode of the equipment.
- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criteria III, "Design Control." The safety-related I&C systems are designed in compliance with Criterion III as discussed in Section 7.2.2. This criteria is included in establishing the regulatory requirements for the environmental program as discussed in Appendix 3.C for prototype designs.
- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criteria XI, "Test Control." This criteria is included in establishing the test procedures for the environmental program as discussed in Appendix 3.C.
- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criteria XVII, "QA Records." This criteria is included in establishing the regulatory requirements for the environmental program as discussed in Appendix 3.C.
- 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," establishes the specific requirements for the environmental qualification of certain electric equipment located in a "harsh" environment. Environmental qualification of electric equipment located in a "mild" environment is not included within the scope of 10 CFR 50.49. A "mild" environment is defined as an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences. This section assures conformance to 10 CFR 50.49 for the environmental qualification of electrical equipment performing a design safety function that is located in a harsh environment. See Appendix 3.C for more details.

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Equipment Identification

Equipment identification includes electrical and mechanical equipment that perform a design function related to safety for a Design Basis Accident (DBA) or Infrequent Event (IE) that results in a significant change in environmental conditions within the plant that has the potential to result in environmentally induced common cause failures. The identification of equipment that requires environmental qualification is specific to:

a) Equipment that is relied upon to detect and mitigate a DBA or IE that produces a harsh environment.

b) Equipment with design function related to safety that is relied upon for its ability to achieve or maintain a safe shutdown condition for a DBA or IE that produces a harsh environment.

c) Certain post-accident monitoring equipment.

The equipment subject to environmental qualification consists of mechanical, electrical, and I&C equipment located in either harsh or mild environments. NuScale Equipment required to be environmentally qualified has one or more of the following design functions related to safety: reactor trip, engineered safeguards actuation, post accident monitoring, or containment isolation.

For electrical and mechanical devices located in mild environments, compliance with the environmental design provisions of GDC 4 are generally achieved and demonstrated by proper incorporation of all relevant environmental conditions in the design process, including the equipment specification compliance.

The list of equipment that is in harsh environments and required to be environmentally qualified is provided in Table 3.11-1. The equipment listed applies to an individual module. Equipment location zones indicated in Table 3.11-1 are shown in Table 3.11-2.

3.11.1.2 Definition of Environmental Conditions

The environmental conditions considered in design include anticipated operational occurrences and normal, accident and post-accident environmental conditions. The environmental parameters (e.g., radiation, temperature, chemical effects, humidity from steam, pressure, wetting, submergence) applicable to the various environmental conditions in specific plant building and room locations are specified in Appendix 3.C.

Aging and synergistic effects of environmental conditions are considered when such effects are believed to have a significant affect on equipment performance and are further discussed in Appendix 3.C.

Service conditions are the environmental, physical, mechanical, electrical, and process conditions anticipated and/or experienced by equipment during operation of the plant. Operation includes both normal and abnormal operations. Abnormal conditions occur during plant transients, system transients, or in conjunction with certain equipment or system failures.

Electromagnetic compatibility is a design requirement for plant equipment, especially within digital I&C systems. Electromagnetic compatibility testing requirements specified in Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," for radiated and conducted interference are performed to show that critical equipment will not be adversely affected by electronic interference (EMI) or radio frequency interference (RFI) in the plant environment.

Service condition environments fall into two categories:

- A harsh environment
  - is any significant change from normal (including design basis event and post-accident conditions) that has the potential to result in environmental and/or radiation induced common-cause failure mechanisms. Seismic-related design basis events are excluded from harsh environments. Seismic and dynamic qualification are discussed in Section 3.10.
  - is an environment that is the result of events as cited above that significantly alters the environmental parameters of temperature, pressure, humidity, and/or flooding such as:
    - Temperature:
      - ≥120F and >18F increase above normal operating conditions with >85% RH
    - Humidity:
      - Steam Exposure:
        - >99% RH condensing conditions for electrical equipment
        - ≥85% RH with temperatures ≥120F for electronic equipment
      - Submergence:
        - Areas where equipment is subject to submergence that is not subjected to submergence under normal operating conditions
  - is plant areas where the radiation levels exceed the following thresholds:
    - Greater than 1.0E04 Rads gamma for electrical and mechanical equipment including non-metallics or consumables (e.g., O-rings, seals, packing, gaskets, lube oil, diaphragms).
    - Greater than 1.0E03 Rads gamma for electronic devices and components.
- A mild environment
  - is plant areas where the environment at no time would be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.

– is an area not subject to design basis events (excluding seismic events) and whose radiation levels are less than or equal to the thresholds discussed above for mechanical and electrical equipment.

3.11.1.3 Equipment Post-Accident Operating Times

The post-accident operating time is the period of time, beginning with design basis event initiation, during which the equipment must continue to perform its design function related to safety. The post-accident operating time, or operating time, duration can vary and is based on the required safety function of the equipment. Both operating and "not failing" in a manner detrimental to plant safety can be required safety functions.

A post-accident operating time is determined for the equipment in the EQ Master List. The required post-accident operating time for equipment varies from less than or equal to 1 hour to 2400 hours. The operating times are conservatively based on the operability requirements established for post-accident monitoring equipment and equipment required for long term core cooling. Post-accident operating times are specified in Table 3.11-1 for the equipment/instrumentation listed.

The operating times for electrical and mechanical equipment located in harsh environments listed in Table 3.11-1 are defined and documented in Table 3.C-4 and the EQ Master List.

3.11.2 Qualification Tests and Analysis

3.11.2.1 Environmental Qualification of Electrical Equipment

Electrical equipment, which includes I&C, that is environmentally qualified contains components associated with systems that are essential to emergency reactor shutdown, containment isolation, core cooling, containment and reactor heat removal, or essential to preventing significant release of radioactive material to the environment. The results of the qualification testing or analysis are presented in the equipment qualification record file per Appendix 3.C.

For electrical equipment that is required to function during or following exposure to a harsh environment, compliance with the environmental provisions of GDC 4 are achieved by demonstrating compliance with 10 CFR 50.49. Electrical equipment identified to be in a harsh location, as described in Section 3.11.1.2, are environmentally qualified by type testing or type testing and analysis using the guidance of IEEE Std. 323-1974 (Reference 3.11-2) for harsh environment equipment, IEEE Std. 323-2003 (Reference 3.11-12) (as endorsed by Regulatory Guide 1.209) for mild environment equipment, and related standards that are described in Appendix 3.C, Section 3.C.6. The specific testing, type testing and analysis are described in more detail in Appendix 3.C, Section 3.C.6 and Section 3.C.7.

Regulatory Guides 1.63, 1.73, 1.89, 1.97, 1.152, 1.153, 1.156, 1.158, 1.180, 1.183, 1.209 and 1.211 used for the EQ Program provides guidance for meeting the requirements of GDC 1, 2, 4 and 23; and 10 CFR 50 Appendix B, Criterion III, XI, and XVII, and 10 CFR 50.49. A comparison of the related qualification standards and the associated RG that endorses them is provided in Appendix 3.C. Appendix 3.C also provides a summary of the related qualification standards that are not associated with a Regulatory Guide.

The design does not have any environmentally qualified continuous duty motors. Therefore, the guidance provided by RG 1.40 is not applicable below.

Environmental qualification of electrical and active mechanical equipment meets the relevant guidance documents except as noted and applicable.

Regulatory Guide 1.63 (endorsing IEEE Std. 317-1983 (Reference 3.11-3)):

For external circuit protection of electrical penetration assemblies, IEEE Std. 741-1997 (Reference 3.11-4) is used. Although not endorsed by Regulatory Guide 1.63, the design philosophy would not deviate from the existing RG.

Regulatory Guide 1.73 (endorsing IEEE Std. 382 (Reference 3.11-5)):

This guidance is applicable except for portions directed towards high temperature gas-cooled reactor designs.

Regulatory Guide 1.89 (endorsing IEEE Std. 323 (Reference 3.11-2) and implementing criteria of 10 CFR 50.49):

NUREG-0588 (Reference 3.11-1) Category I guidance may be used to enhance the guidance provided by the RG.

Regulatory Guide 1.97 (as supplemented by RG 1.89): Post Accident Monitoring (PAM) equipment is environmentally qualified in accordance with Regulatory Guide 1.97, Rev 4. PAM equipment is identified as Type A, B, C, D or E, according to RG 1.97, Rev 4, and Type A, B, C and D is environmentally qualified as required by 10 CFR 50.49 and the guidelines of Branch Technical Position (BTP) 7-10. Type E variables are not required to be environmentally qualified. Compliance with RG 1.97, Revision 4, and the method used to identify and qualify this equipment is described in Section 7.2.13.5. The NuScale design does not include any Type A PAM variables by design.

Regulatory Guide 1.152 (endorsing IEEE Std.7-4.3.2-2003 (Reference 3.11-6)):

No exceptions.

Regulatory Guide 1.153 (endorsing IEEE Std. 603-1991 (Reference 3.11-7)):

No exceptions.

Regulatory Guide 1.156 (endorsing IEEE Std. 572-2006 (Reference 3.11-8)):

These criteria are used in conjunction with Regulatory Guide 1.89 as a method of demonstrating compliance pertaining to the environmental qualification of connectors, terminators, and environmental seals in combination of wires as assemblies for service to ensure that the connection assemblies can perform their design functions related to safety.

Regulatory Guide 1.158 (endorsing IEEE Std. 535-1986 (Reference 3.11-9)):

No exceptions.

Regulatory Guide 1.180:

Refer to Section 7.2.2.1 for additional details of EMI/RFI qualification.

Regulatory Guide 1.183:

NuScale Topical Report TR-0915-17565-P (Reference 3.11-10) and Section 12.2.1.13 describes an alternate methodology for source terms for design basis events.

Regulatory Guide 1.209 (endorsing in part IEEE Std. 323-2003 (Reference 3.11-12):

No exceptions.

Regulatory Guide 1.211 (endorsing in part IEEE Std. 383-2003 (Reference 3.11-11)):

No exceptions.

The acceptability of electrical equipment located in a mild environment and not subject to 10 CFR 50.49 or eletromagnetic compatibility is demonstrated and maintained by use of the following types of programs:

- A periodic maintenance, inspection or replacement program based on sound engineering practice and recommendation of the equipment manufacturer, which is updated as required by the results of an equipment surveillance program.
- A periodic testing program used to verify operability of safety-related equipment within its performance specification requirements. System level testing of the type typically required by the plant technical specifications may be used.
- An equipment surveillance program that includes periodic inspections, analysis of equipment and component failures, and a review of the results of the preventive maintenance and periodic testing program.

3.11.2.2 Mechanical Equipment Environmental Qualification

Mechanical equipment environmental qualification is described in Section 3.11.6.

3.11.2.3 Justification for Using Latest IEEE Standards Not Endorsed by a Regulatory Guide

This section provides the description and justification for using the latest IEEE standards not endorsed by current Regulatory Guides for the qualification of equipment. This justification does not preclude the use of versions of IEEE standards that are currently endorsed by Regulatory Guides.

The IEEE has periodically updated the standards to incorporate evolutionary thinking and approaches of the nuclear industry with regard to equipment qualification.

Section 3.11.2.1 provides a summary comparison of the current IEEE standards to be used for equipment qualification and the associated RG and revision that endorse them. Recent IEEE standards, not currently endorsed by the NRC, are discussed and justified below.

3.11.2.3.1 IEEE Std. 741-1997, IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations

Regulatory Position C of RG 1.63, Revision 3 endorses Section 5.4 of IEEE Std. 741-1986 for external circuit protection of electric penetration assemblies. The IEEE Std. 741-1997 (Reference 3.11-4) is incorporated for use in the design as the design philosophy does not deviate from the existing RG.

3.11.3 Qualification Test Results

The summaries and results of qualification tests for electrical and mechanical equipment and components are documented in the equipment qualification record file per Appendix 3.C.

Qualification of equipment in mild environments is based on certification of performance in accordance with applicable regulatory guidance as identified in Section 3.11.2. Additional information is provided in Appendix 3.C and seismic qualification program is described in Section 3.10.

The summaries and results of seismic qualification tests for electrical and mechanical equipment and components in the harsh environment areas are documented in the equipment qualification record file and maintained throughout the life of the plant in accordance with 10 CFR 52.79(a)(10) and 10 CFR 52.80(a).

COL Item 3.11-1: A COL applicant that references the NuScale Power Plant design certification will submit a full description of the Environmental Qualification Program and milestones and completion dates for program implementation.

COL Item 3.11-2: A COL applicant that references the NuScale Power Plant design certification will develop the equipment qualification database and ensure equipment qualification record files are created for the structures, systems, and components that require environmental qualification.

3.11.4 Loss of Ventilation

For equipment and instrumentation that is challenged by a loss of environmental control, such as an increase in area temperature, the heat capacity of the enclosing building concrete will provide a heat sink sufficient to maintain the area temperature within the bounds of the environmental parameters for which the equipment and/or instrumentation was qualified. Within 72 hours of an event resulting in the loss of ventilation, normal HVAC will be restored.

Due to the slow progress of this transient, an operator would have sufficient time to implement corrective actions to restore the HVAC system or provide a temporary alternative means to maintain normal operating temperatures.

The normal and abnormal environmental conditions shown on Appendix 3.C, Table 3.C-6 and Table 3.C-7, reflect anticipated normal and maximum conditions. The HVAC systems in the standard design are non-safety related and are assumed to not be functional during design basis events (except in cases where operation may result in more severe environmental conditions for equipment).

3.11.5 Estimated Chemical and Radiation Environment

3.11.5.1 Chemical Environments

Applicable chemical environments are defined in Appendix 3.C for normal and abnormal operating conditions. The chemical environments from the most limiting design basis event is also considered in the qualification of the equipment and presented in Appendix 3.C.

Chemicals that are used for water chemistry and pH control have been considered as well as the borated water environment that will be present inside containment and outside containment. Water chemistry is discussed in Section 5.2.3.2.1 for primary side water chemistry, Section 6.1.1.2 for the reactor pool and spent fuel pool chemistry, and Section 10.3.5 for the secondary side water chemistry.

3.11.5.2 Radiation Environments

Radiation environments are defined in Appendix 3.C for normal and accident conditions.

Normal operation radiation doses are calculated for initial plant start-up conditions using the source terms and analysis. The radiation doses are continuously monitored during plant life and compared to the calculated doses. If the measured doses are higher than the calculated doses, the EQ Master List will be revised if an affected mild environment becomes harsh. Section 12.3 discusses the assumptions associated with the normal operations dose rates.

The normal operations dose rates for equipment qualification are derived from direct gamma emitted by radioactive fluids. Beta radiation and Bremsstrahlung radiation during normal operations are considered negligible contributors to doses in comparison to the gamma radiation and therefore are omitted. Normal doses within the CNV and other areas also account for neutron fluence, when applicable, by equating the neutron fluence to an equivalent dose in Rad. The loss-of-coolant accident dose rates include a submersion dose and a direct dose contribution. The submersion dose is derived from both the gamma and beta radiation. The beta radiation may be attenuated by low-density equipment enclosures. Alpha radiation is neglected from both the normal and accident equipment qualification dose rates because the alpha particle is easily attenuated by air.

In the event doses are determined to exceed the qualified dose for a specific piece of equipment, a component specific dose calculation may be performed to determine the component specific dose at the specific equipment location. The accident dose rates were calculated based on the methodology presented in Topical Report TR-0915-17565-P and Section 12.2.1.13. The assumptions associated with the accident dose rates are discussed in Section 15.0.3. See also the discussion in Appendix 3.C for additional information on normal and accident dose rates used for environmental qualification.

3.11.6 Qualification of Mechanical Equipment

Mechanical equipment is qualified and documented in accordance with the General Design Criteria 1, 2, 4, and 23 as demonstrated by the approach presented in this section.

GDC 1 and 4 and Appendix B to 10 CFR Part 50 (Criteria III, "Design Control," XI, “Test Control,” and XVII, "Quality Assurance Records") contain the following requirements related to generic equipment qualification methodology which applies to mechanical qualification of equipment:

- Components are designed to be compatible with the postulated environmental conditions, including those associated with loss-of-coolant accidents.
- Measures are established for the selection and review of the suitability of application of materials, parts, and equipment that are essential to safety-related functions.
- Design control measures are established for verifying the adequacy of design.
- Equipment qualification records are maintained and include the results of tests and materials analyses.

Mechanical components, including passive components, are qualified to perform their required functions under the appropriate environmental effects of normal, abnormal, accident, and post-accident conditions as required by GDC 4 and 10 CFR 50 Appendix B. Mechanical equipment qualification verifies the design is capable of functioning during normal, abnormal and accident conditions and includes the effects of the fluid medium (e.g., borated water) on the environmental conditions.

For mechanical equipment located in a mild environment, acceptable environmental design is demonstrated by the design and purchase specifications for the equipment. The specifications contain a description of the functional requirements for a specific environmental zone during normal environmental conditions and anticipated operational occurrences. The programs identified in Section 3.11.2.1 for verifying that electrical equipment located in a mild environment are capable of performing their intended function will also be applied to mechanical equipment located in a mild environment. For mechanical equipment that must function during or following exposure to a harsh environment, compliance with the environmental design provisions of GDC 4 are generally achieved by demonstrating that the non-metallic parts/components of the equipment suitable for the postulated design basis environmental conditions. Safety-related mechanical equipment that performs an active function during or following exposure to harsh environmental conditions will be qualified in accordance with ASME QME-1, Appendix QR-B (Reference 3.11-13), as described in Section 3.10. Documentation and the status of the testing and analysis are performed in accordance with the processes presented in Appendix 3.C.

Mechanical equipment located in harsh environmental zones is designed to perform under all appropriate environmental conditions. The primary focus with mechanical equipment is on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). A list of the mechanical components that contain non-metallic or consumable parts located in harsh environment areas that require EQ is provided in Table 3.11-1.

3.11.7 References

3.11-1 NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment," Revision 1, July 1981.

3.11-2 IEEE Std. 323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

3.11-3 IEEE Std. 317-1983, Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generation Stations.

3.11-4 IEEE Std. 741-1997, IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations.

3.11-5 IEEE Std. 382-1972, Trial-Use Guide for Type Test of Class 1 Electric Valve Operators for Nuclear Power Generating Stations.

3.11-6 IEEE Std. 7-4.3.2-2003, Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations.

3.11-7 IEEE Std. 603-1991, Criteria for Safety Systems for Nuclear Power Generating Stations.

3.11-8 IEEE Std. 572-2006, IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations.

3.11-9 IEEE Std. 535-1986, IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations.

3.11-10 Topical Report TR-0915-17565-P, Accident Source Term Methodology.

3.11-11 IEEE Std. 383-2003, IEEE Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations.

3.11-12 IEEE Std. 323-2003, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

3.11-13 American Society of Mechanical Engineers, ASME QME-1-2012, Qualification of Active Mechanical Equipment Used In Nuclear Facilities.

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
Nuclear Power Module	-					
Containment System (CNT-A013)	-					
I&C DIv I Nozzle	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
I&C DIv II Nozzle	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
PZR Heater Power #1 Nozzle	EQ Zone F DCA EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
PZR Heater Power #2 Nozzle	EQ Zone F DCA EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
I&C Channel A Nozzle	EQ Zone F DCA EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
I&C Channel B Nozzle	EQ Zone F DCA EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
I&C Channel C Nozzle	EQ Zone F DCA EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
I&C Channel D Nozzle	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
CRD Power Nozzle	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
RPI Group #1 Nozzle	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
RPI Group #2 Nozzle	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
MS #1 CIV (MSIV #1)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
MS #2 CIV (MSIV #2)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
MS line #1 Bypass Valve (MSIV Bypass #1)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
MS line #2 Bypass Valve (MSIV Bypass #2)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
FW #1 CIV (FWIV #1)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
FW #2 CIV (FWIV #2)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
FW line #1 Check Valve	EQ Zone G	Harsh	Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
FW line #2 Check Valve	EQ Zone G	Harsh	Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
CVC Discharge CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
CVC Injection CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
CVC PZR Spray CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
RPV High Point Degas CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
RCCW Supply CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
RCCW Return CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
CE CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
CFDS CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
Hydraulic Skid for valve reset	EQ Zone M EQ Zone N	Harsh	Electrical Mechanical	N/A	A	Short Term (<= 1 hr)
Containment Pressure Transducer (Narrow Range)	EQ Zone E EQ Zone F	Harsh	Electrical	N/A	A	Short Term (<= 1 hr)
Containment Pressure Transducer (Wide Range)	EQ Zone E EQ Zone F	Harsh	Electrical	C	A	Extended PAM (100 days)

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
Containment Water Level Sensors (Radar Transceiver)	EQ Zone B EQ Zone C EQ Zone D EQ Zone E EQ Zone F EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG #1 Steam Temperature Sensors (RTD)	EQ Zone G	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
SG #2 Steam Temperature Sensor (RTD)	EQ Zone G	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
CE Inboard CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CE Inboard CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CE Outboard CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CE Outboard CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CFD Inboard CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CFD Inboard CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CFD Outboard CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CFD Outboard CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Inboard RCS Discharge CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Inboard RCS Discharge CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Outboard CIV RCS Discharge Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
CVCS Outboard CIV RCS Discharge Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Inboard RCS Injection CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Inboard RCS Injection CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Outboard RCS Injection CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Outboard RCS Injection CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Inboard PZR Spray Line CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Inboard PZR Spray Line CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Outboard PZR Spray Line CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Outboard PZR Spray Line CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Inboard RPV High-Point Degasification CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Inboard RPV High-Point Degasification CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Outboard RPV High-Point Degasification CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
CVCS Outboard RPV High-Point Degasification CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
RCCW Supply Inboard CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
RCCW Supply Inboard CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
RCCW Supply Outboard CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
RCCW Supply Outboard CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
RCCW Return Inboard CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
RCCW Return Inboard CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
RCCW Return Outboard CIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
RCCW Return Outboard CIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
FW Supply to SG1 and DHR HX1 CIV/FWIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
FW Supply to SG1 and DHR HX1 CIV/FWIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
FW Supply to SG1 and DHR HX1 CIV/FWIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
FW Supply to SG1 and DHR HX1 CIV/FWIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
FW Supply to SG2 and DHR HX2 CIV/FWIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
FW Supply to SG2 and DHR HX2 CIV/FWIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
FW Supply to SG2 and DHR HX2 CIV/FWIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location$_{(1)}$	EQ Environment	Qualification Program	PAM$_{(2)}$	EQ Category$_{(3)}$	Operating Time
FW Supply to SG2 and DHR HX2 CIV/FWIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG1 Steam Supply CIV/MSIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG1 Steam Supply CIV/MSIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG1 Steam Supply CIV/MSIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG1 Steam Supply CIV/MSIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG1 Steam Supply CIV/MS Bypass Isolation Valve Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG1 Steam Supply CIV/MS Bypass Isolation Valve Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG1 Steam Supply CIV/MS Bypass Isolation Valve Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG1 Steam Supply CIV/MS Bypass Isolation Valve Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG2 Steam Supply CIV/MSIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG2 Steam Supply CIV/MSIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG2 Steam Supply CIV/MSIV Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG2 Steam Supply CIV/MSIV Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG2 Steam Supply CIV/MS Bypass Isolation Valve Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
SG2 Steam Supply CIV/MS Bypass Isolation Valve Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG1 Steam Supply CIV/MS Bypass Isolation Valve Close Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
SG1 Steam Supply CIV/MS Bypass Isolation Valve Open Position Sensor	EQ Zone G	Harsh	Electrical	C	A	Extended PAM (100 days)
Steam Generator System (SGS-A014)						
Thermal relief valves	EQ Zone C	Harsh	Mechanical	N/A	B	Extended Term (<= 720 hr)
Control Rod Drive System (CRDS-A022)	-					
Control Rod Drive Coils	EQ Zone E	Harsh	Electrical	N/A	A	Short Term (<= 1 hr)
Rod Position Indication (RPI) Coils	EQ Zone E	Harsh	Electrical	N/A	B	Extended Term (<= 720 hr)
CRDM Control Cabinet	EQ Zone N	Harsh	Electrical	N/A	A	Short Term (<= 1 hr)
Rod Position Indication Cabinets (Train A/B)	EQ Zone N	Harsh	Electrical	N/A	B	Long Term (<= 72 hr)
CRDS Cooling Water Piping and Pressure Relief Valve	EQ Zone E EQ Zone F	Harsh	Mechanical	N/A	B	Extended Term (<= 720 hr)
Reactor Coolant System (RCS-A030)	-					
PZR Control Cabinet	EQ Zone K EQ Zone L	Harsh	Electrical	N/A	A B	A Short Term (<= 1 hr)
Reactor Safety Valve Position Indicator	EQ Zone E	Harsh	Electrical	N/A	B	Extended Term (<= 720 hr)
Reactor Safety Valves	EQ Zone E	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
Narrow Range Pressurizer Pressure Elements	EQ Zone D EQ Zone E	Harsh	Electrical	N/A	A	Short Term (<= 1 hr)
Wide Range RCS Pressure	EQ Zone D EQ Zone E	Harsh	Electrical	C	A	Extended PAM (100 days)

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
PZR/RPV Level	EQ Zone E EQ Zone F EQ Zone G	Harsh	Electrical	B	A	Extended Term (<= 720 hr)
Narrow Range RCS Hot Leg Temperature Element	EQ Zone C	Harsh	Electrical	N/A	A	Short Term (<= 1 hr)
Wide Range RCS Hot Leg Temperature Element	EQ Zone C	Harsh	Electrical	B	A	Extended Term (<= 720 hr)
Wide Range RCS Cold Leg Temperature Element	EQ Zone B	Harsh	Electrical	N/A	B	Extended Term (<= 720 hr)
RCS Flow Transmitter (Ultrasonic)	EQ Zone B	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
NSSS Primary Systems	-					
Chemical and Volume Control System (CVCS-B010)	-					
DWS Supply Isolation Valve	EQ Zone J	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
DWS Supply Isolation Valve	EQ Zone J	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
DWS Supply Isolation Valve Position Indication	EQ Zone J	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
DWS Supply Isolation Valve Position Indication	EQ Zone J	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Discharge Spoolpiece Drain Valve	EQ Zone J	Harsh	Electrical Mechanical	N/A	A	Short Term (<= 1 hr) Extended Term (<= 720 hr)
Discharge PSS Isolation Valve	EQ Zone J	Harsh	Electrical Mechanical	N/A	A	Short Term (<= 1 hr) Extended Term (<= 720 hr)
RPV High Point Degasification Isolation Valve	EQ Zone G	Harsh	Electrical Mechanical	C	A	Short Term (<= 1 hr) Extended PAM (100 days)
Spray Check Valve	EQ Zone G	Harsh	Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
Injection Check Valve	EQ Zone G	Harsh	Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
Emergency Core Cooling System (ECCS-B020)	-					

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
Reactor Vent Valve	EQ Zone E	Harsh	Mechanical	N/A	A	Intermediate Term (<= 36 hr) Extended Term (<= 720 hr)
RVV Position Indication	EQ Zone E	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Reactor Recirculation Valve	EQ Zone B	Harsh	Mechanical	N/A	A	Intermediate Term (<= 36 hr) Extended Term (<= 720 hr)
RRV Position Indication	EQ Zone B	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
RVV Trip Valve	EQ Zone I	Harsh	Electrical Mechanical	N/A	A B	Intermediate Term (<= 36 hr) Extended Term (<= 720 hr)
RRV Trip Valve	EQ Zone I	Harsh	Electrical Mechanical	N/A	A B	Intermediate Term (<= 36 hr) Extended Term (<= 720 hr)
Trip Valve Position Indication	EQ Zone I	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Reset Valve	EQ Zone I	Harsh	Electrical Mechanical	N/A	A	Intermediate Term (<= 36 hr)
Reset Valve Position Indication	EQ Zone I	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Decay Heat Removal System (DHRS-B030)	-					
DHRS Actuation Valve (2 per side)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Short Term (<= 1 hr)
DHRS Condenser Outlet Temperature (2 per side)	EQ Zone I	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
DHRS Condenser Outlet Pressure (3 per side)	EQ Zone I	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
DHRS Valve Position Indicator (2 for open, 2 for close per side)	EQ Zone G	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
SG Steam Pressure (4 per side)	EQ Zone G	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Containment Evacuation System (B190)	-					
PSS Sample Panel Inlet Isolation Valve	EQ Zone M	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
PSS Sample Panel Outlet Isolation Valve	EQ Zone M	Harsh	Electrical Mechanical	N/A	A	N/A A Extended Term (<= 720 hr)
BOP Primary Systems	-					

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
Fuel Handling Equipment (FHE-B140)	-					
Fuel Handling Machine	EQ Zone H	Harsh	Electrical Mechanical	N/A	B	Extended Term (<= 720 hr)
Reactor Pool Cooling System (RPCS-B173)	-					
Instrumentation - temperature (24 total)	EQ Zone I	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Ultimate Heat Sink (UHS-B175)	-					
Pool Level instruments	EQ Zone I	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Balance of Plant Systems	-					
Main Steam System (MSS-C010)	-					
Secondary Main Steam Isolation Valve	EQ Zone M	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
Secondary Main Steam Isolation Valve Close Position Indicator	EQ Zone M	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Secondary Main Steam Isolation Valve Open Position Indicator	EQ Zone M	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Secondary Main Steam Isolation Valve	EQ Zone M	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
Secondary Main Steam Isolation Valve Close Position Indicator	EQ Zone M	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Secondary Main Steam Isolation Valve Open Position Indicator	EQ Zone M	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Secondary Main Steam Isolation Bypass Valve	EQ Zone M	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
Secondary Main Steam Isolation Bypass Valve Close Position Indicator	EQ Zone M	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Secondary Main Steam Isolation Bypass Valve Open Position Indicator	EQ Zone M	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Secondary Main Steam Isolation Bypass Valve	EQ Zone M	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
Secondary Main Steam Isolation Bypass Valve Close Position Indicator	EQ Zone M	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Secondary Main Steam Isolation Bypass Valve Open Position Indicator	EQ Zone M	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Condensate and Feedwater System (FWS-C020)	-					
Feedwater Regulating Valve A/B	EQ Zone M	Harsh	Electrical Mechanical	D	A	Extended Term (<= 720 hr)
Feedwater Regulating Valve A/B Position Indication	EQ Zone M	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Feedwater Supply Check Valve	EQ Zone M	Harsh	Mechanical	N/A	A	Extended Term (<= 720 hr)
Instrumentation and Controls	-					
Module Protection System (MPS-E011)	-					
Separation Group A - Under-the-Bioshield Temperature Sensors	EQ Zone G	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Separation Group B - Under-the-Bioshield Temperature Sensors	EQ Zone G	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
Separation Group C - Under-the-Bioshield Temperature Sensors	EQ Zone G	Harsh	Electrical	D	A	Extended Term (<= 720 hr)

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
Separation Group D - Under-the-Bioshield Temperature Sensors	EQ Zone G	Harsh	Electrical	D	A	Extended Term (<= 720 hr)
MCR Isolation Switches	EQ Zone K	Harsh	Electrical	N/A	B	Extended Term (<= 720 hr)
Neutron Monitoring System (NMS-E013)	-					
NMS-Flood Highly Sensitive Neutron Detectors (for CNV flooding events)	EQ Zone I	Harsh	Electrical	B	A	Extended Term (<= 720 hr)
NMS - Excore Neutron Detectors	EQ Zone I	Harsh	Electrical	B	A	Extended Term (<= 720 hr)
In-Core Instrumentation System (ICI-E034)	-					
In-core instrument string/ temperature sensors	EQ Zone E EQ Zone F	Harsh	Electrical	C	A	Extended PAM (100 days)
In-core instrument string sheath	EQ Zone E	Harsh	Mechanical	N/A	B	Extended Term (<= 720 hr)
Radiation Monitoring System (RMS-E120)	-					
RM system that monitors PAM B & C variables	EQ Zones G	Harsh	Electrical	C	A	Extended PAM (100 days)
Buildings and Structures	-					
Reactor Building Cranes (RBC-F011)	-					

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description	Location(1)	EQ Environment	Qualification Program	PAM(2)	EQ Category(3)	Operating Time
Reactor Building Crane	EQ Zone H	Harsh	Electrical Mechanical	N/A	B	Extended Term (<= 720 hr)

Notes:
1. Environmental Zone Locations are delineated in Table 3.11-2.
2. PAM Type Variables:
   - Type B: those variables that provide information to indicate whether plant design functions related to safety are being accomplished.
   - Type C: those variables to be monitored to provide information to indicate whether the primary reactor containment, the fuel cladding, or the reactor coolant pressure boundary remain intact and do not have a potential to be breached.
   - Type D: those variables that provide information to indicate the operation of individual safety systems and other systems that perform design functions related to safety. These variables are to help the operator make appropriate decisions in using the individual systems performing design functions related to safety in mitigating the consequences of an accident.
   - For PAM variables that are assigned multiple types (e.g., B, C, D), the indicated type reflects the type that results in the longest operating time requirement.
3. EQ Categories:
   - A Equipment that will experience the environmental conditions of design basis accidents for which it must function to mitigate said accidents, and that will be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.
   - B Equipment that will experience the environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that will be qualified to demonstrate the capability to withstand the accident environment for the time during which it must not fail with safety margin to failure.
4. This listing is based on a single module evaluation and does not consider multi-module interactions because the secondary module(s) effects that may be created by the primary module affected are enveloped by their own qualifications.

Table 3.11-2: Environmental Qualification Zones - Reactor Building

EQ Zone(1)	Description	Environment
A	Room 010-022, Containment Vessel - bottom of containment (6") to bottom of upper core plate (142")	Harsh
B	Room 010-022, Containment Vessel - bottom of upper core plate (142") to bottom of riser transition (236")	Harsh
C	Room 010-022, Containment Vessel - bottom of riser transition (236") to bottom of baffle plate (587")	Harsh
D	Room 010-022, Containment Vessel - bottom of baffle plate (587") to top of pressurizer (697")	Harsh
E	Room 010-022, Containment Vessel - top of pressurizer (697") to bottom of torispherical head (841")	Harsh
F	Room 010-022, Containment Vessel - bottom of torispherical head (841") to top of containment (904")	Harsh
G	Room 010-022, Module pool bay vapor space - outside containment and under the BioShield (Top of Module) (Figure 1.2-19: Reactor Building East and West Section View)	Harsh
H	Rooms 010-022, 010-422, and 010-423 above pool level to ceiling (RXB Pool Room Vapor Space) (Figure 1.2-16: Reactor Building 100'-0"' Elevation thru Figure 1.2-18: Reactor Building 145'-6" Elevation)	Harsh
I	Room 010-022, 010-023 and 010-024 up to top of pool level (RXB Pool Room liquid space) (Figure 1.2-10: Reactor Building 24'-0" Elevation)	Harsh
J	Rooms 010-101, 010-102, 010-103, 010-104, 010-005, 010-106, 010-107, 010-112, 010-114, 010-115, 010-116, 010-117, 010-118, 010-119, 010-120, 010-121, 010-122, 010-123, 010-125, 010-126, 010-127, 010-128, 010-129, 010-130, 010-131, 010-133, 010-134 (Figure 1.2-12: Reactor Building 50'-0" Elevation)	Harsh
K	Rooms 010-201, 010-202, 010-203, 010-204, 010-005, 010-206, 010-207, 010-208, 010-242, 010-275 (Figure 1.2-14: Reactor Building 75'-0" Elevation)	Harsh
L	Rooms 010-201, 010-202, 010-203, 010-204, 010-005 (Figure 1.2-15: Reactor Building 86'-0" Elevation)	Harsh

Table 3.11-2: Environmental Qualification Zones - Reactor Building (Continued)

EQ Zone(1)	Description	Environment
M	Rooms 010-005, 010-401, 010-402, 010-403, 010-404, 010-405, 010-406, 010-407, 010-408, 010-409, 010-410, 010-411, 010-412, 010-414, 010-415, 010-416, 010-417, 010-418, 010-419, 010-420 (Figure 1.2-16: Reactor Building 100'-0" Elevation)	Harsh
N	Rooms 010-005, 010-501, 010-502, 010-503, 010-504, 010-506, 010-507, 010-508, 010-509, 010-510 (Figure 1.2-17: Reactor Building 126'-0" Elevation)	Harsh

Note:

1) EQ Zones listed are those areas within the Reactor Building that are harsh environments and contain equipment that requires environmental qualification.

3.12 ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports

3.12.1 Introduction

This section addresses the design of the piping systems and piping supports used in Seismic Category I, Seismic Category II, and nonsafety-related systems. The information in this section primarily addresses ASME Class 1, 2, and 3 piping systems. The analysis of the piping also considers interaction of non-Seismic Category I piping and associated supports with Seismic Category I piping and associated supports.

NuScale has adapted a graded level of detail approach in piping design. This approach is discussed in the March 4, 2014 NRC white paper - Piping Level of Detail for Design Certification (Reference 3.12-12). Piping system designs (e.g., layout, pipe size) for the systems within the NuScale Power Module (NPM) are generally complete and the requirements for the design, analysis, materials, fabrication, inspection, examination, testing, certification, packaging, shipping, and installation of these systems are documented in an ASME design specification for Class 1, 2, & 3 piping. The highest level of detail is complete for Class 1 reactor coolant pressure boundary (RCPB) piping (NPS 2) inside containment and Class 2 main steam, feedwater and decay heat removal system (DHRS) lines up to the first 6-way rigid restraint beyond the containment isolation valves. Preliminary analyses are performed for all of these systems in order to confirm the adequacy of the piping layout and support locations. Preliminary analyses consider deadweight, thermal expansion/contraction and seismic loads (either static or dynamic). For preliminary analyses, ASME Class 2 rules may be used for ASME Class 1 piping. Additionally, detailed stress analyses are performed for two representative systems, the RCS discharge line and the feedwater line. The RCS discharge piping is selected because it is representative of the ASME Class 1 piping with respect to deadweight, seismic, thermal transient and fatigue loading. The discharge line is longer than the other Class 1 lines, with more seismic supports and longer spans between restraints. Therefore, this analysis presents the more challenging analysis case. The feedwater line is selected because it experiences bounding loads for all Class 2 systems with respect to the leak-before-break (LBB) analysis. Detailed stress analyses include loads from deadweight, seismic (dynamic), thermal expansion/contraction, and for Class 1 lines, fatigue (including environmentally assisted fatigue evaluation in conformance to Regulatory Guide (RG) 1.207. The results of these analyses confirm the acceptability of the piping designs.

3.12.2 Codes and Standards

10 CFR 50, Appendix A, General Design Criterion (GDC) 1 requires that structures, systems, and components (SSC) must be designed to quality standards commensurate with the importance of the safety functions to be performed. GDC 2 requires that SSC be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, and floods without the loss of their safety function. GDC 4 requires that the nuclear power plant SSC be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). GDC 14 requires that reactor coolant pressure boundary of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and gross rupture. GDC 15 requires reactor coolant systems

(RCSs) and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design condition of normal operation, including anticipated operational occurrences.

Codes and standards used in the design of piping systems and piping supports are consistent with 10 CFR 50, Appendix A, GDCs 1, 2, 4, 14, 15, and 10 CFR 50 Appendix S as discussed in the following sections. The design codes for ASME Class 1, 2 and 3 piping systems are described below.

3.12.2.1 ASME Boiler and Pressure Vessel Code

The design code specified for ASME Code Section III Class 1, 2, and 3 piping in the NuScale design is in Reference 3.12-1. The conditions of use for ASME BPVC Section III (Reference 3.9-1) is applied in accordance with 10 CFR 50.55a (b)(1) as applicable to the 2013 Edition.

The portions of the Code which provide the design requirements for ASME Class 1, 2, 3 piping and supports are provided below:

- Class 1 piping is designed under the design requirements of BPVC Section III, Subarticle NB-3600.
- Class 2 Piping is designed under the design requirements of BPVC Section III, Subarticle NC-3600.
- Class 3 piping is designed under the design requirements of BPVC Section III, Subarticle ND-3600.
- Class 1, 2, 3 Piping supports are designed under the design requirements of BPVC Section III, Subarticle NF.

Note that there are some specific exceptions in the procedures of the Code which allow for analyzing components to other Section III subarticles in some circumstances (e.g. Class 2 components designed to Subarticle NB-3600).

Quality Group D (RG 1.26) piping is designed and analyzed to the latest edition of ASME B31.1 (Reference 3.12-3).

3.12.2.2 ASME Code Cases

ASME Code Cases may be used if they are either conditionally or unconditionally approved in Regulatory Guide (RG) 1.84, Revision 36.

3.12.2.3 Design Specification

Design specifications are required for ASME Code Class 1, 2, and 3 piping, piping components and associated supports per the ASME BPVC Section III. Additionally, conformance to these Design Specifications for the as-designed piping, piping components, and associated supports is required per the Code to be documented in Design Reports.

3.12.3 Piping Analysis Methods

3.12.3.1 Experimental Stress Analyses Methods

Experimental stress analysis methods are not used to qualify piping for the NuScale Power Plant design.

3.12.3.2 Modal Response Spectrum Method

The effects of the ground motion during a safe shutdown earthquake (SSE) event are transmitted through structures and components to the piping systems at support and anchorage locations. Seismic Category I piping systems are required to be designed to withstand the effects of the SSE and maintain the capability of performing their safety functions. A dynamic method of analysis used for piping systems is the response spectrum method. This analysis method applies in-structure response spectra (which are amplified from the fundamental seismic ground motion spectra) to the piping system in all three directions. The response spectra are determined from time-history motion of the structure applied through single-degree-of-freedom harmonic oscillators. The maximum response of the oscillators throughout the duration of the event, for a range of natural frequencies, is taken to be the response spectra curve.

The in-structure response spectra are applied to the locations where the piping system is attached to or supported by the structure, such as piping supports or vessel nozzles. In-structure response spectra of the NPM are determined using dynamic analysis of a a three-dimensional, finite element model of the reactor module structural system as described in NuScale Power Module Analysis Technical Report TR-0916-51502 (Reference 3.12-13) for piping that is attached to the NuScale Power Module (NPM). For piping which is attached to the building, the in-structure response spectra of the Reactor Building is used, which is described in Section 3.7. The response spectrum analysis is performed using either the uniform support motion (USM) method or the independent support motion (ISM) method.

Analysis using the response spectrum method is performed linearly by transforming the coupled equations of motion for a multiple degree-of-freedom system into a set of uncoupled modal response equations. The maximum modal responses are evaluated and combined using approximate rules to account for phasing of the modes. The combination of maximum modal responses is a generally conservative approach. The modal responses and spatial responses of the piping system are combined using the methods described below.

3.12.3.2.1 Development of In-structure Response Spectra

To perform the response spectrum analysis, an in-structure response spectra must be developed for the structures that support the piping system anchors. The methods and guidance in RG 1.122 Revision 1 are used to develop the in-structure response spectra.

The in-structure response spectra shall include accelerations for three orthogonal directions (two horizontal and one vertical) from the time history motions of the supporting structure. Uncertainties in the structural frequencies which represent uncertainty or approximations of material and structural properties are accounted for by smoothing and peak broadening the raw in-structure response spectra. The methods and guidance of RG 1.122 are used for smoothing and peak broadening the raw spectra. If the frequency broadening is not determined using the frequency dependent procedure in RG 1.122, then the response spectra are peak broadened by ±15 percent.

3.12.3.2.2 Uniform Support Motion

For piping systems which may be supported at multiple points within a structure the seismic motions of each support location may vary. An acceptable approach for analyzing these piping systems is to define a uniform response spectrum (URS) that envelops all of the individual response spectra at the various support locations. This method is referred to as the uniform support motion (USM) method. The methods and guidance of RG 1.92 is used for combining modal and spatial responses for USM method of analysis. Either Revision 3 or Revision 1 of RG 1.92 may be used for the NuScale design. Generally, piping for the NuScale design is analyzed using AutoPIPE software, which does not currently have the capability to separate periodic and rigid components of modal responses. If the software used for analysis does not have the capability to comply entirely with Revision 3 of RG 1.92, the Revision 1 may be used as long as the missing mass effects are also considered. Revision 3 states:

> "The methods of combining modal responses, described in Revision 1, remain acceptable. If however, applicants for new licenses choose to use Revision 1 methods for combining modal responses, their analyses should address the residual rigid response of the missing mass modes as discussed in Regulatory Positions C.1.4.1 and C.1.5.1 of this guide."

When performing analysis of piping systems by the USM method, damping values are be applied as permitted by RG 1.61 Revision 1. For analysis of NuScale piping systems, a single damping value of 4 percent for analysis of SSE loads (for all frequencies) is used as permitted by RG 1.61. Frequency dependent damping is not used, though it is conditionally permitted by RG 1.61. If the analysis of a piping model includes other non-piping components (such as supports or structural elements which have different damping values per RG 1.61) then composite modal damping values are determined using one of the two techniques as follows:

(1) Mass weighted damping: $$\bar{\beta}_j = \{\varphi\}^T[\bar{M}]\{\varphi\}$$ Eq. 3.12-1

or

(2) Stiffness weighted damping: $$\beta_j = \frac{\{\varphi\}^T[\bar{K}]\{\varphi\}}{K^*}$$ Eq. 3.12-2

where,

$$K^* = \{\varphi\}^T[K]\{\varphi\}$$

$[K]$ = assembled stiffness matrix,

$\bar{\beta}_j$ = equivalent modal damping ratio of the jth mode,

$[\bar{K}]$, $[\bar{M}]$ = the modified stiffness or mass matrix constructed from element matrices formed by the product of the damping ratio for the element and its stiffness or mass matrix, and

$\{\varphi\}$ = jth normalized modal vector.

Note that when composite modal damping is determined using these methods, the damping shall not exceed 20 percent of critical. Equations for determining system composite modal damping are provided in ASCE 4-98 (Reference 3.12-14).

3.12.3.2.3 Modal Combination

The individual modal responses of the piping system due to URS input are not simply summed at each location because it is unlikely that the maximum individual modal responses of all piping system supports would occur at the same time during a seismic event. Therefore, modal responses are combined using the methods of Regulatory Guide 1.92 to obtain the representative maximum response of interest from the maximum individual modal responses.

When performing response spectra analyses which comply with Revision 1 of Regulatory Guide 1.92, modal responses of the piping system are only considered below a defined cutoff frequency at which spectral accelerations approximately return to the zero period acceleration (ZPA). Above the ZPA frequency the system is considered to be rigid because the components are not significantly excited by the seismic ground or in-structure motion. However, nuclear power plant SSCs may have important natural vibration modes at frequencies higher than the ZPA frequency due to more rigidly restrained components or significant lumped masses near rigid restraints which are not considered in the low frequency modal analysis. Therefore, the contribution of mass associated with modes higher than the ZPA are accounted for as described in Section 3.2.3.2.6.

When performing response spectra analyses which comply with Revision 3 of Regulatory Guide 1.92, the system modal responses are considered to be periodic in the region of amplified spectral displacement, velocity, and acceleration (Regions AB, BC, and CD in Figure 1 of RG 1.92, Revision 3). In the transition region from amplified periodic spectral acceleration to rigid spectral acceleration (region DE in figure 1 of RG 1.92 Revision 3), the response consists of both periodic and rigid components. In the high-frequency regions (regions EF and FG in Figure 1 of RG 1.92, Revision 3 response is considered to be rigid. The combination of modal response components are treated differently in Regulatory Guide 1.92 Revision 3 depending on whether a given mode includes only periodic components, only rigid components, or both periodic and rigid components. Combining all the periodic and rigid response components in accordance with procedures of Regulatory Guide 1.92 Revision 3 for all modes provides the total system response to the URS.

3.12.3.2.4 Uniform Support Motion Periodic Modal Responses

When performing response spectrum analysis using the USM method, periodic modal responses shall be combined using the methods and guidance of Regulatory Guide 1.92 Revision 1 or Revision 3. If all the frequencies of the modes are sufficiently separated then the square root of the sum of the square (SRSS) method is used:

$$R = \left(\sum_{k=1}^{N} R_k^2 \right)^{\frac{1}{2}} \quad \text{Eq. 3.12-3}$$

where,

R = the representative maximum response due to the input component of the earthquake,

R_k = the peak response due to the kth mode, and

N = the number of significant modes.

The SRSS method is not applicable if closely spaced modes exist in which case an alternative method of combining modal responses is required. The criteria for defining closely spaced modes is provided by Regulatory Guide 1.92, Revision 3 and the determination is dependent on the critical damping ratio:

1) For critical damping ratios ≤ 2 percent, modes are considered closely spaced if the frequencies are within 10 percent of each other (i.e., for fi < fj, fj ≤ 1.1 fi). Where fi and fj are frequencies of adjacent modes.

2) For critical damping ratios >2 percent, modes are considered closely spaced if the frequencies are within five times the critical damping ratio of each other (i.e., for fi < fj and 5 percent damping, fj ≤ 1.25 fi; for fi < fj and 10 percent damping, fj ≤ 1.5 fi).

For a system which has closely spaced modes, the double sum methods are used to combine the periodic modal responses. These double sum equations include modal correlation coefficients which are uniquely defined, depending on the method chosen for evaluating the correlation coefficient.

The modal correlation coefficients are provided in the applicable revision of Regulatory Guide 1.92.

Absolute Doublesum - RG 1.92 Rev. 1: Equation (8)

$$R = \left(\sum_{k=1}^{N}\sum_{s=1}^{N}|R_k R_s|\varepsilon_{ks}\right)^{\frac{1}{2}} \quad \text{Eq. 3.12-4}$$

where,

R = the representative maximum response of the element due to an input component of the earthquake,

R_k = the peak response of the element due to the k^{th} mode,

R_s = the peak response of the element attributed to the s^{th} mode,

N = the number of significant modes, and

ε_{ks} = the modal correlation coefficient for modes k and s.

Signed Doublesum - RG 1.92 Rev. 3: Equation (1)

$$R_{pI} = \left(\sum_{i=1}^{N}\sum_{j=1}^{N}\varepsilon_{ij}R_{p_i}R_{p_j}\right)^{\frac{1}{2}} \quad \text{Eq. 3.12-5}$$

where,

R_{pI} = combined periodic response for the I^{th} component of seismic input motion (I = 1, 2, 3, for one vertical and two horizontal components),

N = the number of significant modes,

ε_{ij} = the modal correlation coefficient for modes *i* and *j*,

R_{p_i} = periodic response or periodic component of a response of mode *i*, and

R_{p_j} = periodic response or periodic component of a response of mode *j*.

3.12.3.2.5 Uniform Support Motion Rigid Components of Modal Response

In the transition region where modal responses consist of both periodic and rigid components, the response components can be separated by the methods in

Regulatory Guide 1.92 Revision 3. Once separated, these rigid components responses and residual rigid responses are combined algebraically.

3.12.3.2.6 Residual Rigid Response

The contribution of the "missing mass" of piping systems above the ZPA is accounted for using the method provided in Section C.1.4 of Regulatory Guide 1.92 Revision 3.

For the missing mass method, the modal responses are determined for those modes with natural frequencies less than the ZPA. Then for each degree-of-freedom included in the dynamic analysis, the fractions of degree-of-freedom mass included and not included in the summation of all modes are determined. Modes higher than the ZPA are assumed to respond in phase with the ZPA and with each other; therefore, they are combined algebraically and applied to all of the degree-of-freedom masses not included in the low frequency modal analysis (below the ZPA). Additional discussion of the calculation of the missing mass response is provided in Appendix A of Regulatory Guide 1.92 Revision 3.

An alternative approach to including the contribution of high-frequency modes is to use the Static ZPA method provided in Regulatory Guide 1.92 Revision 3.

When combining modal responses using Regulatory Guide 1.92, Revision 1, the residual rigid response of the missing mass modes is accounted for in accordance with Regulatory Positions C.1.4.1 and C.1.5.1 of Regulatory Guide 1.92, Revision 3. The residual rigid response is obtained using the missing mass method of Regulatory Position C.1.4.1. For each of the three components of seismic input motion, the residual rigid response and the modal response calculated with Revision 1 of Regulatory Guide 1.92 are combined using SRSS for the response spectrum method (RG 1.92 Revision 3, Regulatory Position C.1.5.1).

3.12.3.2.7 Uniform Support Motion Complete Inertial Response

The complete (periodic plus rigid) response spectrum analysis solution for each of the three orthogonal component motions (two horizontal and one vertical) is calculated using the methods in Regulatory Guide 1.92 Revision 3. Note that two complete solution methods are presented and either method may be used as long as the applicable required conditions are met. When combining modal responses in accordance with Revision 1 of Regulatory Guide 1.92, the “missing mass” associated with these rigid response components is required to be incorporated by the process described in Section 3.12.3.2.6.

3.12.3.2.8 Directional Combination

Once the complete inertial response is determined, the responses of piping system components due to the seismic inputs in all three orthogonal directions is obtained by SRSS combination method per RG 1.92 Revision 3.

3.12.3.2.9 Seismic Anchor Motion

In addition to dynamic inertia loads imparted to the piping system, the effects of piping anchor motion (displacement) shall also be considered. The maximum relative support displacements are obtained from the structural response calculations or, from the applicable in-structure response spectra.

Support displacements are imposed on the supported piping in the most unfavorable combination. For piping systems where all the support locations are within a single structure or on a single component, the seismic motions may be considered to be in-phase, and the relative displacement between supports may be neglected. However, where supports are located on different components or structures, or when the support motions may not be in-phase, the support motions are conservatively assumed to move out-of-phase when evaluating relative displacements between supported locations.

Analyses of piping systems due to seismic anchor motions are performed statically. The system response due to inertial effects and due to anchor motions are combined by the absolute sum method for USM analysis, when combining of the results is necessary. When performing analysis of inertial effects using the ISM method, the response due to anchor motions are combined with inertial effects by the SRSS method (NUREG-1061, Volume 4) (Reference 3.12-2).

3.12.3.3 Independent Support Motion Method

The USM method can result in considerable overestimation of seismic responses. For piping systems with multiple supports located in a single structure or which are supported by more than one structure, the independent support motion (ISM) method is used as an alternate method. This method is described in NUREG-1061, Volume 4 (Reference 3.12-2). This method of analysis is performed by grouping piping supports (such as supports attached to the same portion of a structure) and applying a single response spectrum to each group. One group of supports is moved at a time using the input response spectrum specified for those supports, with all other groups being stationary.

When performing the ISM method of analysis, modal responses and spatial components are combined using the methods and guidance of NUREG-1061, Volume 4 (Reference 3.12-2). For each mode and direction, seismic responses from the individual grouped analyses are combined by absolute summation as recommended by NUREG-1061, Volume 4. Then spatial (directional) and modal component responses of the piping system are combined, respectively. Damping values from Regulatory Guide 1.61 Revision 1 shall be used when performing analysis using the ISM method. See Section 3.12.3.2.2 for discussion of appropriate damping.

If the ISM method is used for piping systems supported at multiple locations on the NPM, the criteria presented in NUREG-1061, Volume 4 is followed.

3.12.3.4 Time-History Method

Seismic analysis of piping systems may also be performed using the time history method (as opposed to the response spectrum method or the equivalent static method). The time history method can provide more realistic results for multiply-supported systems but it requires increased analytical effort. Therefore, the time history method of analysis for seismic input is generally reserved for major components. Analysis of piping system response or component response due to other transient loads such as water hammer, steam hammer, and impingement may also be performed using the time history method (see Section 3.12.5.3).

Time history analysis can be performed by direct integration of the coupled equations of motion or by modal superposition. When the time history method is used to analyze the seismic response of NuScale piping systems the modal superposition method is used. The modal superposition method is performed by decoupling the multiple degree-of-freedom equations of motion by changing the equations of motion from normal (displacement) coordinates to modal coordinates. The equations are solved linearly as single degree-of-freedom equations and then the results for all modes are combined at each time step. Regulatory Guide 1.92 Revision 1 provides acceptable procedures for combining modal responses. Regulatory Guide 1.92 Revision 3 provides acceptable procedures for combining periodic and rigid modal responses of piping components and for including “missing mass” contribution (above the ZPA frequency). Contribution of mass above the ZPA frequency is included in modal superposition time history analyses as described in Regulatory Position C.1.4 of Regulatory Guide 1.92 Revision 3. As stated in C.1.4.1 of RG1.92 Revision 3, the missing mass contribution, scaled to the instantaneous acceleration, is algebraically summed with the transient solution at the corresponding time.

Time step sensitivity evaluations are performed for piping systems which are analyzed by the time history method to show that the selected time step provides acceptable convergence.

Damping of piping systems using the time history method of analysis is per Section 3.12.3.2.2.

For time history analysis where the three components of earthquake motion are calculated separately, the representative maximum response of a piping component can be determined by taking the SRSS of the maximum responses for each of the three spatial components. As stated in Regulatory Position C.2.2 of RG 1.92 Revision 3, if the three components of the earth motion are statistically independent, the maximum response of a piping component can be obtained from algebraic summation of the three component responses at each time step. Alternatively, if the three components of input motion are statistically independent, a single time history analysis may be performed with all three components of earthquake motion applied simultaneously; this effectively achieves algebraic summation.

3.12.3.5 Damping Values

When performing analysis of piping systems by the USM method, damping values are applied per RG 1.61. For analysis of NuScale piping systems, the single damping value of 4 percent (for all frequencies) is used in accordance with RG 1.61. Frequency dependent damping is not used, though it is conditionally permitted by RG 1.61. If the analysis of a piping model includes other non-piping components (such as supports or structural elements which have different damping values per RG 1.61), then composite modal damping values are determined using the techniques discussed in Section 3.12.3.2.2.

3.12.3.6 Inelastic Analysis Method

Inelastic analysis methods are not used for any NuScale piping system analysis.

3.12.3.7 Equivalent Static Load Method

One method of analyzing seismic effects on a piping system is to use an equivalent static load method. This is a simplified analysis method in which a constant acceleration force is applied to lumped masses of piping system components at their center of gravity locations. Bounding seismic acceleration values are determined for each direction based on the dynamic properties of the system. The static acceleration values are applied to the piping system model in each of the three orthogonal directions to obtain the response forces and resulting component stresses. Analyses results from each of the three directions are combined by the SRSS method.

Linear equivalent static analyses of seismic loads on piping systems is discussed in Section 3.7.3

3.12.3.8 Non-seismic/Seismic Interaction (II/I)

The majority of the Seismic Category I piping is located inside of, or outside of and attached to, the containment vessel. All piping, vessels, components and structures inside containment are Seismic Category I. Therefore, the NuScale Power Plant has few Seismic II/I considerations.

For those few cases, such as on the top of the containment where it is not possible or practical to isolate the seismic piping, non-seismic piping which is located in proximity to the seismic Category I piping is classified as seismic Category II and is analyzed and qualified to the same seismic criteria as the seismic Category I piping thereby precluding adverse interaction during the SSE.

The dynamic effects of non-seismic piping which is attached to seismic Category I piping are accounted for by including some portion of the connected non-seismic piping (and supports) in the model of the Category I piping. The non-seismic piping attached to seismic Category I piping is designed such that the adverse interaction during the SSE is precluded. The attached non-Category I piping, up to the first anchor beyond the interface is designed not to cause a failure of the Category I piping during the SSE.

Interaction Evaluation

Non-seismic piping and components may be located in proximity to safety-related piping without being classified as seismic Category II if an interaction evaluation is performed to verify that no adverse interaction with the functionality of seismic Category I components will occur due to the failure of the non-seismic piping during seismic events. Non-seismic piping components are assumed to fail by being put into a freefall condition, and interactions with safety-related components are evaluated based on their relative locations.

For non-seismic piping systems, all non-seismic supports are assumed to fail and the flanged connections are also assumed to fail. Non-seismic piping which is welded is assumed to fail at rigid constraint locations. These assumptions for interaction evaluations are made to give the most bounding interaction effects.

3.12.3.9 Seismic Category I Buried Piping

The NuScale design does not include any ASME Code Class 1, 2, or 3 piping which is directly buried in soil.

Note, that for the NuScale design the only ASME BPVC Section III Class 1, 2, or 3 piping that is not directly connected or within an NPM is the ASME Class 3 assured makeup line to the reactor pool which provides a means to add inventory to the pool via temporary equipment for long term beyond design basis event support.

If a licensee desires this line to be directly buried in soil, additional analysis methodologies are required to be provided because the ASME BPVC Section III does not address all applicable loads of buried piping.

3.12.4 Piping Modeling Technique

3.12.4.1 Computer Codes

The computer codes ANSYS and AutoPIPE are used for the analyses of ASME Code Class 1, 2 and 3, and ASME B31.1 piping.

ANSYS

The computer program ANSYS is used for the design and analysis of NuScale piping systems. This program is used for analysis of piping for applied static loads and for dynamic loads. The dynamic analyses required for seismic evaluations such as response spectrum analysis and time history analysis are performed using ANSYS.

ANSYS is developed by ANSYS Corporation and maintained by NuScale. ANSYS includes pipe elements which have been verified and validated to Nuclear Regulatory Commission (NRC) standards (such as NUREG/CR-1677) (Reference 3.12-16). Additionally, ANSYS is used if a detailed stress analysis (i.e., NB-3200) is performed in lieu of a NB/NC/ND-3600 piping analysis.

AutoPIPE

The computer program AutoPIPE is used for the design and analysis of NuScale piping systems. AutoPIPE is used for analysis of piping due to static loads and for dynamic loads. AutoPIPE also performs design checks for ASME Code Class 1, 2, and 3 and ASME

B31.1 piping. The dynamic analyses required for seismic evaluations such as response spectrum analysis and time history analysis are performed using AutoPIPE.

AutoPIPE is developed by Bentley and maintained by NuScale, and has been verified and validated to NRC standards (such as NUREG/CR-1677).

3.12.4.2 Dynamic Piping Model

Analytical piping system models are constructed in computer programs to define the masses, geometries, and constraints required to perform the required analyses. These system models are assembled in a three dimensional coordinate system using finite elements. The elements used for piping system models include elastic pipe and beam elements which have stiffness properties which represent equivalent pipe geometry or other piping components. Lumped masses are used at locations of piping components such as valves and flanges. The finite elements are connected at nodes within the model. Nodes are located at structural discontinuities (such as tees, lumped masses, supports locations, nozzle connections etc.) or other locations of interest. Piping supports can be modeled as beam elements or as simple springs with appropriate stiffness values in the constrained directions.

Piping system mass such as the pipe, pipe contents, and insulation weight are modeled as distributed mass. If some of these masses cannot be modeled as distributed then they are modeled by using multiple smaller elements with appropriately divided lumped masses. However, lumped mass spacing are not to be exceeded one half of the length that would produce a natural frequency equal to the ZPA frequency of the seismic input for an equivalent simply supported beam. This ensures that the piping system response remains representative during dynamic analyses.

Torsional effects of eccentric masses (such as a valve operator) are accounted for in the modeling of piping systems if determined to be significant on a case by case basis.

Rigid components of piping systems (natural frequencies above the ZPA frequency) are included in the piping model by placing lumped masses which are rigidly linked to the piping, with the lumped masses located coincident to the centers of gravity of these components. Any integral shield restraints (ISRs) on piping are included in the model as lumped masses. Flexible components are included by using beam elements and lumped masses to maintain representative dynamic response.

The mass of a piping support is included in the piping model (as opposed to a simple constrained boundary condition) if the support mass is greater than 10 percent of the total mass of the supported piping span. The subject span is defined as all piping components to the next support location in both directions from the support considered for inclusion in the model.

3.12.4.3 Piping Benchmark Program

AutoPIPE and ANSYS Comply with NRC benchmarks as described in Section 3.12.4.1.

COL Item 3.12-1: A COL applicant that references the NuScale Power Plant design certification may use a piping analysis program other than the programs listed in Section 3.12.4.1;

however, the applicant will implement a benchmark program using the models for NuScale Power Plant standard design.

3.12.4.4 Decoupling Criteria

Decoupling Criteria

The NuScale reactor design is compact, such that there is not a large amount of ASME Class 1, 2 and 3 piping associated with each NPM. Therefore, the piping runs are relatively short and analytical models are generally terminated at structural anchors, which effectively isolate the system from additional static and dynamic effects from beyond the anchor. These structural anchors are typically vessel nozzles, but may also be pipe supports which restrain all six degrees of freedom. It is allowed to terminate the analytical model of a piping system at a location without a structural anchor if decoupling criteria is satisfied or if adjacent analytical models sufficiently overlap. These methods are discussed below.

All of the ASME Class 1 piping is located inside of the containment vessel, with the exception that the welds connecting the containment isolation valves to the nozzles outside containment are also Class 1 for lines which contain primary coolant. ASME Class 2 piping is located both inside containment and outside containment, but entirely within the NPM disconnect flanges (i.e., does not extend to the reactor building. Therefore, generally, Class 1 or Class 2 piping runs are completely modeled between anchors and do not require decoupling or any other model termination method. There is a small amount of ASME Class 3 piping connected to each NPM which extends to the Reactor Building. All Class 3 piping is small diameter (NPS 2 or smaller); therefore, decoupling or overlap methods are used for small branch lines, such as instrument lines.

An analytical model may be terminated at a location where the structural interaction between adjacent segments of piping is limited and can be sufficiently accounted for using standard methods. This approach may be used at locations where there is a significant change in pipe size, such as branch lines of larger piping. Branch lines (such as instrument lines) which are smaller than the main run of the analyzed piping may be excluded from the analysis if it is sufficiently small compared to the run pipe. Branch lines may be excluded from the run piping analysis if the nominal diameter of the branch is less than or equal to 1/3 of the run nominal diameter or if the moment of inertia of the branch line is less than or equal to 1/25 of the run pipe. These criteria ensure that the effects of the smaller decoupled line on the larger piping can generally be considered negligible. However, any stress intensification factors and stress indices associated with the connection of the smaller line are considered in the analysis of the larger piping. Also, additional mass of the branch are considered for inclusion in the model of the larger piping to account for the decoupled line. When included, the added mass is at least half of the mass of the portion of the decoupled line up to the nearest support. Decoupling is not permitted if there is a relatively large mass (e.g. large valve or fitting) on the branch line in the span between the connection to the larger pipe and the nearest support.

Decoupling of portions of piping systems meets the following criteria from SRP 3.7.2 to ensure negligible error of the system dynamic response due to the decoupling. The mass ratios of the decoupled systems and subsystems, R_m, and the frequency ratio, R_f, are used as criteria to ensure negligible interaction:

R_m = Total mass of supported subsystem / Total mass of the supporting system

R_f = Fundamental freq. of supported subsystem / Dominant freq. of the support system

The following criteria are acceptable:

i. If $R_m < 0.01$, decoupling can be done for any R_f

ii. If $0.01 \leq R_m \leq 0.1$, decoupling can be done if $0.8 \geq R_f \geq 1.25$

iii. If $R_m > 0.1$, a subsystem model should be included in the primary system model

A separate analysis of the smaller decoupled piping may still be required where the dynamic load inputs (e.g., at a branch connection point) are determined from the larger run piping analysis. The connection of the smaller line to the larger pipe is modeled as an anchor in the analysis of the smaller line, with any associated stress intensification factors and stress indices applied. Static displacements of the larger piping, including those due to weight, thermal expansion and contraction, and seismic loads, are applied at the connection. If the larger piping is determined to be rigid (i.e. the fundamental frequency is above the cutoff frequency), it is acceptable to apply response spectra at the connection which envelopes those of the nearest supports on both the larger piping and the decoupled line. If the larger piping is not determined to be rigid, the inertial seismic loads (e.g. time histories, response spectra) for the decoupled line shall be generated from analysis of the larger piping, in order to account for amplification of the loads.

Overlap Region Methodology

It is always preferred to model an entire piping system with relevant connections and supports included in the same analysis. If it is not feasible to analyze a piping system as a single model then the overlap region methodology and conditions for the overlap region provided in NUREG/CR-1980 (Reference 3.12-11) are used. As discussed above, the NuScale Power Plant is compact and no Class 1 or Class 2 piping have overlapping models. A limited amount of Class 3 or B31.1 piping may have overlapping regions.

In accordance with the recommendations of NUREG/CR-1980 (Reference 3.12-11) the region of overlap is selected to be in a rigid (or relatively rigid) portion of the piping system per Section 2 (conclusions and recommendations) of NUREG/CR-1980 for specific analysis criteria, including the required stiffness of the overlap region.

All piping system analyses which include the overlap region are required to show acceptable results for the piping components and supports in the overlap region.

3.12.5 Piping Stress Analysis Criteria

3.12.5.1 Seismic Input Envelope Versus Site-Specific Spectra

The standard plant piping is evaluated using the certified seismic design response spectra (CSDRS) and the high frequency certified seismic design response spectra (CSDRS-HF) described in Section 3.7.1.1.

The floor response spectra are described in detail in Section 3.7.2.5.

COL Item 3.12-2: A COL applicant that references the NuScale Power Plant design certification will confirm that the site-specific seismic response is within the parameters specified in Section 3.7. A COL applicant may perform a site-specific piping stress analysis in accordance with the methodologies described in this section, as appropriate.

3.12.5.2 Design Transients

The piping systems design considers the design transients as discussed in Section 3.9.1.

3.12.5.3 Loadings and Load Combinations

Pressure

The design differential pressure between the inside and outside of the piping pressure boundary components (P_{des}) is used for the analysis of all ASME Code Class 1, 2, and 3 piping, as well as for ASME B31.1 piping. The minimum required piping wall thickness for ASME Code Class 1, 2, and 3 piping is calculated using NB-3640, NC-3640, and ND3640 at the design pressure using material properties from ASME Section II (Reference 3.12-17) Part D at the applicable design temperature. The design pressure of piping systems includes allowances for pressure addition sources (such as pumps), pressure surges, control system error, and system configuration effects such as static pressure heads. Design pressures (P_{des}) and maximum service pressures (P) are used in load combinations as noted in Table 3.12-1 and Table 3.12-2 for calculating stresses considering the condition and service level.

Deadweight

The deadweight of the piping system components is calculated by applying the standard acceleration due to gravity (1g) to the mass of the pipe, the pipe contents, any insulation, and other piping components.

Thermal Expansion

The loads on piping components and supports due to restrained thermal expansions and contractions (TE) are considered in the design and analysis of piping systems. Thermal loads appropriate to the mode of operation being analyzed are applied.

The anchors of piping systems may also be subject to thermal expansion, such as thermal anchor motions of equipment nozzles (such as those of the RPV and CNV), support/restraints, and run piping for decoupled branch lines. Thermal anchor motions less than or equal to 1/16th inch are excluded from consideration; this distance represents an industry standard for acceptable gaps in pipe supports upon installation. For decoupled branch lines, Thermal anchor motions are obtained from the applicable analysis of the run pipe.

The reference temperature for thermal analysis of piping systems is taken as 70 degrees Fahrenheit. At this reference temperature thermal loads are zero. For ASME Code Class 2 and 3 piping systems with an operating temperature of 150 degrees Fahrenheit or less, thermal analysis is not required except when required due to interface with other piping.

Buoyancy

Buoyancy loads (B) are used for piping that is submerged during an applicable load case. Buoyancy is calculated based on the weight of the water displaced.

Seismic

The analyses of ASME Code Class 1, 2 and 3 piping systems and other seismic Category I piping systems include the loads from inertial accelerations and seismic anchor motions (SAMs) due to the seismic ground motions associated with the SSE. All SAMs greater than 1/16th inch are included. Seismic effects are included in piping analyses as Service Level D loads. The applicable in-structure amplifications are used for piping systems supported by other structures and components (such as the reactor building or reactor module).

For the NuScale plant, the operating basis earthquake (OBE) is defined as 1/3 of the SSE. Any operating reactors will be shut down in the event of an earthquake which exceeds the OBE and checks for damages will be performed by operators. Due to the selection of the OBE as 1/3 of the SSE, the OBE effects are not included as design loads (as allowed by 10 CFR 50 Appendix S), but the OBE cyclic effects are included in fatigue evaluations of ASME Code Class 1 piping.

Relief Valve Thrust

Reaction loads are imparted onto piping system components when relief valves are actuated open. The loads depend on the valve size, valve capacity, the fluid properties, and the valve opening time. For the NuScale design, loads are considered for actuation of reactor safety valves and for actuation of ECCS valves.

Guidance for the design and analysis of safety valve installations is provided in ASME Section III (Reference 3.12-1) Code nonmandatory Appendix O. The analysis of these loads is discussed further in Section 2.12.5.9.

Water and Steam Hammer

Pressures waves are created when the flow of fluid in a piping system is abruptly altered. This can be initiated by mechanisms such as rapid valve actuation, pumps starting, or the collapsing of steam voids. If water or steam hammer loads are credible and significant for a piping system or portion of piping, they are included in the analysis. Thermal-hydraulic modeling software such as RELAP5 or AFT Impulse are used to determine water and steam hammer loads.

Wind, Hurricane, Tornado Loads

There is no ASME Code Class 1 and 2 piping in the NuScale design that is routed in areas that are exposed to wind, hurricane, or tornado loads. If any ASME Code Class 3 piping is routed in locations exposed to wind, hurricanes or tornadoes, the design basis wind loads for the plant (i.e. the design conditions used for the buildings) are included. Should a COL applicant that references the NuScale Power Plant design certification find it necessary to route Class 1, 2, and 3 piping not included in the NuScale Power Plant design certification so that it is exposed to wind, hurricanes, or tornadoes, it must be designed to the plant design basis loads for these events.

Design Basis Pipe Break Loads

The loads due to design basis pipe breaks (DBPB) are included in the analysis of ASME Class 1, 2 and 3 piping for the appropriate service conditions. Loads are imparted onto piping system components in the form of pipe whip, jet impingement, elevated temperatures, and hydraulic dynamic effects. Breaks in the main steam and feedwater lines (inside containment) meet the leak-before-break criteria of NUREG 0800 Section 3.6.3 as discussed in Section 3.6, and therefore, pipe breaks of these lines are not postulated. However, DBPB loads do include the impact of small break loss-of-coolant accident (LOCA), main steam, and feedwater line breaks outside the leak-before-break analyzed zone.

Thermal and Pressure Transient Loads

Thermal and pressure transient loads are included for the analysis of ASME Code Class 1 piping. For ASME Code Class 1 piping, these transient loads are included as Service Level A and B loads and their effects are determined by calculating the primary plus secondary stress intensity ranges as the piping system goes from one load set (such as pressure, temperature, moment, and force loading) to any other load set which follows it in time.

For ASME Code Class 2 and 3 piping, transient loads are also considered in the analyses by using the bounding pressure and temperature ranges in individual load combination cases.

The design and analysis of ASME Code Class 1, 2, and 3 piping systems use the applicable design transients addressed in Section 3.9.1

Hydrotests

All piping systems are subject to hydrostatic testing at a pressure higher than the design pressure upon initial assembly of the piping system. The hydrostatic test loads are included for analysis for applicable load cases. The additional weight of the test fluid is considered for the total load of the hydrostatic test (e.g. if the normal service fluid is gas but the test fluid is liquid).

Load Combinations

Using the methodology and equations from the ASME Section III code (Reference 3.12-1), pipe stresses are calculated for various load combinations. The ASME Code includes design limits for Service Levels A, B, C, and D, and testing. Load combinations for ASME Code Class 1 piping are given in Table 3.12-1. Class 2 and 3 load combinations are given in Table 3.12-2.

3.12.5.4 Combination of Modal Responses

The modal combination methods used in response spectrum analyses for piping are addressed in Section 3.12.3.2.

3.12.5.5 Fatigue Evaluation of ASME Code Class 1 Piping

ASME Code Class 1 piping systems and piping components are analyzed for fatigue effects due to cyclic loads. These cyclic loads include all applicable thermal transients, hydraulic transients, and external loads such as seismic. Analysis is performed in accordance with the methods and requirements of ASME Code Section III NB-3650.

Additionally, the fatigue analysis of ASME Code Class 1 components incorporate the effects of the light-water reactor environment in accordance with the requirements of Regulatory Guide 1.207 and NUREG/CR-6909.

For fatigue analysis of ASME Code Class 1 piping components, the seismic load includes a minimum of one SSE and five OBE events in accordance with the guidance of SRP 3.7.3. The number of cycles per earthquake can be obtained from the time history used for the system analysis, or a simplified approach is permitted in which a minimum of ten maximum stress cycles per earthquake is used. Alternatively, an equivalent load is considered to be two SSE events, each with ten maximum stress cycles (total of twenty cycles) or the number of fractional vibratory cycles may be used (but with an amplitude not less than 1/3 of the maximum SSE amplitude) when derived in accordance with Annex D of IEEE Std-344 (Reference 3.12-5). When this method is used, and if the amplitude of the vibration is taken as 1/3 of the amplitude of the SSE, then 312 fractional amplitude SSE cycles are considered.

3.12.5.6 Fatigue Evaluation of ASME Code Class 2 and 3 Piping

Design and analysis of ASME Code Class 2 and 3 piping systems and piping components considers fatigue effects if they are subject to a total number of equivalent full temperature cycles greater than 7000 per NC-3611.2. Instead of analyzing ASME Code Class 2 and 3 components for specific cyclic loads (as for Class 1 components using cumulative usage factors), the fatigue effects are addressed by applying stress range reduction factors as provided in NC/ND-3611.2(e) to the allowable stress range for thermal expansion stresses.

3.12.5.7 Thermal Oscillations in Piping Connected to the Reactor Coolant System

The piping sections that can not be isolated and are connected to the reactor coolant system (RCS) can experience temperature stratification and oscillation due to mixing with stagnant lower temperature fluid with the higher temperature fluid at the connection interface (due to turbulent penetrating flow or leakage past an isolation component). These thermal conditions add fatigue loads to piping components due to constrained thermal deflections that must be accounted for by analysis or they can be precluded by design. Thermal oscillations in RCS connected piping were determined to be the cause of pressure boundary component failures at multiple operating nuclear plants as described in NRC Bulletin 88-08 including supplements. Therefore, unisolable sections of piping connected to the RCS of the NuScale design are evaluated for susceptibility to temperature oscillations which may affect the integrity of the components.

The screening criteria and evaluation methodology of Electric Power and Research Institute (EPRI) technical report TR-103581 (Reference 3.12-7) is used to assess unisolable piping connected to the RCS for thermal oscillations in the NuScale design. EPRI TR-1011955 (Reference 3.12-6) is also used as a supplement in the screening process.

For thermal stratification to occur in unisolable piping connected to the RCS which could impose additional fatigue loads on pressure boundary components, the following conditions must exist:

- An isolation component (e.g. a valve) exists in the design with the potential for leakage, which separates stagnant, colder fluid from the RCS. In this configuration a pressure differential must also exist across the isolation component to drive flow through a potential leakage path.
- An unisolable section of stagnant branch piping connected to the RCS that is oriented horizontally or oriented vertically which then transitions to a horizontal run within the span of turbulent RCS penetration (from the point of interface between the branch and the RCS)

Additional fatigue loads are imposed on components when a mechanism exists to promote cycling of the stratified conditions. Depending on the mechanism a large number of fatigue load cycles can be imposed on piping components over their service life. Mechanisms for thermal cycling can be intermittent leaking valves or varying turbulent penetration flow due to changes in RCS velocity in the region of unisolable branch connections.

The following screening criteria are used to evaluate if piping in the NuScale design is susceptible to thermal stratification or cycling:

- Pipe lines, or parts of lines less than or equal to 1 NPS are not susceptible to significant thermal stratification/cycling loads because flow in small lines tends to mix more rapidly than in larger pipe and because small pipe is generally more flexible which reduces the load effects.
- Piping that is connected to the RCS which is vertically oriented or greater than 45 degrees to horizontal for a significant distance is not susceptible to significant thermal loadings due to stratification.
- If sufficient continuous flow is present during normal operations through the line connected to the RCS then thermally stratified conditions are precluded.

The following piping systems connected to the RCS in the NuScale design and are evaluated:

- chemical and volume control system RCS discharge piping
- chemical and volume control system RCS injection piping
- pressurizer spray lines
- reactor pressure vessel high point degasification piping
- emergency core cooling system (ECCS) hydraulic lines

The RCS discharge line, RCS injection line, and pressurizer spray lines are not stagnant during power operations, therefore these lines are not susceptible to the adverse thermal stratification or oscillations. The RPV high point degasification line is not filled with liquid, therefore this line is also not susceptible. The ECCS lines are normally stagnant and have horizontal portions but they are smaller than NPS 1 and therefore are not considered to be susceptible to failure from thermal stratification or cycling.

3.12.5.8 Thermal Stratification

Thermal Stratification is discussed in Section 3.12.5.8.1 through Section 3.12.5.8.3.

3.12.5.8.1 Pressurizer Surge Line Stratification

NRC Bulletin 88-11 was issued in response to a condition in an operating plant in which the measured pressurizer surge line deflections did not reflect analysis results. The bulletin requested that operating PWRs examine pressurizer surge lines, evaluate for thermal stratification conditions, and perform additional analysis to account for these additional loads on surge line components. Additionally, applicants for PWR operating licenses were requested to demonstrate that surge line components meet applicable design codes and FSAR commitments with consideration of loads caused by thermal stratification. The NuScale Power Plant design does not have a pressurizer surge line. Therefore, NRC Bulletin 88-11 is not applicable.

3.12.5.8.2 Spray Line Stratification

The portions of the spray lines that are Class 1 are primarily in a vertical orientation which reduces the susceptibility to thermal stratification. Additionally, a small, constant flow of spray bypass normally precludes stagnant fluid in these lines. A regenerative heat exchanger provides heating of the spray fluid which reduces the temperature differential between spray fluid and the pressurizer.

3.12.5.8.3 Feedwater Line Stratification

NRC Bulletin 79-13 was issued in response to a condition in an operating plant in which cracking in feedwater lines (in feedwater elbows adjacent to steam generator nozzles) resulted in leakage inside containment and the subsequent inspections resulted in discovery of cracks in the feedwater lines of several nuclear power plants. Cyclic thermal gradients occurring during zero and low power operations was determined to be a primary contributing factor to the development of cracks in these lines.

The NuScale Power Plant feedwater lines are designed to minimize adverse loading due to thermal stratification. The steam generator feedwater nozzles (located on the feedwater inlet plenums) and the adjacent feedwater lines are either vertical or angled downward from the horizontal to minimize thermal stratification load.

3.12.5.9 Safety Relief Valve Design, Installation, and Testing

The design of safety valves and relief valves for the overpressure protection of ASME Class 1, 2, and 3 components considers the recommendations of the ASME Code (Reference 3.12-1) Nonmandatory Appendix O. Appendix O of the ASME Code includes valve arrangement considerations as well as guidance for determining loads required to be included in the analysis as a result of valve actuation. Appendix O categorizes pressure relief device installations in two configurations; closed discharge systems and open discharge systems. Closed discharge systems are relief devices that discharge into a distant location through a pipe connected directly to the relief valve, and open discharge systems are relief devices that discharge to atmospheric conditions.

For NuScale, relief valves which discharge into containment are considered to be an open discharge system configuration. Open discharge systems are analyzed with applicable reaction forces including the effects of the suddenly applied load. This is achieved by static methods using a dynamic load factor or by modeling the system and performing a dynamic analysis.

The acceptance criteria of SRP 3.9.3 are included in the design and analysis of ASME Code Class 1, 2, and 3 pressure relief devices:

- Load combinations include the most severe combination of the applicable loads due to internal fluid weight, momentum and pressure, dead weight of valves and piping, thermal load under heatup, steady state and transient valve operation, reaction forces when valves are discharging, and seismic forces.
- The contribution from reaction forces and moments are included by use of static analysis with a dynamic load factor or by using the maximum instantaneous values of forces and moment for each location as determined by dynamic system analysis. A dynamic load factor of 2.0 is used or guidance provided in ASME B31.1 (Reference 3.12-3) Nonmandatory Appendix II is used to calculate an appropriate dynamic load factor.
- Where more than one relief valve or safety valve is installed to protect the same pressure boundary, the sequence of valve openings which induce the maximum instantaneous value of stress at each location is used for loading at that location.
- Stresses are evaluated and applicable stress limits satisfied for all components of the pipe run and connecting systems for which safety/relief valves are installed.

Closed discharge configurations are not statically analyzed using dynamic load factors. These configurations are analyzed for forces on piping components in the discharge flow path of the relief device during the initial time period of the transient. These loads are determined similarly to water hammer and steam hammer events.

Load combinations and stress criteria are provided in Table 3.12-1 for ASME Code Class 1 and Table 3.12-2 for Class 2, and 3 piping. For the NuScale design, loads from ASME Code Class 1, 2, and 3 pressure relief devices (such as the reactor safety valves and the emergency core cooling system vent valves) are considered, although they may not be mounted on piping systems, because the discharge fluid interacts with other piping inside containment.

3.12.5.10 Functional Capability

10 CFR 50, GDC 2 requires, in part, that all components essential for safe shutdown of the plant shall be designed to withstand the effects of natural phenomena with appropriate combinations of normal and accident conditions. As stated in NUREG-1367, the function of a piping system is to convey fluid from one location to another, therefore the functional capacity of piping systems might be lost if sufficient deformation is sustained, even if pressure boundary integrity is maintained. NUREG-1367 concludes that piping system functional capability is maintained for all Service Level D loading conditions provided that:

1) Dynamic loads are reversing. This includes loads due to earthquakes, building-filtered loads such as those due to vibration of buildings caused by relief-valve actuation in boiling-water reactors, and pressure wave loads (not slug-flow fluid hammer).

2) Dynamic moments are calculated using an elastic response spectrum analysis with +/-15 percent peak broadening and with not more than 5 percent damping.

3) Steady-state (e.g., weight) stresses do not exceed 0.25 *Sy*.

4) D_o/t does not exceed 50.

5) External pressure does not exceed internal pressure.

Note: *Sy* is yield strength of material, D_o is pipe outside diameter, and t is wall thickness as discussed in NUREG-1367.

These requirements are invoked for Service Level D plant events for ASME Class 1, 2, and 3 piping which is required to transfer fluid during those events.

Alternatively, functional capability can be shown by meeting Service Level B stress limit/acceptance criteria for Service Level D loads.

3.12.5.11 Combination of Inertial and Seismic Anchor Motion Effects

The design of Seismic Category I piping includes both inertial and anchor movement effects caused by an SSE. The design of Seismic Category I piping and supports includes analysis of the inertial and anchor movement effects of the SSE event. Discussion of seismic anchor motion effects is provided in Section 3.12.3.2.9

3.12.5.12 Operating Basis Earthquake as a Design Load

As noted in Section 3.7, the ground motion of the OBE for the NuScale Power Plant design is equal to one-third of the ground motion of the SSE. Therefore, the OBE is not used as a design load for the Nuscale Plant. However, the cyclic effects of the OBE are conservatively considered in the fatigue analysis for Class 1 piping. Section 3.7.4 notes that, in the event of an earthquake which meets or exceeds the OBE ground motion, plant shutdown is required and requires the COL applicant to have a seismic monitoring system and a seismic monitoring program to inspect designated SSC for functional damage.

3.12.5.13 Welded Attachments

For ASME Class 1 piping, no welded attachments to the piping are permitted for support or restraint of the piping due to design and service loads. Welded attachments for ASME Class 2 and 3 piping and for Class 1 piping for other functions not associated with maintaining structural integrity of the piping pressure boundary (e.g., whip/rupture restraint) are permitted provided the effects of the attachment on the piping are considered in accordance with ASME Code, Section III Nonmandatory Appendix Y.

3.12.5.14 Minimum Temperature for Thermal Analyses

No thermal analysis is required for piping systems with an operating temperature equal to or less than 150 degrees Fahrenheit.

3.12.5.15 Intersystem Loss-of-Coolant Accident

Piping systems that normally operate at low pressure that interface with the RCS and are subjected to the full RCS pressure are designed for the design pressure of the RCS.

3.12.5.16 Effects of Environment on Fatigue Design

In accordance with the methodology described in RG 1.207, the effects of reactor coolant environment are considered when performing fatigue analyses for Class 1 piping and components.

3.12.6 Piping Support Design Criteria

3.12.6.1 Applicable Codes

Piping supports of ASME Code Class 1, 2, and 3 piping are classified to the same Class 1, 2, or 3 classification as the piping they support. These supports are designed, manufactured, tested, and installed to the requirements of ASME Code, Section III, Subsection NF. ASME Code Class 1, 2, and 3 supports are designed and analyzed for Design and Service Levels A, B, C and D and Test conditions. When analyzing supports for Service Level D loads, criteria of Appendix F of the ASME Code is used. For Class 1 linear-type and plate-and-shell type supports, the additional stress limit criteria of Regulatory Guide 1.124 Revision 3 and Regulatory Guide 1.130 Revision 3 also are met.

Subsection NF of the ASME Code categorizes piping supports into three types, and specific requirements are provided for each type of support. The three types of supports are described as plate and shell type, linear type, and standard supports. Plate and shell type supports are fabricated from plate and shell elements (such as a skirt or saddle) and are normally subject to a biaxial stress field (NF-1212). A linear type support is defined as acting under essentially a single component of direct stress, but may also be subject to shear stresses. Examples of linear type supports are tension/compression struts, beams subject to bending, trusses, frames, rings, arches, and cables (NF-1213). Standard supports are typified by the supports described in MSS SP-58 (Reference 3.12-15) which consist of standard catalog parts (Figure NF-1214-1). Standard support capacities may be determined by load rating procedures (e.g. NF-3280), plate and shell analysis, or by linear analysis.

Standard supports for Seismic Category II piping are designed, manufactured, tested and installed in accordance with Subsection NF of the ASME Code. For Seismic Category II pipe supports other than standard supports (including pipe supports formed by combining standard support parts with structural elements), the non-standard structural elements are designed, manufactured, installed, and tested in accordance with ANSI/AISC N690.

Non-seismic piping supports used for ASME B31.1 piping meet the requirements of ASME B31.1 (Sections 120 and 121). For nonstandard supports, structural elements are designed using guidance from ANSI/AISC N690. For standard supports and for standard support parts used in nonstandard supports, vendor requirements are be met along with the applicable ASME B31.1 requirements.

The structural elements of supports for non-seismic piping (supports unanalyzed for seismic effects) are designed using guidance from the AISC Steel Construction Manual Reference 3.12-9 and standard piping support parts meet and are used within vendor catalog requirements. Expansion anchors and other steel embedments in concrete used for non-seismic piping supports are designed for concrete strength in accordance with ACI-349 (Reference 3.12-10).

3.12.6.2 Jurisdictional Boundaries

There are two jurisdictional boundaries of piping supports, these are the boundary between the support and the supported or restrained piping, and the boundary between the support and the anchor structure or component. As stated in NF-1131 of the ASME Code, the jurisdictional boundaries between ASME Class 1, 2, and 3 supports and other components, including piping systems, meet the requirements of NB/NC/ND-1132 as applicable to the class of the component. In the NuScale design, most of the Class 1, 2, and 3 piping supports are supported by the containment vessel.

The jurisdictional boundary between the piping and support is typically at the outer surface of the pipe for supports that are not welded directly to the piping. Piping supports which have welded attachments to the piping follow the jurisdictional boundary guidance in NB/NC/ND-1132. For support members which serve a structural function which are welded to the piping (such as lugs), the weld between the support member and the piping are be considered part of the piping. Local stresses on the piping due to any welded attachment which form part of a piping support are evaluated in accordance with applicable ASME Code requirements for the piping.

For ASME Class 1 and 2 piping systems in the NuScale design, all pipe supports are attached to the reactor module and not a building structure, while some ASME Class 3 components are supported by a building structure. For pipe supports attached to the surface of other components (such as the containment vessel), the support boundary is at the surface of the component; the weld shall be considered part of the component. In the case of the containment vessel, pipe support welds conform to the requirements of the containment vessel.

The boundary for piping supports that are attached to building steel are at the interface with the building steel and the weld conform to the requirements of Subsection NF of the ASME Code. The boundary for piping support attachments to concrete building structures is at the surface of the building structure (e.g., baseplate or embedded plate) and the weld conforms to the requirements of Subsection NF of the ASME Code.

Piping systems that are designed and analyzed to ASME B31.1 follow the jurisdictional boundary requirements of ND-1132 of the ASME Code.

3.12.6.3 Loads and Load Combinations

The required load combinations for ASME Code Class 1, 2, and 3 supports are shown in Table 3.12-3.

3.12.6.4 Pipe Support Base Plate and Anchor Bolt Design

All of the Class 1 and 2 pipe supports are supported by the containment vessel; therefore, base plates are not used for any Class 1 or Class 2 pipe supports. Some Class 3 pipe supports may be supported off of the building and may use base plates.

When used, the concrete anchor bolts are evaluated using ACI-349 (Reference 3.12-10), subject to the conditions and limitations of RG 1.199. This guidance accounts for the proper consideration of anchor bolt spacing and distance to a free edge of concrete. In addition, all aspects of the anchor bolt design, including baseplate flexibility and factors of safety, are used in the development of anchor bolt loads as addressed in NRC Bulletin 79-02.

3.12.6.5 Use of Energy Absorbers and Limit Stops

There are no energy absorbers or limit stops used for ASME Code Class 1, 2 or 3 piping.

3.12.6.6 Use of Snubbers

There are no snubbers used for ASME Code Class 1, 2 or 3 piping.

3.12.6.7 Pipe Support Stiffness

In piping system analysis models, pipe supports are modeled using either the actual stiffness of the support structure or with an arbitrarily selected rigid stiffness using checks for support deflection in the restrained direction(s) to verify acceptable values. Where variable spring supports are used, the actual stiffness are modeled. Linear type supports may also be modeled using beam elements within piping models.

For the analysis of ASME Code Class 1, 2, and 3 piping, the support stiffness are modeled consistently throughout the piping model. All supports in the model use their actual stiffness or all supports use a rigid stiffness, except that variable spring supports are modeled with their actual stiffness independent of the method used for the remainder of the supports. Piping supports are designed and selected to preclude having natural frequencies in the unrestrained direction(s) that tend to amplify the attached support structure mass.

For ASME Code Class 1, 2, or 3 supports modeled as rigid in the piping system analysis, two checks for deflection are performed. One check compares the deflection in the restrained direction(s) to a maximum of 1/16th inch for SSE loads or the minimum support design loads. Another check compares the deflection in the restrained direction(s) to a maximum of 1/8th inch for the worst case deflection for any of the specified load combinations. When evaluating pipe support deflections, any dynamic flexible elements of the attaching components or building structure are also considered.

3.12.6.8 Seismic Self-Weight Excitation

The seismic response of components (e.g. vessels) and structures on which pipe supports are attached, due to the effects of the SSE is included in seismic pipe support analysis. Inertial response of the support mass are evaluated using dynamic analysis methods (such as the response spectrum method) similar to that used for the pipe system analysis. Alternatively, the equivalent-static analysis procedure described in Section 3.7.3 may be used to determine pipe support responses due to self-weight excitation. RG 1.61 provides damping values for welded steel and bolted steel connections. When using the uniform response spectrum method, the seismic response of piping supports due to excitation of the pipe support mass, the seismic piping inertial response, and the loads from seismic anchor motions are combined by absolute sum.

Generally, pipe supports are modeled as rigid in piping analyses, using default stiffnesses of the analysis software. If the pipe support stiffnesses do not meet the requirements, then the actual support stiffnesses are determined for all supports in the model, and the piping analyses are re-performed using the determined stiffnesses and including the mass of each support. This procedure ensures that the dynamic response of the supports which are not rigid are adequately characterized in piping support analyses.

Design of Supplementary Steel As discussed in Section 3.12.6.1, all Seismic Category I pipe supports in the NuScale design are designed to Subsection NF of the ASME Code and seismic Category II pipe supports are designed to ANSI/AISC N690. This includes any supplemental steel required to connect the structural elements of pipe supports to the attaching components or building structures. The jurisdictional boundaries are determined in accordance with Subsection NF of the ASME Code (see Section 3.12.6.2) which includes supplemental steel within the support boundary.

Supplemental steel used for non-seismic pipe supports are designed using the AISC Steel Construction Manual (Reference 3.12-9).

3.12.6.9 Consideration of Friction Forces

Frictional forces on pipe supports due to thermal expansion in the unrestrained direction(s) are determined using deadweight and thermal loads normal to the applicable support member. Friction forces due to other loads are not considered.

Friction load is determined using an appropriate coefficient of friction. A minimum coefficient of friction value of 0.30 is used for steel to steel (Reference 3.12-4).

3.12.6.10 Pipe Support Gaps and Clearances

A nominal cold condition gap of 1/16th inch is included radially for all pipe supports. These gaps allow unrestrained radial thermal expansion of piping and unrestrained rotation. A support gap of 1/16th inch around piping provides a total maximum unrestrained span of 1/8th inch in a given direction (in the plane of restraint). Pipe support gaps in the unrestrained direction(s) are specified large enough to accommodate the maximum deflection of the piping systems at the support.

3.12.6.11 Instrumentation Line Support Criteria

The design and analysis of supports for ASME Code Class 1, 2, and 3 instrumentation lines is equivalent to that used for piping supports; this includes the loads, load combinations, and ASME Code acceptance criteria. The design loads include deadweight, thermal expansion, and seismic loads. The load combinations are applied to appropriate ASME design Service Levels in the same manner as piping supports.

Similar to piping supports, analysis and acceptance criteria are in accordance with ASME Code Subsection NF for supports of seismic Category I instrumentation lines. Analysis and acceptance criteria are in accordance with ANSI/AISC N690 for supports of Seismic Category II instrumentation lines. Analysis and acceptance criteria are in accordance with the AISC Steel Construction Manual (Reference 3.12-9) for supports of non-seismic instrumentation lines.

3.12.6.12 Pipe Deflection Limits

Due of the compact size of the NuScale reactor module, the small pipe sizes, and the limited amount of pipe, standard pipe supports are generally not used for ASME Code Class 1, 2, or 3 piping inside the reactor module. However, where standard piping supports or standard piping support parts are used, the manufacturer's recommended deflection limits are followed.

In the NuScale design, spring supports are not used for ASME Code Class 1 and 2 piping. Some ASME Code Class 3 supports may use spring supports. If spring supports are used, the “working range” given in manufacturer catalog load tables is used to determine travel range limits

Where rods or strut supports are used in the design, a tolerance of 1 degree is applied to the manufacturer given swing angle limit. Correspondingly, the installation tolerances of these types of supports is 1 degree. Maximum displacements and rotations at flexible piping joints in ASME B31.1 piping are verified to be within the manufacturer's recommended limits.

The NuScale Power Plant does not use any specialized stiff pipe clamps that would induce high local stresses on the pipe, as discussed in NRC Information Notice 83-80.

3.12.7 References

3.12-1 American Society of Mechanical Engineers Boiler and Pressure Vessel Code, 2013 edition, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," no addenda, New York, NY.

3.12-2 "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Other Loads and Load Combinations," NUREG 1061 Volume 4, December 1984.

3.12-3 American Society of Mechanical Engineers, Power Piping - ASME Code for Pressure Piping B31, ASME B31.1, New York, NY.

3.12-4 Welding Research Council, Inc. "Position Paper on Nuclear Plant Pipe Supports," WRC Bulletin 353, May 1990, Shaker Heights, OH.

3.12-5 Institute of Electrical and Electronics Engineers, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," IEEE Standard 344-1987, Piscataway, NJ.

3.12-6 Electric Power Research Institute, "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 1)," EPRI TR-1011955, June 22, 2011, Palo Alto, CA.

3.12-7 Electric Power Research Institute, "Thermal Stratification, Cycling, and Striping (TASCS)," EPRI TR-103581, July 7, 1999, Palo Alto, CA.

3.12-8 American National Standards Institute/American Institute of Steel Construction, "Specification for Safety-Related Steel Structures for Nuclear Facilities," ANSI/AISC N690-12, January 31, 2012, Chicago, IL.

3.12-9 American Institute of Steel Construction, "Steel Construction Manual," 14th ed. Chicago, IL.

3.12-10 American Concrete Institute, "Code Requirements for Nuclear Safety Related Concrete Structures (ACI 349-13) and Commentary," July 10, 2014, Farmington Hills, MI.

3.12-11 NUREG/CR-1980, Dynamic Analysis of Piping Using the Structural Overlap Method.

3.12-12 March 1, 2014 NRC white paper- Piping Level of Detail for design Certification (ML14065A067).

3.12-13 NuScale Power Module Analysis Technical Report TR-0916-51502

3.12-14 American Society of Civil Engineers, “Seismic Analysis of Safety-Related Nuclear Structures and Commentary,” ASCE 4-98, Reston, VA.

3.12-15 MSS SP-58 - Pipe Hangers and Supports - Materials, Design, Manufacture, Selection, Application, and Installation - SP-58-2009.

3.12-16 NUREG/CR-1677, “Piping Benchmark Problems. Volume I and II.”

3.12-17 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section II, “Materials,” American Society of Mechanical Engineers.

Table 3.12-1: Required Load Combinations for Class 1 Piping

Plant Event	Service Level	Load Combination(10)	Allowable Limit(7)
Design	-	P_{des} + DW + B + DFL (1)	Eq. (9) NB-3652
Normal Operation	A	Range [P + DFL + TE + TAM + CTL]	Eq. (10) NB-3653.1
		Range [TE + TAM + CTL](2)	Eq. (12) NB-3653.6(a)
		Range [P + DFL](2)	Eq. (13) NB-3653.6(b)
		Range [P + DFL + TE + TAM + CTL + TRG]	Eq. (11) or (14)(2) NB-3653.6(c)
		Range [TRG]	NB-3653.7
Transients	B	P_{MAX}	NB-3654.1
		P + DW + B + DFL	Eq. (9) NB-3654.2(a)
		Range [P + DFL + TE + TAM + CTL]	Eq. (10) NB-3654.2(b)
		Range [TE + TAM + CTL](2)	Eq. (12) NB-3654.2(b)
		Range [P + DFL](2)	Eq. (13) NB-3654.2(b)
		Range [TRG]	NB-3654.2(b)
		Range [P + DFL + TE + TAM + CTL ± OBE(3) + TRG]	Eq. (11) or (14)(2)(3) NB-3654.2(b)
Maximum Service Pressure	C(6)	P_{MAX}	NB-3655.1
Design Basis Pipe Break		P + DW + B + DFL + DBPB	Eq. (9) NB-3655.2(a)
Steam Generator Tube Rupture(9)		P + DW + B + DFL + SGTR	
Maximum Service Pressure	D(5)	P_{MAX}	NB-3656(a)(1)
Pipe Breaks		P + DW + B + DFL + MSPB/FWPB	Eq. (9) NB-3656(a)(2)
Pipe Breaks + SSE		P + DW + B + DFL ± SRSS(SSE + MSPB/FWPB/DBPB)(4)	
SAM		Maximum of: Range of Moments [TE + TAM + SAM_{SSE} / 2] OR Range of Moments [SAM_{SSE}]	NB-3656(a)(3) NB-3656(b)(4)
Rod Ejection Accident (REA) (9)		P + DW + B + DFL + REA	Eq. (9) NB-3656 (a)(2)(8)

Table 3.12-1: Required Load Combinations for Class 1 Piping (Continued)

Plant Event	Service Level	Load Combination[10]	Allowable Limit[7]
Pressure Test	Test	DW + B + H	NB-3657
		DW + B + CLIRT	NB-3226

Notes:

(1) DFL for Service Level A are considered for Design Condition.

(2) Load combination is only applicable for those load sets that do not meet the Eq. (10)

(3) OBE loading is only applicable to the fatigue analysis required by ASME Section III, NB-3650 considering the effects of the PWR environment in accordance with the requirements of RG 1.207 and NUREG/CR-6909. When determining applicability of Eq. (14) for the fatigue evaluation, OBE loading are considered for the load combinations specified for Eqs (10) and (13) also. OBE includes both inertial and SAM combined by absolute sum.

(4) Dynamic loads are combined considering the time phasing of the events in accordance with References RG 1.92 and NUREG-0484.

(5) The rules in NB-3656(b) or ASME III Nonmandatory Appendix F may be used as an alternative to NB-3656(a) to evaluate these conditions independent of all other Design and Service Loadings.

(6) If the total number of postulated occurrences for Service Level C conditions result in more than 25 stress cycles having an alternating stress intensity (S_{alt}) greater than the Sa value at 10^6 cycles determined from the applicable fatigue design curves given in ASME BPVC Section III Mandatory Appendix I, those cycles in excess of 25 stress cycles are included in the fatigue analysis (see NB-3113(b)).

(7) ASME Code equations and the stress limits / acceptance criteria are as defined in the referenced citation from ASME BPVC Section III, Subsection NB.

(8) In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.

(9) Dynamic load due to SG tube failure or rod ejection accident is negligible.

(10) Applicable loads are defined in Section 3.12.5.3 and Table 3.9-2.

Table 3.12-2: Required Load Combinations for Class 2 & 3 Piping

Plant Event	Service Level	Load Combination[11]	Allowable Limit[7]
Design	Design	P_{des} + DW + B + DFL[2]	Eq. (8) NC/ND-3652
Normal Operation / Transients	A / B[1]	[P + DW + B + DFL] [3]	Eq. (9a) or (9b) NC/ND-3653.1(a) or (b)
		TE + TAM	Eq. (10a)[4] NC/ND-3653.2(a)
		Non-Repeated Anchor Movement (Building Settlement, etc.)	Eq. (10b) NC/ND-3653.2(b)
		P_{des} + DW + B + TE + TAM + DFL	Eq. (11)[4] NC/ND-3653.2(c)
Permissible Pressure	C	P_{MAX}	NC/ND-3654.1
Design Basis Pipe Break		P + DW + B + DFL + DBPB	Eq. (9a) or (9b) NC/ND-3654.2(a)
Steam Generator Tube Rupture[9]		P + DW + B + DFL + SGTR	
Rod Ejection Accident (REA)[9]	D	P + DW + B + DFL + REA	Eq. (9) NC/ND -3655(a)(2)[10]
Permissible Pressure	D[6][8]	P_{MAX}	NC/ND-3655(a)(1)
Pipe Breaks		P + DW + B + DFL + MSPB/FWPB	Eq. (9a) or (9b) NC/ND-3655(a)(2)
Pipe Breaks + SSE		P + DW + B + DFL ± SRSS(SSE + MSPB/FWPB/DBPB)[5]	
SAM		Maximum of: Range of Moments [TE + TAM + SAM_{SSE} / 2] OR Range of Moments [SAM_{SSE}]	NC/ND-3655(a)(3) NC/ND-3655(b)(4)

Notes:

(1) Evaluation of OBE loads (both inertia and SAM) is not required for Class 2 & 3 piping.

(2) DFL for Service Level A are considered for Design Condition.

(3) Applicable for Level B only.

(4) Requirements of either Eq. 10a or Eq. 11 are met, not both.

(5) Dynamic loads are combined considering the time phasing of the events in accordance with RG 1.92 and NUREG-0484.

(6) The rules in NC/ND -3655(b) or ASME III Nonmandatory Appendix F may be used as an alternative to NC/ND-3655(a) to evaluate these conditions independent of all other Design and Service Loadings.

(7) ASME Code equations and stress limits / acceptance criteria are as defined in the referenced citation from ASME BPVC Section III, Subsections NC or ND as applicable.

(8) For Service Level D, the requirements related to maintaining functional capability which may require piping primary stresses to be limited to Service Level B criteria given in NC/ND-3653.1(a) or (b) are met.

(9) Dynamic load due to SG tube failure or rod ejection accident is negligible.

(10) In accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.

(11) Applicable loads are defined in Section 3.12.5.3 and Table 3.9-2.

Table 3.12-3: Required Load Combinations for Class 1, 2, & 3 Supports

Plant Event[1]	Service Level	Load Combination[2]	Allowable Limit[6][7]
Design	Design	DW + B + TE + TAM + F + DFL	Design
Normal Operations	A	DW + B + TE + TAM + F + DFL	Level A
Transients	B	DW + B + TE + TAM + DFL	Level B
Transients + OBE[3]	B	DW + B + TE + TAM + DFL ± OBE[3]	-
Design Basis Pipe Break[4]	C	DW + B + TE + TAM + DFL + DBPB	Level C
SG Tube Rupture[9]		DW + B + TE + TAM + DFL + SGTR	
Rod Ejection Accident (REA) [8]	D	DW + B + TE + TAM + DFL + REA	Level C
Main Steam and Feedwater Pipe Breaks	D	DW + B + TE + TAM + DFL + MSPB/FWPB	Level D
DBPB/MSPB/FWPB + SSE[4]		DW + B + TE + TAM + DFL ± SRSS(SSE[4] + MSPB/FWPB/DBPB) [5]	
Pressure Test	-	DW + B	Test

Notes:

(1) Fatigue analysis of Class 1 supports are evaluated in accordance with the ASME BPVC Section III.

(2) Applicable loads are defined in Section 3.12.5.3.

(3) OBE loading is only applicable to the fatigue analysis required by Class 1 linear supports subjected to greater than 20,000 cycles of thermal loading as determined from the detailed piping system analysis that satisfies provisions of ASME Section, III NF-3300. OBE includes both inertial and SAM combined by absolute sum.

(4) SSE includes both inertial and SAM combined by absolute sum.

(5) Dynamic loads are combined considering the time phasing of the events in accordance with References RG 1.92 and NUREG-0484.

(6) Stress limits are as defined in NF-3131 of ASME BPVC Section III, Subsection NF for the specified level as applicable to the type of support and class of construction.

(7) For Class 1 linear type and plate-and-shell type supports, the additional stress limit criteria of RG 1.124 and RG 1.130 are met.

(8) Dynamic load due to SG tube failure or Rod Ejection Accident is negligible.

3.13 Threaded Fasteners (ASME Code Class 1, 2, and 3)

This section addresses the application of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1 (Reference 3.13-1), to the design of Class 1, 2, and 3 pressure-retaining threaded fasteners. Threaded fasteners and bolted connections, herein called threaded fasteners unless specified differently, include the bolts, studs, and washers that are associated with Class 1, 2, and 3 pressure retaining joints. Fasteners used for the containment vessel (CNV) are addressed in Section 6.1.

The selection, design, fabrication, installation and inspection of threaded fasteners in the Class 1, 2 and 3 systems meet the criteria of 10 CFR 50.55a, including 10 CFR 50.55a(b)(4), which permits the use of code cases per Regulatory Guide (RG) 1.84 Revision 36.

The threaded fastener design complies with General Design Criteria (GDCs) 1, 4, 14, 30 and 31. Further discussion of compliance with the GDCs are provided in this section.

- GDCs 1 and 30 require that structures, systems, and components (SSC) be designed to quality standards commensurate with the importance of the safety function to be performed. GDCs 1 and 30 are met as the bolting design is in conformance with the criteria of ASME BPVC, Section III and RG 1.65 Revision 1 as described below.
- GDC 4 requires that SSC accommodate the effects of, and that they are compatible with, the environmental conditions of normal and accident conditions. GDC 4 is met by protecting the ASME Class 1, 2, and 3 threaded fasteners from the adverse impacts from lubricants and sealants and by using stainless steels or nickel-base alloys that are resistant to boric acid corrosion.
- GDC 14 is met by designing the threaded fasteners to ASME Class 1 criteria.
- GDC 31 is met by conformance with the requirements of 10 CFR 50, Appendix G, which establishes fracture toughness requirements. Thus the probability of a rapid fracture of the threaded fasteners is minimized satisfying the requirements of GDC 31.

10 CFR 50, Appendix B, Criterion XIII, requires that measures be established to control the cleaning of material and equipment to prevent damage or deterioration. RG 1.28 Revision 4 provides quality assurance criteria for cleaning fluid systems and associated components that comply with 10 CFR 50 Appendix B. The design for threaded fasteners meets the cleaning criteria in RG 1.28.

3.13.1 Design Considerations

The design and analysis of pressure boundary threaded fasteners complies with ASME Class 1, 2 and 3 requirements. Class 1 pressure boundary threaded fasteners are designed in accordance with ASME BPVC, Section III (Reference 3.13-1), Subsection NB. Class 2 and 3 threaded fasteners are designed in accordance with Subsection NC and ND requirements, respectively.

3.13.1.1 Materials Selection

ASME Code Section III provides acceptable standards for selecting threaded fastener material identified in ASME Code, section II (Reference 3.13-2). ASME Section II (Reference 3.13-2) provides the material properties for threaded fasteners for ASME

Class 1, 2, and 3 applications. The applicable criteria used for material selection for ASME Class 1, 2, and 3 threaded fasteners are listed in Table 3.13-1. Materials used for the threaded fasteners are selected for the associated environmental conditions for the lifetime of the plant. Only proven materials for the specific application and environment are used. Bolting material selection satisfies applicable requirements of EPRI TR-101108, "Boric Acid Corrosion Evaluation (BACE) Program, Phase - Task 1 Report," (Reference 3.13-6), EPRI NP-5985, "Boric Acid Corrosion of Carbon and Low-Alloy Steel Pressure-Boundary Components in PWRs," (Reference 3.13-7), and EPRI NP-5558-SL, "Boric Acid Application Guidelines for Intergranular Corrosion Inhibition," (Reference 3.13-8).

The reactor pressure vessel closure studs, nuts, and washers use SB-637 UNS N07718 (Alloy 718), instead of low alloy steels such a SA-540 Grade B23 or B24. The selection of Alloy 718 over traditional low alloy steels is to prevent general corrosion when the bolting is submerged during plant startup and shutdown process. Because of its resistance to general corrosion, the concerns addressed by RG 1.65 position 2(b) do not apply to Alloy 718. Alloy 718 is an austenitic, precipitation hardened, nickel-base alloy permitted for bolting materials by ASME BPVC Code Section III (Reference 3.13-1), Subsection NB-2128.

Being a nonferrous material, the fracture toughness requirements of ASME B&PV Code, Section III (Reference 3.13-1), Subsection NB-2311 exempts Alloy 718 from fracture toughness test requirement in NB-2300. The minimum required room temperature yield strength of SB-637 Alloy 718 is 150 ksi, exceeding the 150 ksi maximum limit in RG 1.65 position 1(a)(i). Because Alloy 718 is nonferrous, it is not subject to the fracture toughness requirements in 10 CFR 50 Appendix G or RG 1.65. Hence, the concern addressed by RG 1.65 position 1(a)(i) is not applicable to Alloy 718.

Alloy 718 is resistant to stress corrosion cracking (SCC) when exposed to high temperature primary reactor coolant, although limited SCC was observed inside reactor vessel internals (Reference 3.13-3). However, SCC is unlikely for reactor vessel closure bolting because it will be submerged at a much lower temperature than reactor coolant temperature. In order to improve SCC resistance, the bolting materials receive a final solution anneal in the range of 1800-1850 degrees F for one hour followed by a two-step aging treatment consisting of 8 hours at 1325 degrees F and 8 to 10 hours at 1150 degrees F. This heat treatment process provides better resistance to SCC and is within the limits of ASME Section II (Reference 3.13-2) material specification for Alloy 718.

Consistent with RG 1.65, lubricant will be selected in accordance with the guidance in NUREG-1339 (Reference 3.13-4). Lubricants containing molybdenum sulfide are prohibited. Based on the above discussion, Alloy 718 bolting material for closure is in compliance with RG 1.65 requirements except for the requirements not applicable to Alloy 718 bolting as described above.

3.13.1.2 Special Materials Fabrication Processes and Controls

The criteria for mechanical property testing of threaded fasteners complies with the requirements of ASME BPVC, Section II (Reference 3.13-2), Part A and Part B as noted in Table 3.13-1. Threaded fastener materials are chosen from proven materials for the specific application and environment and are used after evaluation of the potential for degradation, including galvanic corrosion and SCC. The bolting materials selected for the ASME Class 1, 2 and 3 threaded fasteners are discussed in Sections 4.5, 5.2, 5.3, 6.1 and 6.2.

Fabrication and examination of threaded fasteners are performed in accordance with the criteria in Table 3.13-1 for ASME Code Class 1, 2 and 3 systems.

Lubricants used for the threaded fasteners covered by this section will be selected in accordance with the guidance in NUREG-1339 (Reference 3.13-4) to avoid galvanic corrosion and SCC. Lubricants containing molybdenum sulfide are prohibited.

3.13.1.3 Fracture Toughness Requirements for Threaded Fasteners Made from Ferritic Materials

The pressure-retaining Class 1, 2 and 3 components made of ferritic material meet the requirements of ASME BPVC, Section III (Reference 3.13-1), Subsections NB-2300, NC-2300 and ND-2300 respectively (Table 3.13-1). For pressure-retaining components of the reactor coolant pressure boundary, the requirements are supplemented by the additional requirements set forth in 10 CFR 50, Appendix G.

3.13.1.4 Pre-Service Inspection Requirements

Pressure boundary Class 1, 2 and 3 threaded fasteners are examined in accordance with ASME BPVC, Section XI (Reference 3.13-5), Subsections IWB-2200, IWC-2200 and IWD-2200 respectively for pre-service inspection.

3.13.1.5 Certified Material Test Reports (QA Records)

All Pressure-retaining Class 1, 2 and 3 threaded fasteners are certified in accordance with Subsection NCA-3861 and Subsection NCA-3862 and are furnished with certified material test reports (CMTRs) in accordance with the criteria of ASME BPVC, Section III (Reference 3.13-1) Subsections NB-2130, NC-2130 and ND-2130, respectively.

Material identification is required for all Class 1, 2 and 3 threaded fasteners per ASME BPVC, Section III (Reference 3.13-1), Subsections NB-2150, NC-2150, ND-2150, respectively. CMTRs for ASME Section III Class 1, 2, and 3 threaded fasteners will be retained in accordance with 10 CFR 50.71.

3.13.2 Inservice Inspection Requirements

Inservice Inspection for ASME Class 1, 2, and 3 threaded fasteners is in accordance with the ASME BPVC, Section XI (Reference 3.13-5) (see Table 3.13-2), as required by 10 CFR 50.55a, except where specific written relief has been granted by the NRC.

COL Item 3.13-1: A COL applicant that references the NuScale Power Plant design certification will provide an inservice inspection program for ASME Class 1, 2 and 3 threaded fasteners or describe the implementation program, including milestones, completion dates and expected conclusions. The program will identify the applicable edition and addenda of ASME BPVC, Section XI and ensure compliance with 10 CFR 50.55a.

3.13.3 References

3.13-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section III, "Rules for Construction of Nuclear Facility Components," American Society of Mechanical Engineers.

3.13-2 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section II, "Materials," American Society of Mechanical Engineers.

3.13-3 A. R. McIlree, "Degradation of High Strength Austenitic Alloys X-750, 718 and A286 in Nuclear Power Systems," 1st International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors, NACE, 1984.

3.13-4 U.S. Nuclear Regulatory Commission, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," NUREG-1339, June 1990.

3.13-5 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers.

3.13-6 Electric Power Research Institute, "Boric Acid Corrosion Evaluation (BACE) Program, Phase - Task 1 Report," TR-101108, Palo Alto, CA, December 1993.

3.13-7 Electric Power Research Institute, "Boric Acid Corrosion of Carbon and Low Alloy Steel Pressure Boundary Components in PWRs," NP- 5985, Palo Alto, CA., August 1988.

3.13-8 Electric Power Research Institute, "Boric Acid Application Guidelines for Intergranular Corrosion Inhibition," NP-5558, Palo Alto, CA, December 1987.

Table 3.13-1: ASME BPV Code Section III Criteria for Selection and Testing of Bolted Materials

Code Category		ASME Class 1 Criteria	ASME Class 2 Criteria	ASME Class 3 Criteria
Material Selection		NCA-1220 and NB-2128	NCA-1220 and NC-2128	NCA-1220 and ND-2128
Material test coupons and specimens for ferritic steel material (tensile test criteria)	Heat Treatment Criteria	NB-2210	NC-2210	ND-2210
	Test coupons requirements bolting and studing materials	NB-2221 NB-2224	NC-2221 NC-2224.3	ND-2221 ND-2224.3
Fracture toughness requirements	Materials to be impact tested	NB-2311	NC-2311	ND-2311
	Types of impact test	NB-2321	NC-2321	ND-2321
	Test coupons	NB-2322	NC-2322	ND-2322
	Acceptance standards	NB-2333	NC-2332.3	ND-2333
	Number of impact tests necessary	NB-2345	NC-2345	ND-2345
	Retesting	NB-2350	NC-2352	ND-2352
	Calibration of test equipment	NB-2360	NC-2360	ND-2360
Examination criteria for bolts, studs, and nuts		NB-2580	NC-2580	ND-2580
Certified material test report criteria		NCA-3860	NCA-3860	NCA-3860

Note 1: Section III paragraphs listed in this table represent those specified in the 2013 Edition of Section III.

Table 3.13-2: ASME BPV Code Section XI Examination Categories for Inservice Inspections of Mechanical Joints in ASME Code Class 1, 2, and 3 Systems that are Secured by Threaded Fasteners

<table>
<tr><th>Code Category</th><th>ASME Class 1 Criteria</th><th>ASME Class 2 Criteria</th><th>ASME Class 3 Criteria</th></tr>
<tr><td rowspan="2">Specific bolting inspection</td><td>Table IWB-2500-1 Exam. Cat. B-G-1 for bolting greater that 2 inches in diameter</td><td rowspan="2">Table IWC-2500-1, Exam. Cat. C-D for bolting greater than 2 inches in diameter</td><td rowspan="2">Not Applicable - Currently there are no examination categories that correspond to those that exist for ASME Class 1 and 2 bolting.</td></tr>
<tr><td>Table 1WB-2500-1, Exam. Cat. B-G-2 for bolting less than or equal to 2 inches in diameter</td></tr>
<tr><td>System pressure tests</td><td>Table IWB-2500-1, Exam. Cat. B-P</td><td>Table IWC-2500-1, Exam. Cat. C-H</td><td>Table IWD-2500-1, Exam. Cat. D-B</td></tr>
</table>

Note 1: Section XI paragraphs listed in this table represent those specified in the 2013 Edition of Section XI.