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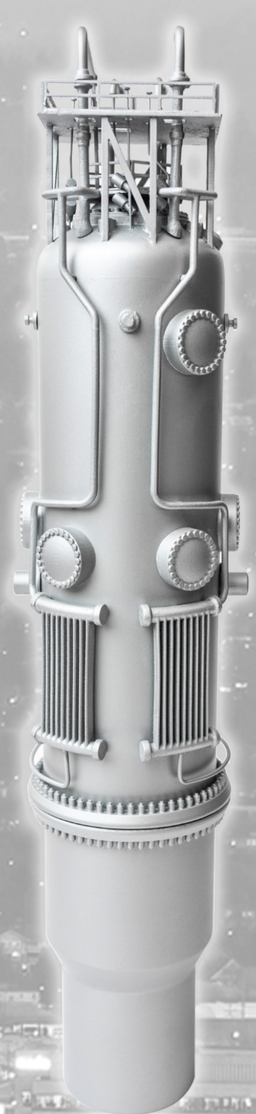
NuScale Standard Plant
Design Certification Application

Chapter Three
Design of Structures, Systems, Components and Equipment



PART 2 - TIER 2

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CHAPTER 3 DESIGN OF STRUCTURES, SYSTEMS, COMPONENTS AND EQUIPMENT

3.1 Conformance with U.S. Nuclear Regulatory Commission General Design Criteria

This section addresses design compliance with the General Design Criteria (GDC) in 10 CFR 50, Appendix A, for safety-related and when appropriate, risk-significant structures, systems, and components (SSC).

The following sections state the criterion and then address how the criterion is implemented in the NuScale Power Plant design. The section provides a statement regarding the conformance or exception, as well as a list of sections where additional information on conformance is presented.

In certain cases, NuScale meets the intent of the GDC or has developed a principal design criterion (PDC) to address the specific design of the NuScale Power Plant pressurized water reactor.

3.1.1 Overall Requirements

3.1.1.1 Criterion 1-Quality Standards and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Implementation in the NuScale Power Plant Design

NuScale's quality assurance (QA) program satisfies the requirements of 10 CFR 50 Appendix B and ASME NQA-1-2008 and NQA-1a-2009 addenda, "Quality Assurance Requirements for Nuclear Facility Applications" (Reference 3.1-1). As such, the NuScale QA program provides confidence that the SSC that are required to perform safety-related and risk-significant functions will perform the functions satisfactorily. NuScale's QA program is described in the NuScale Quality Assurance Program Description (QAPD).

NuScale plant SSC are assigned safety and QA classifications based on their safety and risk-significant functions. The QA classification is used to identify and apply appropriate QA requirements for safety-related and risk-significant SSC. The safety and QA classifications assigned to NuScale plant SSC are indicated in Table 3.2-1.

Compliance with recognized codes, standards, and design criteria is documented in appropriate records associated with plant design, procurement, fabrication, inspection, erection, and testing and maintained throughout the life of the plant.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 1.

Relevant FSAR Chapters and Sections

Section 3.2	Classification of Structures, Systems, and Components
Section 3.9	Mechanical Systems and Components
Section 3.10	Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment
Section 3.11	Environmental Qualification of Mechanical and Electrical Equipment
Section 3.13	Threaded Fasteners (ASME Code Class 1, 2, and 3)
Chapter 5	Reactor Coolant System and Connecting Systems
Chapter 6	Engineered Safety Features
Chapter 7	Instrumentation and Controls
Section 9.1.5	Overhead Heavy Load Handling System
Section 9.3	Process Auxiliaries
Chapter 17	Quality Assurance and Reliability Assurance

3.1.1.2 Criterion 2-Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Implementation in the NuScale Power Plant Design

The safety-related SSC in the NuScale Power Plant are designed to withstand the effects of natural phenomena based on parameters selected to bound the hazardous characteristics associated with the natural phenomena of most potential plant sites. The design bases for safety-related SSC reflect this envelope of natural phenomena, including appropriate combinations of the effects of normal operating and accident conditions. The NuScale Power Plant's site design parameters are listed in Table 2.0-1. Seismic and quality group classifications, and other pertinent standards and information are provided in Table 3.2-1.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 2.

Relevant FSAR Chapters and Sections

Chapter 2	Site Characteristics and Site Parameters
Section 3.2	Classification of Structures, Systems, and Components
Section 3.3	Wind and Tornado Loadings
Section 3.4	Water Level (Flood) Design
Section 3.5	Missile Protection
Section 3.7	Seismic Design
Section 3.8	Design of Category I Structures
Section 3.9	Mechanical Systems and Components
Section 3.10	Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment
Section 3.11	Environmental Qualification of Mechanical and Electrical Equipment
Section 3.12	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports
Chapter 5	Reactor Coolant System and Connecting Systems
Chapter 6	Engineered Safety Features
Section 7.1	Fundamental Design Principles
Section 8.3	Onsite Power Systems
Section 9.1.2	New and Spent Fuel Storage
Section 9.1.3	Spent Fuel Pool Cooling and Cleanup System
Section 9.3	Process Auxiliaries

Section 9.4.1 Control Room Area Ventilation System

Chapter 15 Transient and Accident Analyses

3.1.1.3 Criterion 3-Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Implementation in the NuScale Power Plant Design

The NuScale Power Plant fire protection design and program ensure that the requirements of 10 CFR 50.48 and GDC 3 are met. The SSC are designed and located to minimize the probability and effects of fires and explosions. Noncombustible and fire-resistant materials are used throughout the plant where fire is a potential risk to safety-related systems. Fire barriers ensure that redundant, safety-related systems and components are separated to assure that a fire in one area will not affect the redundant systems and components in an adjacent area from performing their safety functions.

Buildings that contain equipment required for safe shutdown are compartmentalized to minimize the impacts of a fire. These divisions and sub-divisions ensure adequate equipment and cable separation meet the enhanced fire protection criteria. Compartmentalization is achieved by using properly rated fire barriers, fire doors, fire dampers, and penetration seals to prevent the spread of fire between areas.

The fire protection system and equipment is designed in accordance with the guidance provided in Regulatory Guide 1.189, Revision 2, and applicable National Fire Protection Association codes. This ensures that the fire detection and fighting systems provided have the capacity and capability to minimize the adverse effects of fires and that their rupture or inadvertent operation does not significantly impair the safety capability of other SSC.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 3.

Relevant FSAR Chapters and Sections

Section 9.3 Process Auxiliaries

Section 9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

Section 9.5	Other Auxiliary Systems
Appendix 9A	Fire Hazard Analysis
Section 11.2	Liquid Waste Management System
Section 11.3	Gaseous Radioactive Waste Management System

3.1.1.4 Criterion 4-Environmental and Dynamic Effects Design Bases

Structures, systems, and components important to safety shall be 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. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Implementation in the NuScale Power Plant Design

The design of safety-related and risk-significant SSC is such that the effects of environmental conditions associated with normal operation, maintenance testing, and postulated accidents, including LOCAs, are accommodated. The NuScale Power Plant design appropriately protects against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the NuScale Power Module (NPM) and prevents piping failure using leak-before-break methodology.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 4.

Relevant FSAR Chapters and Sections

Section 3.3	Wind and Tornado Loadings
Section 3.4	Water Level (Flood) Design
Section 3.5	Missile Protection
Section 3.6	Protection against Dynamic Effects Associated with Postulated Rupture of Piping
Section 3.8	Design of Category I Structures
Section 3.9	Mechanical Systems and Components

Section 3.10	Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment
Section 3.11	Environmental Qualification of Mechanical and Electrical Equipment
Section 3.12	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports
Section 3.13	Threaded Fasteners (ASME Code Class 1, 2, and 3)
Section 4.6	Functional Design of Control Rod Drive System
Chapter 5	Reactor Coolant System and Connecting Systems
Chapter 6	Engineered Safety Features
Chapter 7	Instrumentation and Controls
Section 8.3	Onsite Power Systems
Chapter 9	Auxiliary Systems
Chapter 10	Steam and Power Conversion System
Chapter 15	Transient and Accident Analyses

3.1.1.5 Criterion 5-Sharing of Structures, Systems, and Components

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Implementation in the NuScale Power Plant Design

The term NuScale Power Plant refers to the entire site, including up to 12 NPMs and the associated balance of plant support systems and structures. The design considers the safety effects and the risk associated with multi-module plant operation with shared or common systems such that each NPM can be safely operated independent of other NPMs. The plant includes design features that ensure the independence and protection of NPM safety systems during all operational modes. Given a single failure in safety-related SSC in one NPM, these design features ensure that safety functions are capable of being performed in other NPMs. The NuScale Power Plant is designed such that a failure of a shared system, which are nonsafety-related with exception of the ultimate heat sink (UHS), does not prevent the performance of NPM safety functions.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 5.

Relevant FSAR Chapters and Sections

Section 5.4.3	Decay Heat Removal System
Section 6.2	Containment Systems
Section 6.3	Emergency Core Cooling System
Section 6.4	Control Room Habitability
Chapter 7	Instrumentation and Controls
Chapter 8	Electric Power
Chapter 9	Auxiliary Systems
Chapter 10	Steam and Power Conversion System
Chapter 15	Transient and Accident Analyses
Chapter 21	Multi-Module Design Considerations

3.1.2 Protection by Multiple Fission Product Barriers

3.1.2.1 Criterion 10-Reactor Design

The reactor core and associated coolant, control, and protection systems shall be 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.

Implementation in the NuScale Power Plant Design

The reactor core and associated coolant, control, and protection systems are designed with appropriate margin such that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

During AOOs and low probability events that may result in a plant shutdown, the NuScale Power Plant is designed such that the reactor will be brought to subcritical conditions and maintained in safe shutdown. The reactor core is designed to maintain integrity over a complete range of power levels and sized with sufficient heat transfer area and coolant flow such that SAFDLs are not exceeded.

Safety analysis design limits are established to demonstrate conformance with GDC 10. These limits ensure that the fuel boundary is not breached, thus leaving the first fission product barrier intact. SAFDLs also ensure that the fuel system dimensions remain within operational tolerances and that the functional capabilities are not reduced below those assumed in the safety analysis.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 10.

Relevant FSAR Chapters and Sections

Section 3.9.5	Reactor Vessel Internals
Section 4.2	Fuel System Design
Section 4.3	Nuclear Design
Section 4.4	Thermal and Hydraulic Design
Chapter 7	Instrumentation and Controls
Section 9.3.4	Chemical and Volume Control System
Chapter 15	Transient and Accident Analyses

3.1.2.2 Criterion 11-Reactor Inherent Protection

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Implementation in the NuScale Power Plant Design

The reactor core and associated coolant systems are designed such that inherent reactivity control is provided during changing plant conditions. The two main feedback effects that compensate for a rapid increase in reactivity are the fuel Doppler temperature reactivity coefficient and the fuel moderator temperature coefficient.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 11.

Relevant FSAR Chapters and Sections

Section 4.3	Nuclear Design

3.1.2.3 Criterion 12-Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Implementation in the NuScale Power Plant Design

The NuScale reactor core is designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible. Oscillations are evaluated at the beginning, middle, and end of the equilibrium cycle. The NuScale reactor core is stable with respect to axial and radial stability, as discussed in Section 4.3.2.

Oscillations in core power can be readily detected by the fixed in-core detector system, which continuously monitors the core flux distribution.

The reactor core and associated coolant, control, and protection systems ensure that power and hydraulic oscillations that can result in conditions exceeding SAFDLs are not possible. Hydraulic stability protection is achieved by the regional exclusion method. The module protection system (MPS) enforces this regional exclusion by ensuring the NPM maintains adequate riser subcooling.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 12.

Relevant FSAR Chapters and Sections

Section 4.3 Nuclear Design

Section 4.4 Thermal and Hydraulic Design

Section 15.9 Stability

3.1.2.4 Criterion 13-Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Implementation in the NuScale Power Plant Design

Instrumentation and controls are provided to monitor variables and systems over their anticipated ranges for normal operations, AOOs, and postulated accident conditions to assure adequate safety. The design of the NuScale safety-related instrument and control systems is based on independence, redundancy, predictability and repeatability, and diversity and defense-in-depth. The appropriate controls are provided to the NPM with sufficient margin to ensure these variables and systems remain within the prescribed operating ranges.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 13.

Relevant FSAR Chapters and Sections

Chapter 6	Engineered Safety Features
Chapter 7	Instrumentation and Controls
Chapter 9	Auxiliary Systems
Chapter 15	Transient and Accident Analyses

3.1.2.5 Criterion 14-Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Implementation in the NuScale Power Plant Design

The reactor pressure vessel (RPV) and pressure retaining components associated with the reactor coolant pressure boundary (RCPB) are designed and fabricated with sufficient margin to assure the RCPB behaves in a non-brittle manner and to minimize the probability of abnormal leakage, rapidly propagating fracture, and gross rupture. The RCPB materials meet the fabrication, construction, and testing requirements of the ASME Boiler and Pressure Vessel Code (BPVC), Section III Division 1, Subsection NB (Reference 3.1-2) and the materials selected for fabrication of the RCPB meet the ASME BPVC, Section II (Reference 3.1-3) requirements.

The primary and secondary water chemistry, along with the water chemistry for the pools forming the ultimate heat sink, is controlled to monitor for chemical species that can affect the RCPB integrity. Sampling and analysis of reactor coolant and pool water samples verify that key chemistry parameters are within prescribed limits and that impurities are properly controlled. This provides assurance that corrosion is mitigated and will not adversely affect the RCPB.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 14.

Relevant FSAR Chapters and Sections

Section 3.9	Mechanical Systems and Components
Section 3.12	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components, and Associated Supports
Section 3.13	Threaded Fasteners (ASME Code Class 1, 2, and 3)

Chapter 5 Reactor Coolant System and Connecting Systems

Section 9.3 Process Auxiliaries

Section 10.3.5 Water Chemistry

Section 10.4.6 Condensate Polishing System

3.1.2.6 Criterion 15-Reactor Coolant System Design

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Implementation in the NuScale Power Plant Design

The overpressure protection system is designed with sufficient capacity to prevent the RCPB from exceeding 110 percent of design pressure during normal operations and AOOs. The system ensures that design limits are not exceeded during an anticipated transient without scram. The overpressure protection system is able to perform its function assuming a single active failure and concurrent loss of normal AC power.

Overpressure protection is provided by the reactor safety valves and in accordance with the requirements of ASME Code, Section III Division 1, Subsection NB for the RCPB and Subsection NC (Reference 3.1-4) for the secondary side of the steam generator and decay heat removal system (DHRS).

Conformance or Exception

The NuScale Power Plant design conforms to GDC 15.

Relevant FSAR Chapters and Sections

Section 3.9 Mechanical Systems and Components

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

Chapter 5 Reactor Coolant System and Connecting Systems

Chapter 7 Instrumentation and Controls

Chapter 15 Transient and Accident Analyses

3.1.2.7 Criterion 16-Containment Design

Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Implementation in the NuScale Power Plant Design

The containment and associated systems are designed to establish an essentially leak-tight barrier against an uncontrolled release of radioactivity to the environment, and assures that containment design conditions are not exceeded for as long as the postulated accident conditions require. The integrity of the containment vessel (CNV) and the passive isolation barriers, along with the isolation of the lines that penetrate primary containment accomplish the provisions of GDC 16.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 16.

Relevant FSAR Chapters and Sections

Section 3.8.2 Steel Containment

Section 6.2 Containment Systems

3.1.2.8 Criterion 17-Electric Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Implementation in the NuScale Power Plant Design

The NuScale Power Plant is designed with passive safety-related systems for safe shutdown, core and spent fuel assembly cooling, containment isolation and integrity, and RCPB integrity. Electrical power is not relied upon to meet SAFDLs or to protect the RCPB as a result of AOOs or postulated accidents. The availability of electrical power sources does not affect the ability to achieve and maintain safety-related functions.

Although not relied on to ensure plant safety-related functions are achieved, the design of the AC and DC power systems includes provisions for independence and redundancy.

Conformance or Exception

The NuScale Power Plant design does not conform to GDC 17. The NuScale design supports an exemption from the criterion.

Relevant FSAR Chapters and Sections

Chapter 8 Electric Power

Chapter 15 Transient and Accident Analyses

3.1.2.9 Criterion 18-Inspection and Testing of Electric Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Implementation in the NuScale Power Plant Design

The electric power supply systems in the NuScale Power plant do not contain any safety-related or risk-significant SSC that are required to meet GDC 18. Although not relied on to meet GDC 18, the plant design does include provisions for testing and inspecting of power supply systems.

Conformance or Exception

The NuScale Power Plant design does not conform to GDC 18. The NuScale design supports an exemption from the criterion.

Relevant FSAR Chapters and Sections

Chapter 8 Electric Power

3.1.2.10 Criterion 19-Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under 50.67, shall meet the requirements of this except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in 50.2 for the duration of the accident.

Implementation in the NuScale Power Plant Design

The NuScale design supports an exemption from the provisions of GDC 19. The following PDC has been adopted:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.
>
> Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) as defined in 10 CFR 50.2 for the duration of the accident.

> Equipment at appropriate locations outside the control room shall be provided with a design capability for safe shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe shutdown condition.

The NuScale Power main control room contains the instrumentation and controls necessary to operate the NPMs safely under normal conditions and to maintain them in a safe condition under accident conditions, including a LOCA. Adequate protection is provided to permit access and occupancy of the control room so that personnel do not receive a whole body dose greater than 5 rem.

Heating, ventilation, and air conditioning are normally provided to the main control room by the control room ventilation system. Redundant toxic gas detectors, smoke detectors, and radiation detectors are provided in the outside air duct, upstream of both the control room ventilation system filter units and the bubble tight outdoor air isolation dampers. Upon detection of a high radiation level in the outside air intake, the system is realigned so that 100 percent of the outside air passes through the control room ventilation system filter unit. When power is unavailable, or if high levels of radiation are detected downstream of the charcoal filtration unit, the control room ventilation system filter unit is stopped, the outside air intake is automatically isolated, and the bubble-tight isolation dampers are closed. Once the control room envelope dampers are closed, the control room envelope is maintained for up to 72 hours by the control room habitability system.

The NuScale main control room (MCR) is designed with the ability to place the reactors in safe shutdown in the event of an MCR evacuation event, and for safe shutdown to be maintained without operator action thereafter. Prior to evacuating the MCR, operators trip the reactors, initiate decay heat removal and initiate containment isolation. These actions result in passive cooling that achieves safe shutdown of the reactors. Operators can also achieve safe shutdown of the reactors from outside the MCR in the MPS equipment rooms within the reactor building. Following shutdown and initiation of passive cooling from either the MCR or the MPS equipment rooms, the NuScale design does not rely on operator action, instrumentation, or controls outside of the MCR to maintain safe shutdown condition. The design includes a remote shutdown station (RSS) for monitoring of the plant if the MCR is evacuated. There are no displays, alarms, or controls in the RSS credited to meet the requirements of principal design criterion (PDC) 19 as there is no manual control of safety-related equipment allowed from the RSS.

Conformance or Exception

The NuScale Power Plant design departs from GDC 19 and supports an exemption from the criterion. The NuScale Power Plant design conforms to PDC 19.

Relevant FSAR Chapters and Sections

Section 5.4.3	Decay Heat Removal System
Section 6.4	Control Room Habitability
Section 7.1	Fundamental Design Principles

Section 9.4.1	Control Room Area Ventilation System
Section 9.5	Other Auxiliary Systems
Appendix 9A	Fire Hazard Analysis
Section 11.5	Process and Effluent Radiation Monitoring Instrumentation and Sampling
Section 12.3	Radiation Protection Design Features
Chapter 15	Transient and Accident Analyses
Section 18.7	Human-System Interface Design

3.1.3 Protection and Reactivity Control Systems

3.1.3.1 Criterion 20-Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Implementation in the NuScale Power Plant Design

The MPS monitors process parameters that are directly related to equipment mechanical limitations, monitors parameters that directly affect the heat transfer capability of the NPM, and automatically executes safety-related functions in response to out-of-normal conditions. The MPS, in response to the NPM exceeding an analytical safety limit, trips the reactor. The MPS also actuates the engineered safety features actuation system (ESFAS) when specified setpoints are exceeded to prevent or mitigate damage to the reactor core and RCS.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 20.

Relevant FSAR Chapters and Sections

Chapter 7	Instrumentation and Controls
Chapter 15	Transient and Accident Analyses

3.1.3.2 Criterion 21-Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Implementation in the NuScale Power Plant Design

The MPS incorporates the design principles of redundancy and independence such that no single failure results in the loss of the protective function. The MPS has four redundant groups of signal conditioning and trip determination, two divisions of reactor trip systems (RTSs) and ESFAS, and redundant communication paths. Each safety function uses two-out-of-four voting logic with two independent divisions of RTS and ESFAS so that a single failure will not prevent the safety function from being accomplished. The MPS SSC are designed to be tested and calibrated while retaining the capability to accomplish its required safety function. The MPS is designed for high functionality and to permit periodic testing during operation, including the ability to test channels independently to determine if failures or a loss of redundancy have occurred.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 21.

Relevant FSAR Chapters and Sections

Chapter 7	Instrumentation and Controls
Section 9.3.4	Chemical and Volume Control System

3.1.3.3 Criterion 22-Protection System Independence

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Implementation in the NuScale Power Plant Design

The MPS equipment is located in the Reactor Building and is designed to enable systems and components required for safe plant operation to withstand natural phenomena, postulated design basis accidents, and design basis threats. The MPS has four redundant groups of signal conditioning and trip determination, two divisions of RTS and ESFAS, and redundant communication paths. Each safety function uses two-out-of-four voting logic with two independent divisions of RTS and ESFAS so that a single failure will not prevent the safety function from being accomplished. The MPS SSC are designed to be tested and calibrated while retaining the capability to accomplish its required safety function. The MPS is designed for high functionality and to permit periodic testing during operation, including the ability to test channels independently to determine if failures or a loss of redundancy have occurred. To the extent practical, functional diversity and diversity in component design is used to perform the protection functions and prevent its loss.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 22.

Relevant FSAR Chapters and Sections

Chapter 7.1	Fundamental Design Principles

3.1.3.4 Criterion 23-Protection System Failure Modes

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Implementation in the NuScale Power Plant Design

The MPS uses self-diagnoses to detect fatal faults and fail into a safe state. The SSC associated with the MPS are provided with a constant signal to maintain a non-actuated state. Upon loss of signal, the SSC fail into a safe state.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 23.

Relevant FSAR Chapters and Sections

Section 3.11	Environmental Qualification of Mechanical and Electrical Equipment
Section 4.6	Functional Design of Control Rod Drive System
Chapter 7	Instrumentation and Controls

3.1.3.5 Criterion 24-Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system.

Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Implementation in the NuScale Power Plant Design

The MPS incorporates redundancy in multiple areas so that a single failure or removal from service will not prevent safety functions from being accomplished when required. The MPS has four redundant groups of signal conditioning and trip determination, two divisions of RTS and ESFAS, and redundant communication paths. Each safety function uses two-out-four voting and there are two independent, diverse, and redundant divisions of RTS and ESFAS so that a single failure will not prevent the safety function from being accomplished.

The MPS does not have any connections between divisions. Qualified, safety-related, one way isolation devices are used to send data from the MPS to nonsafety-related systems and to provide input from nonsafety-related systems to the protection systems.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 24.

Relevant FSAR Chapters and Sections

Chapter 7	Instrumentation and Controls

3.1.3.6 Criterion 25-Protection System Requirements for Reactivity Control Malfunctions

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Implementation in the NuScale Power Plant Design

The setpoints of the MPS will assure that reactor trip or engineered safety feature actuation occurs before the process reaches the analytical limit. The setpoints are chosen to assure the plant can operate and experience expected operational transients without unnecessary trips or engineered safety feature actuations. Chapter 15 safety analyses demonstrate that the control rod drive system (CRDS) with any assumed credible failure of any single active component is capable of performing a reactor trip when plant parameters exceed the reactor trip setpoint, in accordance with GDC 25.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 25.

Relevant FSAR Chapters and Sections

Section 4.3	Nuclear Design

Section 4.6 Functional Design of Control Rod Drive System

Chapter 7 Instrumentation and Controls

Chapter 15 Transient and Accident Analyses

3.1.3.7 Criterion 26-Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Implementation in the NuScale Power Plant Design

The NuScale Power Plant design incorporates two independent reactivity control systems of different design principle: CRDS and the chemical and volume control system (CVCS), in conjunction with the boron addition system.

The CRDS 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 via gravity-dropping of the control rod assemblies (CRAs) 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.

The CVCS operates in conjunction with the boron addition system to satisfy GDC 26 as the second reactivity control system. The CVCS has the ability to control the soluble boron concentration to compensate for fuel depletion during operation and xenon burnout reactivity changes, to assure acceptable fuel design limits are not exceeded. The CVCS is designed to maintain the reactor as subcritical under cold conditions.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 26.

Relevant FSAR Chapters and Sections

Section 3.9.4 Control Rod Drive System

Section 4.3 Nuclear Design

Section 4.6 Functional Design of Control Rod Drive System

Section 9.3 Process Auxiliaries

Chapter 15 Transient and Accident Analyses

3.1.3.8 Criterion 27-Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Implementation in the NuScale Power Plant Design

GDC 27 is not applicable to the NuScale design. The following PDC has been adopted:

> The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions with all rods fully inserted.

Consistent with GDC 27, this PDC requires that the reactivity control systems function, together with heat removal systems, to protect the core from unacceptable damage under accident conditions. This protection function is met by providing sufficient reactivity control such that core cooling is maintained under accident conditions, analyzed using conservative methodology and assumptions including margin equivalent to the highest worth rod stuck out. Under the NuScale design basis, during normal operation sufficient negative reactivity is maintained (instantaneous shutdown margin) to ensure that the capability to cool the core is maintained under accident conditions by rapid control rod insertion with the highest worth rod stuck out.

The PDC also includes a post-accident holddown criterion specific to the NuScale design. This provision requires the control rods to be capable of maintaining the core subcritical under cold conditions following a postulated accident, without margin for the highest worth rod stuck out. Conservative analysis indicates that a post-accident return to power could occur following initial shutdown, under the condition that the highest worth CRA does not insert. The CVCS system is capable of providing negative reactivity but is not credited in this analysis since it is not a safety-related system. Section 15.0.6 demonstrates that the passive heat removal safety systems provide sufficient thermal margin such that a return to power does not result in the failure of the fuel cladding fission product barrier, as demonstrated by not exceeding SAFDLs for the analyzed events.

The reactivity control capability required by either GDC 27 or PDC 27 provides assurance that even if a postulated accident damages fuel, continued core cooling will not be precluded and thus accident consequences can be maintained within acceptable limits. The NuScale design assures that fuel cladding integrity is maintained for all design basis events, including postulated accidents, such that the effect of a postulated return to power with failed fuel has not been evaluated in the analysis of accident consequences. Therefore to preclude unanalyzed accident consequences, NuScale's design basis implements PDC 27 in Chapter 15 to prohibit fuel failures under postulated accident conditions.

Conformance or Exception

The NuScale Power Plant design departs from GDC 27 and supports an exemption from the criterion. The NuScale Power Plant design conforms to PDC 27.

Relevant FSAR Chapters and Sections

Section 3.9.4	Control Rod Drive System
Section 4.2	Fuel System Design
Section 4.3	Nuclear Design
Section 4.6	Functional Design of Control Rod Drive System
Section 6.3	Emergency Core Cooling System
Section 9.3.4	Chemical and Volume Control System
Chapter 15	Transient and Accident Analyses

3.1.3.9 Criterion 28-Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Implementation in the NuScale Power Plant Design

The NuScale design places limits on the worth of CRAs, the maximum CRA withdrawal rate, and the CRA insertion. The maximum worth of control rods and control rod insertion limits preclude rupture of the RCPB due to a rod withdrawal or rod ejection accident. Section 15.4 addresses plant safety associated with the reactivity insertion rates.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 28.

Relevant FSAR Chapters and Sections

Section 4.3	Nuclear Design
Section 4.6	Functional Design of Control Rod Drive System
Chapter 7	Instrumentation and Controls
Section 9.3.4	Chemical and Volume Control System
Chapter 15	Transient and Accident Analyses

3.1.3.10 Criterion 29-Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Implementation in the NuScale Power Plant Design

The CRDS and the protection systems are designed to assure a high probability of performing the required safety-related functions in the event of AOO.

The CRDS can perform safety-related functions to control the reactor within fuel and plant limits during AOOs despite a single failure of the system. The CRDS performs a safe shutdown via gravity-dropping of the CRAs on a reactor trip signal or loss of power. The CRDS maintains an ASME BPVC, Section III Division 1, Subsection NB Class 1 boundary for the reactor coolant during normal, upset, emergency, and faulted operating conditions. The safety-related reactor trip function of the CRDS is initiated by MPS through the RTS. The CRDS performs a reactor trip when plant parameters exceed the reactor trip setpoint. Therefore, the reactor is placed in a subcritical condition with any assumed credible failure of any single active component.

The protection systems are designed with sufficient redundancy and diversity to assure high probability of accomplishing their safety-related functions in the event of AOOs.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 29.

Relevant FSAR Chapters and Sections

Section 3.9.4	Control Rod Drive System
Section 4.6	Functional Design of Control Rod Drive System
Chapter 7	Instrumentation and Controls
Section 9.3.4	Chemical and Volume Control System

Chapter 15	Transient and Accident Analyses

3.1.4 Fluid Systems

3.1.4.1 Criterion 30-Quality of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Implementation in the NuScale Power Plant Design

The RPV and pressure retaining components associated with the RCPB are designed, fabricated, and tested in accordance with ASME BPVC, Section III Division 1, Subsection NB, Class 1 are consistent with 10 CFR 50.3 and 10 CFR 50.55a.

The containment evacuation system supports two methods for detecting and, to the extent practical, identifying the source of reactor coolant leakage. These leak detection methods are CNV pressure monitoring and containment evacuation system sample tank level change monitoring. Both leak detection methods are consistent with the guidance in Regulatory Guide 1.45.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 30.

Relevant FSAR Chapters and Sections

Section 3.2	Classification of Structures, Systems, and Components
Section 3.9.6	Functional Design, Qualification and Inservice Testing Program for Pumps, Valves and Dynamic Restraints
Section 3.13	Threaded Fasteners (ASME Code Class 1, 2, and 3)
Section 5.2	Integrity of Reactor Coolant Boundary
Section 5.3	Reactor Vessel
Section 9.3.6	Containment Evacuation System and Containment Flooding and Drain System
Section 11.5	Process and Effluent Radiation Monitoring Instrumentation and Sampling

3.1.4.2 Criterion 31-Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Implementation in the NuScale Power Plant Design

Overpressure protection is provided for the RCPB during low temperature conditions to assure the pressure boundary behaves in a non-brittle manner and the probability for rapidly propagating fracture is minimized. The ferritic materials provide sufficient margin to account for uncertainties associated with flaws and the effects of service and operating conditions.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 31.

Relevant FSAR Chapters and Sections

Section 3.13 Threaded Fasteners (ASME Code Class 1, 2, and 3)

Section 5.2 Integrity of Reactor Coolant Boundary

Section 5.3 Reactor Vessel

Section 6.1 Engineered Safety Feature Materials

3.1.4.3 Criterion 32-Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Implementation in the NuScale Power Plant Design

Components which are part of the RCPB are designed and provided with access to permit periodic inspection and testing requirements for ASME BPVC, Section III Division 1, Subsection NB Class 1 pressure-retaining components in accordance with ASME BPVC, Section XI Division 1 (Reference 3.1-5) pursuant to 10 CFR 50.55a(g). Equipment that may require inspection or repair is placed in an accessible position to minimize time and radiation exposure during refueling and maintenance outages. Plant technicians may access components without being placed at risk for dose or situations where excessive plates, shields, covers, or piping must be moved or removed in order to access components.

The RPV material surveillance program monitors changes in the fracture toughness properties. Specimens are periodically removed and tested in order to monitor changes in fracture toughness in accordance with "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels," ASTM E185-82 (Reference 3.1-6), as required by 10 CFR 50, Appendix H. Table 5.3-2 lists the specimen matrix for the NuScale material surveillance program requirements.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 32.

Relevant FSAR Chapters and Sections

Section 3.9.6 Functional Design, Qualification and Inservice Testing of Pumps, Valves and Dynamic Restraints

Section 5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

Section 5.3.1 Reactor Vessel Materials

3.1.4.4 Criterion 33-Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Implementation in the NuScale Power Plant Design

The CVCS provides reactor coolant makeup during normal operation for small leaks in the RCPB, but is not relied upon during a design basis event. The RPV and CNV design retain sufficient RCS inventory that, in conjunction with safety actuation setpoints to isolate CVCS from the RCS and operation of emergency core cooling system (ECCS), adequate cooling is maintained and the SAFDLs are not exceeded in the event of a small break in the RCPB.

Conformance or Exception

The NuScale Power Plant design does not conform to GDC 33. The NuScale design supports an exemption from the criterion.

Relevant FSAR Chapters and Sections

Section 8.2 Offsite Power System

Section 8.3 Onsite Power Systems

Section 9.3.4 Chemical and Volume Control System

3.1.4.5 Criterion 34-Residual Heat Removal

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Implementation in the NuScale Power Plant Design

The NuScale design supports an exemption from the power provisions of GDC 34. The following PDC has been adopted:

> A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that the system safety function can be accomplished, assuming a single failure.

The decay and residual heat removal safety function is performed by the DHRS flowpath and containment isolation function of the containment system performed by the main steam isolation valves (MSIVs), the main steam isolation bypass valves, and feedwater isolation valves.

The DHRS is a closed-loop, passive condenser design that utilizes circulation flow from the steam generators to dissipate residual and decay core heat to the UHS. The DHRS consists of two independent subsystems, each capable of performing the system safety function in the event of a single failure. The DHRS actuation valves actuate upon loss or an interruption of electrical power.

Conformance or Exception

The NuScale Power Plant design conforms to PDC 34.

Relevant FSAR Chapters and Sections

Section 5.4.3 Decay Heat Removal System

Section 8.2 Offsite Power System

Section 8.3 Onsite Power Systems

Chapter 10 Steam and Power Conversion System

Chapter 15 Transient and Accident Analyses

3.1.4.6 Criterion 35-Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Implementation in the NuScale Power Plant Design

The NuScale design supports an exemption from the power provisions of GDC 35. The following PDC has been adopted:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that the system safety function can be accomplished, assuming a single failure.

The ECCS provides adequate passive heat removal following any loss of reactor coolant event.

The ECCS is fully enclosed inside containment and consists of three reactor vent valves located on the head of the RPV and two reactor recirculation valves located on the side of the RPV. All five valves are closed during normal operation and open when the system is actuated during accident conditions. The reactor vent valves allow steam to flow from the RPV into the CNV, where it then condenses on the CNV walls and collects at the bottom of the CNV. The condensed coolant then reenters the RPV through the reactor recirculation valves and is recirculated to cool the reactor core. The placement of the two reactor recirculation valves assures that the coolant level in the RPV is maintained above the core and the fuel remains covered at all times during ECCS operation.

The ECCS is designed such that no single failure prevents the system from performing its safety function including loss of onsite or offsite electrical power, initiation logic, and single active or passive component failure. The valves are the only active components in the ECCS and are designed to actuate on stored energy. After the actuation, the valves do not require a subsequent change of state or continuous availability of power to maintain their intended safety functions.

Leakage from the RCS to the CNV is detectable by containment pressure instruments, and instrumentation and operation records from the containment evacuation system.

Conformance or Exception

The NuScale Power Plant design conforms to PDC 35.

Relevant FSAR Chapters and Sections

Section 4.2	Fuel System Design
Section 6.3	Emergency Core Cooling System
Section 8.2	Offsite Power System
Section 8.3	Onsite Power Systems
Chapter 15	Transient and Accident Analyses

3.1.4.7 Criterion 36-Inspection of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Implementation in the NuScale Power Plant Design

The ECCS provides accessibility for appropriate periodic inspection of important components in accordance with ASME BPVC, Section III Division 1 to assure the integrity and capability of the system.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 36.

Relevant FSAR Chapters and Sections

Section 6.3	Emergency Core Cooling System

3.1.4.8 Criterion 37-Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Implementation in the NuScale Power Plant Design

The MPS provides the capability to perform periodic pressure and functional testing of the ECCS that ensures operability and performance of system components and the operability and performance of the system as a whole.

Functional testing of ECCS valves under conditions similar to design conditions is only possible with a differential pressure established between the RPV and the CNV because the main valve control chamber must vent to the CNV. These tests are therefore conducted under conditions that are colder than would exist for a required actuation of the ECCS valves and at a lower differential pressure.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 37.

Relevant FSAR Chapters and Sections

Section 3.9.6	Functional Design, Qualification and Inservice Testing of Pumps, Valves and Dynamic Restraints
Section 6.3	Emergency Core Cooling System

3.1.4.9 Criterion 38-Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Implementation in the NuScale Power Plant Design

The NuScale design supports an exemption from the power provisions of GDC 38. The following PDC has been adopted:

> A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of

other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that the system safety function can be accomplished, assuming a single failure.

Containment heat removal is an inherent characteristic assured by the materials and physical configuration of the CNV partially immersed in the UHS. The containment heat removal function is accomplished with the passive transfer of containment heat via the steel wall of the NuScale CNV to the UHS. The design configuration of the CNV and UHS provides the ability to remove containment heat rapidly for accident conditions to establish low containment pressure and temperature, and maintain these conditions for an indefinite period with no reliance on active components or electrical power.

During a postulated design basis loss-of-coolant or other conditions involving mass and energy release into containment, the released inventory is collected and accumulates within the CNV. The reactor coolant inventory condenses and accumulates in the CNV. The subsequent actuation of the ECCS establishes a natural circulation coolant pathway that circulates reactor coolant inventory through the CNV volume back to the RPV and through the reactor core.

Conformance or Exception

The NuScale Power Plant design conforms to PDC 38.

Relevant FSAR Chapters and Sections

Section 6.2.1 Containment Functional Design

Section 6.2.2 Containment Heat Removal

Section 8.2 Offsite Power System

Section 8.3 Onsite Power Systems

Section 9.2.5 Ultimate Heat Sink

3.1.4.10 Criterion 39-Inspection of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Implementation in the NuScale Power Plant Design

The major components that provide for the passive containment heat removal function are designed to allow inspections in accordance with in ASME BPVC, Section XI Division 1. The design permits appropriate periodic examination of the CNV to ensure continuing integrity and capability for heat transfer, i.e., the design allows for inspection of the surfaces for fouling or degradation that could potentially impede heat transfer to the UHS.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 39.

Relevant FSAR Chapters and Sections

Section 6.2.2 Containment Heat Removal

3.1.4.11 Criterion 40-Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Implementation in the NuScale Power Plant Design

The NPM passive containment cooling does not include or require active components to provide the containment heat removal function, thus periodic and operation testing specified by GDC 40 does not apply. Testing of the passive containment heat removal function for LOCA conditions was performed and showed that following a design basis event that results in containment pressurization, containment pressure is rapidly reduced and maintained below the design value without operator action. The continuing operability and performance of the containment heat removal function is ensured through periodic inspections, pursuant to GDC 39. Therefore, the underlying intent of GDC 40 is met.

Conformance or Exception

The NuScale Power Plant design does not conform to GDC 40. The NuScale design supports an exemption from the criterion.

Relevant FSAR Chapters and Sections

Section 3.9.6 Functional Design, Qualification and Inservice Testing of Pumps, Valves and Dynamic Restraints

Section 6.2.2 Containment Heat Removal

3.1.4.12 Criterion 41-Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Implementation in the NuScale Power Plant Design

The NuScale design supports an exemption from the power provisions of GDC 41. The following PDC has been adopted:

> Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.
>
> Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that its safety function can be accomplished, assuming a single failure.

For the NuScale design, there are no containment atmosphere cleanup systems necessary to ensure containment integrity or to reduce fission product release to the environment following postulated accidents. The CNV in conjunction with the containment isolation system is credited to mitigate the consequences of a design basis accident.

Compliance with GDC 41 is met with the NuScale passive design with respect to hydrogen and oxygen control/cleanup. The CNV can withstand the environmental conditions created by burning of hydrogen during the first 72 hours of design basis and beyond design basis accidents, while maintaining structural integrity and safe shutdown capability.

Natural aerosol removal mechanisms inherent in the containment design deplete elemental iodine and particulates in the containment atmosphere. The limited containment leakage and natural fission product control mechanisms result in offsite doses that are less than regulatory limits.

Conformance or Exception

The NuScale design reduces the concentration and quality of fission product release to the environment and ensures CNV integrity is maintained following a postulated design basis accident, thus meeting the intent of PDC 41.

Relevant FSAR Chapters and Sections

Section 6.2.5	Combustible Gas Control in the Containment Vessel
Section 6.5.3	Fission Product Control Systems
Section 8.2	Offsite Power System
Section 8.3	Onsite Power Systems

3.1.4.13 Criterion 42-Inspection of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Implementation in the NuScale Power Plant Design

The design does not include containment atmosphere cleanup systems which are subject to inspections of GDC 42.

Conformance or Exception

The NuScale Power Plant design does not include containment atmosphere cleanup systems which are subject to inspections of GDC 42 and therefore the criterion is not applicable.

Relevant FSAR Chapters and Sections

Section 6.5.3	Fission Product Control Systems

3.1.4.14 Criterion 43-Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Implementation in the NuScale Power Plant Design

The NuScale Power design does not include containment atmosphere cleanup systems which are subject to periodic pressure and functional testing of GDC 43.

Conformance or Exception

The NuScale Power Plant design does not include containment atmosphere cleanup systems which are subject to the periodic pressure and functional testing of GDC 43 and therefore the criterion is not applicable.

Relevant FSAR Chapters and Sections

Section 6.5.3 Fission Product Control Systems

3.1.4.15 Criterion 44-Cooling Water

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Implementation in the NuScale Power Plant Design

The NuScale design supports an exemption from the power provisions of GDC 44. The following PDC has been adopted:

> A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that the system safety function can be accomplished, assuming a single failure.

The cooling water function is provided by the UHS.

The UHS consists of the reactor pool, refueling pool, and spent fuel pool and functions as a cooling water medium for the decay heat removal heat exchangers, NPMs within the reactor pool, and the stored spent fuel assemblies. The UHS maintains the core temperature at acceptably low levels following any LOCA resulting in the initiation of ECCS. The passive cooling feature provided by the UHS does not include active components and does not rely on electrical power to perform its safety function.

The water level of the UHS is monitored by level instrumentation which provides a signal to the spent fuel pool cooling system for the addition of demineralized water as normal makeup when a low pool water level is detected.

Conformance or Exception

The NuScale Power Plant standard design conforms to PDC 44.

Relevant FSAR Chapters and Sections

Section 8.2 Offsite Power System

Section 8.3 Onsite Power Systems

Section 9.2.5 Ultimate Heat Sink

3.1.4.16 Criterion 45-Inspection of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Implementation in the NuScale Power Plant Design

The UHS does not include or require active components to perform its passive cooling function. Leak detection surveillance and level instrumentation are provided to monitor the integrity and capability of the UHS.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 45.

Relevant FSAR Chapters and Sections

Section 9.2.5 Ultimate Heat Sink

3.1.4.17 Criterion 46-Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Implementation in the NuScale Power Plant Design

The UHS requires no active components to perform the required safety functions. The UHS design permits the inspection of important components, such as the pool water level instrumentation, the pool liner, and the outside surfaces of the containment vessels. These inspections and tests assure the system integrity and capability of the UHS heat removal function.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 46.

Relevant FSAR Chapters and Sections

Section 9.2.5 Ultimate Heat Sink

3.1.5 Reactor Containment

3.1.5.1 Criterion 50-Containment Design Basis

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculation model and input parameters.

Implementation in the NuScale Power Plant Design

The CNV is designed to provide a final barrier against release of fission products while accommodating the calculated pressures and temperatures resulting from any design basis LOCA with sufficient margin such that the design leak rates are not exceeded. The CNV design also takes into consideration the pressures and temperatures associated with combustible gas deflagration. The design includes no internal sub-compartments to eliminate the potential for collection of combustible gases and differential pressures resulting from postulated high-energy pipe breaks within containment.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 50.

Relevant FSAR Chapters and Sections

Section 3.8.2 Steel Containment

Section 6.2 Containment Systems

Section 8.3 Containment Electrical Penetration Assemblies

3.1.5.2 Criterion 51-Fracture Prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

Implementation in the NuScale Power Plant Design

The design, fabrication, and construction materials for the CNV system includes sufficient margin to provide assurance that the containment pressure boundary will not undergo brittle fracture and the probability of rapidly propagating fracture will be minimized under operating, maintenance, and postulated accident conditions. The ferritic containment pressure boundary materials satisfy the fracture toughness criteria for ASME BPVC Section III Division 1, Class 1 and 2 components.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 51.

Relevant FSAR Chapters and Sections

Section 6.2.7 Fracture Prevention of Containment Vessel

3.1.5.3 Criterion 52-Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Implementation in the NuScale Power Plant Design

The CNV design allows testing and inspection, other than as anticipated by GDC 52, to assure CNV leakage integrity.

The CNV design utilizes 10 CFR 50, Appendix J, Type B and C tests to quantify containment leakage, thus assuring that the allowable leakage rate values are not exceeded.

Conformance or Exception

The NuScale Power Plant design does not conform to GDC 52. The NuScale design supports an exemption from the criterion.

Relevant FSAR Chapters and Sections

Section 6.2.6 Containment Leakage Testing

3.1.5.4 Criterion 53-Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Implementation in the NuScale Power Plant Design

The CNV is designed to allow for sufficient access for inservice inspection of vessel welds and penetrations, and surveillance testing of containment isolation valves (CIVs) and penetration assemblies pursuant to ASME BPVC, Section XI Division 1 and "Standards and Guides for Operation and Maintenance of Nuclear Power Plants," ASME OM-2012 (Reference 3.1-7).

Conformance or Exception

The NuScale Power Plant design conforms to GDC 53.

Relevant FSAR Chapters and Sections

Section 3.8.2 Steel Containment

Section 6.2.6 Containment Leakage Testing

3.1.5.5 Criterion 54-Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Implementation in the NuScale Power Plant Design

The piping systems that penetrate the CNV are designed with leak detection, isolation, and containment capabilities that are redundant and reliable. The containment isolation components include CIVs and passive containment isolation barriers that are periodically tested to ensure leakage is maintained within acceptable limits. The CIVs close for an ESFAS containment system isolation actuation signal, including when the

MPS detects low AC voltage. The closure times are designed to minimize release of containment atmosphere to the environment.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 54.

Relevant FSAR Chapters and Sections

Section 3.9.6	Functional Design, Qualification and Inservice Testing of Pumps, Valves and Dynamic Restraints
Section 5.2	Integrity of Reactor Coolant Boundary
Section 5.4	Reactor Coolant System Component and Subsystem Design
Section 6.2	Containment Systems

3.1.5.6 Criterion 55-Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Implementation in the NuScale Power Plant Design

The lines that are part of the RCPB and penetrate primary reactor containment are designed to provide adequate containment isolation. The RCS injection line, pressurizer spray supply line, and RCS discharge line, in addition to the reactor high point degasification line, are part of the RCPB and penetrate primary reactor containment. Consistent with GDC 55 except for the location of the isolation valves, two CIVs are provided for each of these lines and are located outside the CNV. Each line features a single-body, dual valve welded directly to a CNV top head nozzle safe-end to provide two containment isolation barriers in series. The isolation valves are Seismic Category 1 components and constructed in accordance with ASME BPVC, Section III, Division 1, Subsection NB.

Conformance or Exception

The NuScale Power design departs from GDC 55. The NuScale design supports an exemption for the lines that depart from the four alternatives for containment isolation valves specified in the criterion.

Relevant FSAR Chapters and Sections

Section 6.2.4 Containment Isolation System

Chapter 15 Transient and Accident Analyses

3.1.5.7 Criterion 56-Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Implementation in the NuScale Power Plant Design

The lines that connect directly to the containment atmosphere and penetrate primary reactor containment are designed to provide adequate containment isolation. The containment evacuation line and the containment flood and drain line connect directly to the containment atmosphere and penetrate primary reactor containment. The control rod drive closed loop cooling system supply and return lines penetrate primary reactor containment and are conservatively treated as if the lines connect directly to containment atmosphere. Consistent with GDC 56 except for the location of the isolation valves, two CIVs are provided for each of the lines and are located outside the CNV. The lines feature a single-body, dual valve welded directly to a containment top head nozzle safe-ends to provide two containment isolation barriers in series. The isolation valves are Seismic Category 1 components and constructed in accordance with ASME BPVC Section III Division 1, Subsection NB.

Conformance or Exception

The NuScale Power design departs from GDC 56. An exemption is provided for the lines that depart from the four alternatives for containment isolation valves specified in the criterion.

Relevant FSAR Chapters and Sections

Section 6.2.4 Containment Isolation System

3.1.5.8 Criterion 57-Closed System Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Implementation in the NuScale Power Plant Design

The lines that penetrate primary reactor containment and are neither part of the RCPB nor connected directly to the containment atmosphere are designed to provide adequate containment isolation. At least one CIV is provided for each of these lines, with exception of DHRS.

The CIV provided for each applicable main steam and feedwater line is a Seismic Category 1, ASME BPVC, Section III Division 1, Subsection NC, Class 2 valve. As noted in Section 3.1.5.7, for the RCCW return and supply lines, two CIVs are provided for each line in a single-body, dual valve. These valves are Seismic Category 1, ASME BPVC, Section III Division 1, Subsection NB, Class 1 components.

The DHRS lines penetrate containment and are neither part of the RCPB nor connected directly to the containment atmosphere. The DHRS is a closed system inside and outside containment and does not have CIVs. Two isolation barriers are provided by the direct connection of the closed-loop DHRS outside containment, and by the closed-loop inside of containment formed by the steam generator system within the RPV, and the connecting piping. The DHRS is a welded Seismic Category I, ASME BPVC, Section III Division 1, Subsection NC, Class 2 design with a design temperature and pressure rating equal to that of the RPV and meets the applicable criteria of NRC Branch Technical Position 3-4, Revision 2.

Conformance or Exception

The NuScale Power Plant design departs from GDC 57. The NuScale design supports an exemption for the lines that depart from the isolation barriers specified in the criterion.

Relevant FSAR Chapters and Sections

Section 5.4.3 Decay Heat Removal System

Section 6.2.4 Containment Isolation System

3.1.6 Fuel and Radioactivity Control

3.1.6.1 Criterion 60-Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Implementation in the NuScale Power Plant Design

The NuScale Power Plant is designed to control and minimize the release of radioactive materials in solid waste and gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation and AOOs. Alarm setpoints, design features, and automated isolation features ensure compliance with GDC 60 and that the limitations of 10 CFR 20 and 10 CFR 50, Appendix I are not exceeded.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 60.

Relevant FSAR Chapters and Sections

Section 9.1.3 Spent Fuel Pool Cooling and Cleanup System

Section 9.2 Water Systems

Section 9.3 Process Auxiliaries

Chapter 11 Radioactive Waste Management

Chapter 15 Transient and Accident Analyses

3.1.6.2 Criterion 61-Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Implementation in the NuScale Power Plant Design

The spent fuel pool cooling system cools the spent fuel assemblies stored in the fuel storage racks in the spent fuel pool for normal operating conditions. Water in the spent fuel pool shields the assemblies and normal makeup for evaporation is provided by the demineralized water system. The UHS performs the cooling and shielding functions under accident conditions. The pool cleanup system purifies the shared body of water in the spent fuel pool, the reactor pool, and the refueling pool that make up the UHS. This system has filters and demineralizers for pool water cleanup, and provisions for periodic sampling.

The large inventory of water in the UHS is a passive source of water that ensures the water level in the spent fuel pool remains above the stored spent fuel assemblies for weeks without additional makeup water to the UHS and without operation of the two active cooling systems. Section 9.2.5 describes performance of the UHS for accident conditions.

The area around the spent fuel pool is serviced by nonsafety-related Reactor Building heating and ventilation system, which controls the release of airborne radionuclides from evaporating UHS pool water for normal operating conditions. For accident conditions, the radiological consequences of a fuel handling accident are addressed in Chapter 15.

The piping penetrations through the walls of the UHS pool and the piping in the pool can not drain the water and adversely affect the inventory of water available for cooling and shielding the spent fuel assemblies.

The design of the spent fuel storage facility, the active pool cooling and cleanup systems, and the UHS satisfy GDC 61.

Permanent plant shielding is described in Section 12.3 and radiation monitoring is described in Section 11.5 and Section 12.3.

Chapter 11 describes the radioactive waste systems and the means provided to confine and filter radioactive material.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 61.

Relevant FSAR Chapters and Sections

Section 9.1	Fuel Storage and Handling
Section 9.2.5	Ultimate Heat Sink
Section 9.3.4	Chemical and Volume Control System
Section 9.4.2	Reactor Building and Spent Fuel Pool Area Ventilation System
Chapter 11	Radioactive Waste Management
Chapter 12	Radiation Protection
Chapter 15	Transient and Accident Analysis

3.1.6.3 Criterion 62-Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Implementation in the NuScale Power Plant Design

The design and controls for operation of the fuel handling equipment and fuel storage racks prevent an inadvertent criticality by use of geometrically safe configurations, as well as plant programs and procedures. Section 9.1 describes criticality safety for handling and storage of new and spent fuel assemblies.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 62.

Relevant FSAR Chapters and Sections

Section 9.1	Fuel Storage and Handling

3.1.6.4 Criterion 63-Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Implementation in the NuScale Power Plant Design

Monitoring for the loss of decay heat removal capability and excessive radiation levels is provided in the fuel storage and radioactive waste systems and associated handling areas for both normal and accident conditions. Information on cooling system performance is provided by the temperature detectors on the inlets and outlets of the heat exchangers in the spent fuel pool cooling system and reactor pool cooling system. The outlet temperature detectors have a high set point for an alarm that alerts operators to determine the cause and ensure adequate active cooling performance. Leakage from the liner in the UHS pools is collected by the pool leakage detection system and directed to sumps in the radioactive waste drain system for detection. Leakage from the piping and equipment in the pool cooling and cleanup systems is also collected by sumps in the radioactive waste drain system for detection. For normal and accident conditions, the UHS system provides redundant pool water level instruments. Radiation monitoring equipment is provided to detect excessive radiation levels and initiate appropriate alarms and procedural actions.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 63.

Relevant FSAR Chapters and Sections

Section 9.1.2	New and Spent Fuel Storage
Section 9.1.3	Spent Fuel Pool Cooling and Cleanup System
Section 9.3.2	Process Sampling System
Section 9.4.2	Reactor Building and Spent Fuel Pool Area Ventilation System
Section 11.5	Process and Effluent Radiation Monitoring Instrumentation and Sampling
Chapter 12	Radiation Protection

3.1.6.5 Criterion 64-Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Implementation in the NuScale Power Plant Design

The NuScale Power Plant provides means to monitor gaseous and liquid radioactivity releases resulting from normal operation, including AOOs, and from postulated accidents.

The primary coolant fluids are not required to be recirculated outside of containment following an accident. Radioactivity levels contained in the facility effluent and discharge paths and in the plant environs are monitored during normal and accident conditions by the radiation monitors.

Area radiation monitors supplement the personnel and area radiation survey provisions of the radiation protection program described in Section 12.5. Process and effluent radiation monitors provide alarm, indication, and archiving features to the main control room. These monitors provide the ability to measure and record the release of radioactive liquids and gases via the effluent release paths and into the plant environs.

Measurement capability and reporting of effluents are based on the guidelines of Regulatory Guides 1.183 and 1.21.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 64.

Relevant FSAR Chapters and Sections

Section 9.1.3 Spent Fuel Pool Cooling and Cleanup System

Section 9.2.2 Reactor Component Cooling Water System

Section 9.2.9 Utility Water Systems

Section 9.3 Process Auxiliaries

Section 9.4.2 Reactor Building and Spent Fuel Pool Area Ventilation System

Chapter 11 Radioactive Waste Management

Chapter 12 Radiation Protection

3.1.7 References

3.1-1 American Society of Mechanical Engineers, *Quality Assurance Requirements for Nuclear Facility Applications*, ASME NQA-1-2008/1a-2009 Addenda, New York, NY.

3.1-2 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Class 1 Components," 2013 edition, Section III Division 1, Subsection NB, New York, NY.

3.1-3 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Materials," 2013 edition, Section II, New York, NY.

3.1-4 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Class 2 Components," 2013 edition, Section III Division 1, Subsection NC, New York, NY.

3.1-5 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Components," 2013 edition, Section XI Division 1, New York, NY.

3.1-6 American Society for Testing and Materials, “Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels,” ASTM E185-1982, Philadelphia, PA.

3.1-7 American Society of Mechanical Engineers, "Standards and Guides for Operation and Maintenance of Nuclear Power Plants," ASME OM-2012, New York, NY.

3.2 Classification of Structures, Systems, and Components

Structures, systems, and components (SSC) are classified according to nuclear safety classification, seismic category, and quality group. This classification aids the determination of the appropriate quality standards and the identification of applicable codes and standards. SSC classification is based on a consideration of both safety-related function (consistent with the definition of safety related in 10 CFR 50.2) and risk significant functions determined as part of the design reliability assurance program. The design reliability assurance program process is described in Section 17.4.

SSC are classified as A1, A2, B1, and B2 in accordance with their safety and risk categories:

- A1 - SSC that are determined to be both safety-related and risk-significant
- A2 - SSC that are determined to be both safety-related and not risk-significant
- B1 - SSC that are determined to be both nonsafety-related and risk-significant
- B2 - SSC that are determined to be both nonsafety-related and not risk-significant

Certain nonsafety-related SSC that perform risk-significant functions require regulatory oversight. The required oversight is identified by the regulatory treatment of nonsafety systems (RTNSS) process as discussed in Section 19.3.

Table 3.2-1 provides the listing of SSC, including designation of classification, seismic category, and quality group. For the listed SSC, Table 3.2-1 also identifies applicable augmented design requirements and the applicable quality assurance program requirements. The systems are listed in Table 3.2-1 alpha-numerically by system codes. Within a given system, the SSC are listed, generally, in the order of the SSC classification (i.e., A1, A2, B1, and B2). Structures that are of conceptual design are listed within double brackets in Table 3.2-1.

Seismic and quality group classification is described in Section 3.2.1 and Section 3.2.2, respectively.

The SSC classification process is applied at the component level based upon the system functions performed. At the system level, system functions are designated as safety-related or nonsafety-related, and risk-significant or not risk-significant. Components are then classified commensurate with the safety and risk-significance of the system function(s) they support. A system that primarily performs safety-related or risk-significant functions may include nonsafety-related, not risk-significant components, on the basis of those components only supporting nonsafety-related, not risk-significant secondary system functions. Similarly, components that support multiple system functions may include multiple design features, each related to the different system functions. Components with any safety or risk design feature are classified on the basis of that feature.

Safety-related SSC and risk-significant SSC are subject to the Quality Assurance program requirements described in Section 17.5 and documented in the applicable quality assurance program column of Table 3.2-1. In addition, all or part of 10 CFR 50 Appendix B has been applied to some nonsafety-related SSC where specific regulatory guidance applies (e.g., Regulatory Guide (RG) 1.29). The application of 10 CFR 50, Appendix B to specific nonsafety-related SSC is included in Table 3.2-1.

In addition to safety and risk significance, the classification methodology includes consideration for “augmented” requirements for those SSC that are by definition nonsafety-related (based on the definition in 10 CFR 50.2). The selection of augmented requirements is based on a consideration of the important functionality to be performed by the nonsafety-related SSC and regulatory guidance applicable to the functionality (e.g., consistent with the functionality specified in General Design Criterion 60 for controlling radioactive effluents, augmented requirements are specified for radwaste systems based on the guidance in RG 1.143). Augmented design requirements, if applicable, are identified in Table 3.2-1.

The principal codes and standards used for the design of safety-related and risk-significant SSC are in accordance with the guidance of Regulatory Guide (RG) 1.26. If additional standards are invoked, they are noted in Table 3.2-1.

COL Item 3.2-1: A COL applicant that references the NuScale Power Plant design certification will update Table 3.2-1 to identify the classification of site-specific structures, systems, and components.

3.2.1 Seismic Classification

Seismic classification of SSC is consistent with the guidance of RG 1.29, “Seismic Design Classification for Nuclear Power Plants,” Revision 5, with the following exception. SSC that meet Staff Regulatory Guidance C.1.i are designated Seismic Category II rather than Seismic Category I consistent with industry precedent and practice. Seismic classification uses the following categories: Seismic Category I, Seismic Category II, Seismic Category III, and Seismic Category RW-IIa. These categories are described in Section 3.2.1.1, Section 3.2.1.2, Section 3.2.1.3, and Section 3.2.1.4, respectively.

Some nonsafety-related SSC are designated Seismic Category I as an augmenting requirement if the function is required following an earthquake.

In addition to RG 1.29, seismic categorization of SSC is also consistent with the guidance in RG 1.143 "Design Guidance For Radioactive Waste Management Systems, Structures, And Components Installed In Light-Water-Cooled Nuclear Power Plants"; and RG 1.189 "Fire Protection For Nuclear Power Plants."

RG 1.143 establishes design criteria for three different levels of radioactive waste content. The application of RG 1.143 with respect to radioactive waste management systems is discussed in Sections 11.2, 11.3 and 11.4. Seismic design expectations for radioactive waste management SSC are discussed in Section 3.2.1.4.

The seismic classification of instrumentation sensing lines is in accordance with RG 1.151, as discussed in Section 7.2.2 and in Section C.1.f of RG 1.29. The use of this guidance assures that the instrument sensing lines used to actuate or monitor safety-related functionality are appropriately classified as Seismic Category I and are capable of withstanding the effects of the SSE.

The design of fire protection systems in accordance with RG 1.189 is described in Section 9.5.1, and its classification is included in Table 3.2-1.

3.2.1.1 Seismic Category I

SSC classified as safety-related are designed to be capable of performing their safety functions during and following a safe shutdown earthquake (SSE). Therefore, these safety-related SSC, including their foundations and supports, are classified as Seismic Category I.

Some SSC classified as nonsafety-related are also designed to be capable of performing their nonsafety-related functions during and following an SSE. These nonsafety-related SSC, including their foundations and supports, are also classified as Seismic Category I.

Seismic Category I SSC are designed to withstand the seismic loads associated with the SSE, in combination with other designated loads, without loss of function or pressure integrity. Development of SSE seismic design loads is addressed in Section 3.7. The design of Seismic Category I structures is addressed in Section 3.8. The seismic design of mechanical systems and components is addressed in Section 3.9. The seismic qualification of mechanical and electrical equipment, including their supports, is addressed in Section 3.10.

Use of Seismic Category I piping is minimized in the NuScale Power Plant design. Drain lines, vent lines, fill lines, and test lines coming off the Seismic Category I piping are treated as part of the Seismic Category I piping.

For systems that are partially Seismic Category I, the Category I portion of the system extends to the first seismic restraint beyond the isolation valves that isolate the part that is Seismic Category I from the non-seismic portion of the system.

At the interface between Seismic Category I and non-seismic systems, the Seismic Category I dynamic analysis requirements are extended to either the first anchor point in the non-seismic system or a sufficient distance into the non-Seismic Category I system so that the Seismic Category I analysis remains valid.

Safety-related and nonsafety-related, Seismic Category I SSC are subject to the pertinent quality assurance program requirements of 10 CFR 50, Appendix B.

3.2.1.2 Seismic Category II

The design requirements in Staff Regulatory Guidance C.1.i in RG 1.29 for protection of Seismic Category I SSC are applied as follows to SSC classified as Seismic Category II. SSC that perform no safety-related function, but whose structural failure or adverse interaction could degrade the functioning or integrity of a Seismic Category I SSC to an unacceptable level or could result in incapacitating injury to occupants of the control room during or following an SSE, are designed and constructed so that the SSE would not cause such failure. These SSC are classified as Seismic Category II.

Because they are not required to remain functional, the Seismic Category II classification is applied only to the portions of systems where a potential for adverse interaction with a Seismic Category I SSC exists. Additionally, nonsafety-related instrument lines from safety related pressure boundaries are required to maintain pressure integrity.

Seismic Category II SSC are subject to the pertinent quality assurance program requirements of 10 CFR 50, Appendix B as noted in Table 3.2-1.

3.2.1.3 Seismic Category III

SSC not classified as Seismic Category I or Seismic Category II are classified as Seismic Category III. This category includes SSC that have no seismic design requirements and SSC that may be subject to seismic design criteria that are incorporated in, or invoked by, an applicable commercial or industry code.

3.2.1.4 Safety Classification RW-IIa

RG 1.143 establishes design criteria for SSC that contain radioactive waste. Within RG 1.143 SSC are grouped based upon the quantity of radioactive material. Specifically, RG 1.143 uses three classifications: RW-IIa, RW-IIb, and RW-IIc. These design criteria are applied in addition to the seismic categorization. Therefore a SSC that is used for radioactive waste must satisfy both criteria. There are no Seismic Category I SSC that have RG 1.143 design requirements. There is one Seismic Category II SSC that does. The Radioactive Waste Building is Seismic Category II due to its proximity to the Reactor Building, and it is RW-IIa due to its design radioactive material content.

RG 1.143 specifies that RW-IIa SSC are designed to withstand ½ of the SSE. As such, the Radioactive Waste Building is designed to both remain intact (satisfying Seismic Category II) when subjected to a full SSE; and intact and functional (satisfying RW-IIa) when subjected to an earthquake with half the force of the SSE.

All other radioactive waste SSC are sufficiently separated from Seismic Category I SSC that they are Seismic Category III.

RG 1.143 classification is included in Table 3.2-1 within the Quality Class column. SSC that are classified as RW-IIb and RW-IIc are designed to industry codes and standards, which conforms with Seismic Category III.

3.2.2 System Quality Group Classification

Quality group A through D classifications of relevant SSC are performed in accordance with the applicable guidance of RG 1.26 and RG 1.143. Refer to Table 3.2-1 for a listing of the identified classifications.

The quality group boundaries are included on piping and instrument drawings as the third character (Code Identifier) in the Piping Line Class Specification Convention. Code Identifiers A - C correspond to ASME Class 1 through 3 and align with quality groups A - C. Code identifier D corresponds to Quality Group D as described in RG 1.26.

Safety-related instrument sensing lines are designed and constructed in accordance with ANSI/ISA-67.02.01-1999 (Reference 3.2-2) as described in RG 1.151. The standard ANSI/ISA-67.02.01-1999 establishes the applicable code requirements and code boundaries for the design and installation of instrument sensing lines interconnecting safety-related piping and vessels with both safety-related and nonsafety-related instrumentation. This is further discussed in Section 7.2.2.

The following subsections also describe the codes and standards applicable to supports for Quality Group A, B, C, and D components. The reactor vessel internals (see Section 3.9.5) and steam generator supports and tube supports (see Section 5.4.1.5) comply with the design and construction requirements of Subsection NG of Section III, Division 1 of the ASME BPVC (Reference 3.2-1).

3.2.2.1 Quality Group A

Quality Group A applies to pressure-retaining components that form part of the reactor coolant pressure boundary, except those that can be isolated from the reactor coolant system by two automatically-closed or normally-closed valves in series.

Quality Group A SSC meet the requirements for Class 1 components in Section III, Division 1 of the ASME BPVC (Reference 3.2-1) and applicable conditions promulgated in 10 CFR 50.55a(b). Supports for Quality Group A SSC meet the requirements for Class 1 supports in Section III, Division 1, Subsection NF of the ASME BPVC and are not separately listed in Table 3.2-1. Exceptions exist for supports within the pressure retaining boundary of the RPV. See Section 3.2.2 and Section 5.4.1.5 for additional information.

The remaining portions of the reactor coolant pressure boundary are in Quality Group B.

3.2.2.2 Quality Group B

Quality Group B applies to water- and steam-containing pressure vessels, heat exchangers (other than turbines and condensers), storage tanks, piping, pumps, and valves that are:

- part of the reactor coolant pressure boundary but are excluded from Quality Group A.
- safety-related or risk-significant systems or portions of systems that are designed for (i) emergency core cooling, (ii) post-accident containment heat removal, or (iii) post-accident fission product removal.
- safety-related or risk-significant systems or portions of systems that are designed for (i) reactor shutdown or (ii) residual heat removal.
- portions of the steam and feedwater systems extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation.
- systems or portions of systems connected to the reactor coolant pressure boundary that cannot be isolated from that boundary during all modes of operation by two normally closed or automatically closable valves.

Quality Group B SSC meet the requirements for Class 2 components in Section III, Division 1 of the ASME BPVC and applicable conditions promulgated in 10 CFR 50.55a(b). Supports for Quality Group B SSC meet the requirements for Class 2 supports in Section III, Division 1, Subsection NF of the ASME BPVC and are not separately listed in Table 3.2-1.

3.2.2.3 Quality Group C

Quality Group C applies to water-, steam-, and radioactive-waste-containing pressure vessels; heat exchangers (other than turbines and condensers); storage tanks; piping; pumps; and valves that are not part of the reactor coolant pressure boundary or included in Quality Group B but part of the following:

- safety-related or risk-significant portions of cooling water and auxiliary feedwater systems that are designed for (i) emergency core cooling, (ii) postaccident containment heat removal, (iii) postaccident containment atmosphere cleanup, or (iv) residual heat removal from the reactor and spent fuel storage pool that (i) do not operate during any mode of normal reactor operation and (ii) cannot be tested adequately
- safety-related or risk-significant portions of cooling water and seal water systems that are designed to support the functioning of other safety-related or risk-significant systems and components
- portions of systems that are connected to the reactor coolant pressure boundary and capable of being isolated from that boundary by two valves during all modes of normal reactor operation
- systems other than radioactive waste management systems that may contain radioactive material and whose postulated failure would result in conservatively calculated potential off-site doses that exceed 0.5 rem to the whole body or its equivalent to any part of the body

Quality Group C SSC meet the requirements for Class 3 components in Section III, Division 1 of the ASME BPVC and applicable conditions promulgated in 10 CFR 50.55a(b). Supports for Quality Group C SSC meet the requirements for Class 3 supports in Section III, Division 1, Subsection NF of the ASME BPVC and are not separately listed in Table 3.2-1.

3.2.2.4 Quality Group D

Quality Group D applies to water and steam-containing components that are not part of the reactor coolant pressure boundary or included in Quality Groups B or C, but are part of systems or portions of systems that contain or may contain radioactive material (and are not radioactive waste management systems).

SSC determined to be Quality Group D in accordance with guidance of RG 1.26 are listed in Table 3.2-1. SSC designated as Quality Group D meet the codes and standards for components identified as applicable for Quality Group D in Table 1 of RG 1.26. Codes and standards for Quality Group D SSC and their supports are as follows:

- Pressure Vessels – ASME BPVC, Section VIII (Reference 3.2-3)
- Piping and Valves – ASME B31.1, Power Piping (Reference 3.2-4)
- Pumps – Manufacturers' standards

- Atmospheric Storage Tanks – API-650 (Reference 3.2-5) or AWWA D-100 (Reference 3.2-6)
- 0-15 psig Storage Tanks – API-620 (Reference 3.2-7)

3.2.3 References

3.2-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Rules for Construction of Nuclear Facility Components," 2013 edition, Section III, New York, NY.

3.2-2 American National Standards Institute/International Society of Automation "Nuclear Safety-Related Instrument-Sensing Line Piping and Tubing Standard for Use in Nuclear Power Plants," ANSI/ISA 67.02.01-1999, Research Triangle Park, NC.

3.2-3 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Rules for Construction of Pressure Vessels," Section VIII, Division 1, New York, NY.

3.2-4 American Society of Mechanical Engineers, "Power Piping," ASME B31.1, New York, NY.

3.2-5 American Petroleum Institute, "Welded Steel Tanks for Oil Storage," API 650, 12th edition, 2013, Washington, DC.

3.2-6 American Water Works Association, "Welded Steel Tanks for Water Storage," AWWA D-100, Denver, Colorado.

3.2-7 American Petroleum Institute, "Design and Construction of Large, Welded, Low-pressure Storage Tanks," API 620, 12th edition, 2014, Washington, DC.

Table 3.2-1: Classification of Structures, Systems, and Components

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
CNTS, Containment System							
All components (except as listed below)	RXB	A1	N/A	Q	None	B	I
• CVC Injection Check Valve • CVC Discharge Excess Flow Check Valve • CVC PZR Spray Check Valve	RXB	B2	None	AQ-S	None	C	I
• CVC Injection & Discharge Nozzles • CVC PZR Spray Nozzle • CVC PZR Spray CIV • CVC RPV High Point Degasification Nozzle • CVC RPV High Point Degasification CIV • RVV & RRV Trip/Reset # 1 & 2 Nozzles • RVV Trip 1 & 2/Reset #3 Nozzles • CVC Injection & Discharge CIVs	RXB	A1	N/A	Q	None	A	I
• NPM Lifting Lugs • Top Support Structure • Top Support Structure Diagonal Lifting Braces	RXB	B1	None	AQ-S	• ANSI N14.6 • NUREG-0612	N/A	I
• CNV Fasteners • Hydraulic skid • CNV Seismic Shear Lug • CNV CRDM Support Frame • Containment Pressure Transducer (Narrow Range) • Containment Water Level Sensors (Radar Transceiver) • SG 1 & 2 Steam Temperature Sensors (RTD)	RXB	A1	N/A	Q	None	N/A	I
CNTS CFDS Piping in containment	RXB	B2	None	AQ-S	None	B	II
Piping from (CES, CFDS, FWS, MSS, and RCCWS) CIVs to disconnect flange (outside containment)	RXB	B2	None	AQ-S	None	D	I
CVCS Piping from CIVs to disconnect flange (outside containment)	RXB	B2	None	AQ-S	None	C	I
CIV Close and Open Position Sensors: • CES, Inboard and Outboard • CFDS, Inboard and Outboard • CVCS, Inboard and Outboard PZR Spray Line • CVCS, Inboard and Outboard RCS Discharge • CVCS, Inboard and Outboard RCS Injection • CVCS, Inboard and Outboard RPV High-Point Degasification • RCCWS, Inboard and Outboard Return and Supply • SGS, Steam Supply CIV/MSIVs and CIV/MSIV Bypasses	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
CIV Close and Open Position Indication • FWS, Supply to SGs and DHR HXs FWIV	RXB	A1	None	Q	None	N/A	I
Containment Pressure Transducer (Wide Range)	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• Containment Air Temperature (RTDs) • FW Temperature Transducers	RXB	B2	None	AQ-S	None	N/A	II
SGS, Steam Generator System							
• SG tubes • Integral steam plenums • Feedwater plenums	RXB	A1	N/A	Q	None	A	I
• SG tube supports • Upper and lower SG supports	RXB	A1	N/A	Q	None	N/A	I

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
• Steam piping inside containment • Feedwater piping inside containment • Feedwater supply nozzles • Main steam supply nozzles • Thermal relief valves • Feedwater plenum access port covers • Steam plenum access port covers	RXB	A2	N/A	Q	None	B	I
Flow restrictors	RXB	A2	N/A	Q	None	N/A	I
RXC, Reactor Core System							
Fuel assembly (RXF)	RXB	A1	N/A	Q	None	N/A	I
Fuel Assembly Guide Tube	RXB	A2	N/A	Q	None	N/A	I
Incore Instrument Tube	RXB	B2	None	AQ-S	None	N/A	I
CRDS, Control Rod Drive System							
• Control Rod Drive Shafts • Control Rod Drive Latch Mechanism	RXB	A1	N/A	Q	None	N/A	I
CRDM Pressure Boundary (Latch Housing, Rod Travel Housing, Rod Travel Housing Plug)	RXB	A2	N/A	Q	None	A	I
CRDS Cooling Water Piping and Pressure Relief Valve	RXB	B2	None	AQ-S	None	B	II
Rod Position Indication (RPI) Coils	RXB	B2	None	AQ-S	None	N/A	I
• Control Rod Drive Coils • CRDM power cables from EDN breaker to MPS breaker • CRDM power cables from MPS breaker to CRDM Cabinets	RXB	B2	None	AQ-S	None	N/A	II
• CRDM Control Cabinet • CRDM Power & Rod Position Indication Cables • Rod Position Indication Cabinets (Train A/B)	RXB	B2	None	AQ	None	N/A	III
CRA, Control Rod Assembly							
All components	RXB	A2	N/A	Q	None	N/A	I
NSA, Neutron Source Assembly							
All components	RXB	B2	None	AQ-S	None	N/A	I
RCS, Reactor Coolant System							
All components (except as listed below)	RXB	A1	N/A	Q	None	A	I
• Reactor vessel internals (upper riser assembly (Note 7), lower riser assembly, core support assembly, flow diverter, and pressurizer spray nozzles)	RXB	A1	None	Q	None	N/A	I
• Reactor vessel internals upper riser bellows-lateral seismic restraining structure	RXB	A1	N/A	Q	None	N/A	I
• Reactor vessel internals upper riser bellows-vertical expansion structure • Narrow Range Pressurizer Pressure Elements • PZR/RPV Level Elements • Narrow Range RCS Hot Leg Temperature Elements • Wide Range RCS Hot Leg Temperature Elements • RCS Flow Transmitters (Ultrasonic)	RXB	B2	N/A	AQ-S	ASME BPVC Section III Division 1 NG guidance	N/A	II
• Wide Range RCS Pressure Elements • Wide Range RCS Cold Leg Temperature Elements	RXB	A2	N/A	Q	None	N/A	I
Reactor Safety Valve Position Indicator	RXB	B2	None	AQ-S	Environmental Qualification Power from EDS	N/A	I
• PZR Control Cabinet • PZR Vapor Temperature Element • PZR heater power cabling from MPS breaker to PZR heaters • Pressurizer Liquid Temperature Element • Narrow Range RCS Cold Leg Temperature Element	RXB	B2	None	AQ-S	None	N/A	II
PZR heater power cabling from ELV breaker to MPS breaker	RXB	B2	None	None	None	N/A	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
CVCS, Chemical and Volume Control System							
DWS Supply Isolation Valves	RXB	A2	N/A	Q	None	C	I
Position Indication for DWS Supply Isolation Valves	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• Discharge Spoolpiece Drain Valve • Discharge Spoolpiece Isolation Valve • Injection Spoolpiece Drain Valve • Pressurizer Spoolpiece Drain Valve • NuScale Power Module Removable Spoolpieces • RPV High Point Degasification Isolation Valve • RPV High Point Degasification Spoolpiece Drain Valve	RXB	B2	None	AQ-S	None	C	I
Hydrogen bottle and distribution assembly including excess flow valve	RXB	B2	None	AQ-S	None	D	II
Pressure Indicating Transmitter for Hydrogen Injection Bottle	RXB	B2	None	AQ-S	None	N/A	II
• Mass Flow Instruments for CVC Injection Line • CVC Discharge Line, • CVC Makeup Line, • LRW Letdown Line (Pressure, Temperature, Flow)	RXB	B2	None	AQ	None	N/A	III
• Other Instrumentation (Pressure, Temperature, Flow, Radioactivity, Boron)	RXB	B2	None	None	None	N/A	III
All other components	RXB	B2	None	None	None	D	III
BAS, Boron Addition System							
All components (except as listed below)	RXB	B2	None	None	None	D	III
• Instrumentation (Pressure, Temperature, Flow, Level, Position) • Hopper Scale • Batch Tank Mixer	RXB	B2	None	None	None	N/A	III
MHS, Module Heatup System							
All components (except as listed below)	RXB	B2	None	None	None	D	III
• Instrumentation (Pressure, Temperature, Level)	RXB	B2	None	None	None	N/A	III
ECCS, Emergency Core Cooling System							
• Reactor Vent Valve (RVV) • RVV Trip Valve • Reactor Recirculation Valve (RRV) • RRV Trip Valve • Reset Valve • Hydraulic lines	RXB	A1	N/A	Q	None	A	I
• RRV Position Indication • RVV Position Indication • Trip Valve Position Indication	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
Reset Valve Position Indication	RXB	B2	None	AQ-S	None	N/A	II
DHRS, Decay Heat Removal System							
SG Steam Pressure Instrumentation (4 per side)	RXB	A1	N/A	Q	None	N/A	I
• Actuation Valve (2 per side) • Condenser (1 per side)	RXB	A2	N/A	Q	None	B	I
• Condenser Outlet Pressure Instrumentation (3 per side) • Condenser Outlet Temperature Instrumentation (2 per side) • Valve Position Indicator (2 for open, 2 for close per side)	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
Level Instrument (2 per side)	RXB	B2	None	AQ-S	None	N/A	II
CRHS, Control Room Habitability System							
All components (except as listed below)	CRB	B2	None	AQ-S	None	N/A	I
• Air Supply Isolation Solenoid Valve Position Indicators • CRE Pressure Relief Isolation Valve Position Indicators	CRB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
• CRE Differential Pressure Transmitters • CRH Bottle Pressure Instruments • Flow Transmitters • Pressure Reducing Valve Pressure Indicators	CRB	B2	None	AQ-S	None	N/A	II
Air compressor and dryer	CRB	B2	None	None	None	N/A	III
CRVS, Normal Control Room HVAC							
All components (except as listed below)	CRB	B2	None	None	None	N/A	III
CRE Isolation Damper Position	CRB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• CRE Isolation Dampers • Fire and Smoke Dampers supporting the MCR • Radiation Monitors (Downstream of charcoal filter unit)	CRB	B2	None	AQ-S	None	N/A	I
Outside Air intake Smoke Detectors	CRB	B2	None	AQ-S	None	N/A	I
• Outside air Isolation Damper Position • Toxic gas detectors	CRB	B2	None	AQ-S	RG 1.78	N/A	I
Outside Air Isolation Dampers for CRV Recirculation Mode	CRB	B2	None	AQ-S	• RG 1.78 • RG 1.140 • Backup diesel powered • Charcoal and HEPA filtered • Maintain Positive Pressure	N/A	II
Ductwork and Associated Components (grilles, etc.) associated with the outside air intake up to the radiation monitors downstream of the filter unit	CRB	B2	None	AQ-S	• RG 1.78 • RG 1.140 • Charcoal and HEPA filtered • Maintain Positive Pressure	N/A	II
Radiation Monitors (upstream of charcoal filter unit)	CRB	B2	None	AQ	• Backup diesel powered • Charcoal and HEPA filtered • Maintain Positive Pressure	N/A	III
• CRV Filter Unit • CRV Supply Air Handling Unit A/B • Ductwork and Associated Components (dampers, grilles, etc.) associated with the MCR or TSC • Isolation Dampers for CRV Filter Unit Bypass	CRB	B2	None	AQ	• RG 1.140 • Backup diesel powered • Charcoal and HEPA filtered • Maintain Positive Pressure	N/A	III
• CRV Battery Exhaust Fan A/B • Temperature Sensors, Room Mounted	CRB	B2	None	AQ	None	N/A	III
RBVS, Reactor Building HVAC							
All components (except as listed below)	RXB, RWB	B2	None	None	None	N/A	III
• RBV Supply AHUs • RBV General Area Exhaust Fans • RBV General Area Exhaust Filter Units • Hot Lab Exhaust Fan	RXB RWB RWB RXB	B2	None	AQ	• RG 1.140	N/A	III
Ductwork and Associated Components (Dampers, grilles, etc) (except for SFP exhaust components)	RXB, RWB	B2	None	AQ	None	N/A	III
• RBV SFP Exhaust Ductwork and associated components (dampers, grills, etc.) • RBV SFP Exhaust Filter Units, including fans	RWB RWB	B2	None	AQ	• RG 1.140	N/A	III
Instrumentation	RXB, RWB	B2	None	AQ	• ANSI N13.1 • ANSI N42.18-2004 • ANSIHPS N13.1-2001 • Environmental Qualification • IEEE 497-2002 with CORR 1 • RG 1.140 • Table 1 of SRP 11.5	N/A	III
LRWS, Liquid Radioactive Waste System							
Degasifiers	RXB	B2	None	AQ	None	RW-IIa	RW-IIa
LCW Collection Tanks	RWB	B2	None	AQ	None	RW-IIb	RW-IIb

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
• Non-Radioactivity Indicating Instrumentation • Drum Dryer • LRW In-line Grab Samplers	RWB, RXB RWB RWB, RXB	B2	None	None	None	N/A	III
Radioactivity Indicating Transmitter	RWB, RXB	B2	None	AQ	ANSI N42.18-2004	N/A	III
All other components	RWB, RXB	B2	None	AQ	None	RW-IIc	III
GRWS, Gaseous Radioactive Waste System							
• Charcoal Guard Bed • Charcoal Decay Beds	RWB	B2	None	AQ	None	RW-IIa	RW-IIa
• Charcoal Drying Heater • Inlet Gas Sampler	RWB	B2	None	None	None	N/A	III
Radiation Indicating Transmitter	RWB	B2	None	AQ	ANSI N13.1-2011	N/A	III
All other components	RWB	B2	None	AQ	None	RW-IIc	III
SRWS, Solid Radioactive Waste System							
Spent Resin Storage Tanks	RWB	B2	None	AQ	None	RW-IIa	RW-IIa
Phase Separator Tanks	RWB	B2	None	AQ	None	RW-IIb	RW-IIb
• Instrumentation • Compactor • In-Line Grab Sampler	RWB	B2	None	None	None	N/A	III
All other components	RWB	B2	None	AQ	None	RW-IIc	III
RWDS, Radioactive Waste Drain System							
All components	RWB, RXB, ANB	B2	None	None	None	D	III
RWBVS, Rad-Waste Building HVAC System							
• Ductwork and Associated Components (Dampers, grilles, etc.) • RXB Exhaust Fan • Instrumentation • RWB Supply Air Handling Unit • RWB Supply Air Fans A/B	RWB	B2	None	AQ	• RG 1.140	N/A	III
All other components	RWB	B2	None	None	None	N/A	III
MAE, Module Assembly Equipment							
• Module Inspection Rack • Module Upender	RXB	B2	None	AQ-S	None	N/A	II
Module Import Trolley	RXB	B2	None	None	None	N/A	III
MAEB, Module Assembly Equipment - Bolting							
RFT Support	RXB	B2	N/A	Q	None	C	I
CNV Support Stand	RXB	B2	None	AQ-S	None	N/A	II
All other components	RXB	B2	None	None	None	N/A	III
FHE, Fuel Handling Equipment							
Fuel Handling Machine	RXB	B2	None	AQ-S	• ANSI/ANS 57.1-1992 • NUREG-0554 • ASME NOG-1	N/A	I
• New Fuel Elevator • New Fuel Jib Crane	RXB	B2	None	AQ-S	None	N/A	II
SFSS, Spent Fuel Storage System							
Spent Fuel Storage Rack	RXB	B2	None	AQ-S	• ANSI/ANS 57.1-1992 • ANSI/ANS 57.2-1983 with additions, clarifications, and exceptions of RG 1.13 • ANSI/ANS 57.3	N/A	I

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
SFPCS, Spent Fuel Pool Cooling System							
• Pumps • Strainers • Valves - (PCUS boundary isolation valves)	RXB	B2	None	AQ	ANSI/ANS 57.2-1983 with additions, clarifications, and exceptions of RG 1.13	D	III
• Flow control orifices • Instrumentation (pressure, temperature, flow, position)	RXB	B2	None	None	None	N/A	III
All other components	RXB	B2	None	None	None	D	III
PCUS, Pool Cleanup System							
All components (except as listed below)	RXB	B2	None	AQ	ANSI/ANS 57.2-1983 with additions, clarifications, and exceptions of RG 1.13	D	III
Instrumentation (Conductivity)	RXB	B2	None	AQ	ANSI/ANS 57.2-1983 with additions, clarifications, and exceptions of RG 1.13	N/A	III
Instrumentation (pressure, temperature, flow, position)	RXB	B2	None	None	None	N/A	III
• Sample Points • Instrumentation (pressure, temperature, flow, position)	RXB	B2	None	None	None	D	III
RPCS, Reactor Pool Cooling System							
• Sample Points • Valves - (PCUS boundary isolation valves)	RXB	B2	None	AQ	ANSI/ANS 57.2-1983 with additions, clarifications, and exceptions of RG 1.13	D	III
• Instrumentation - Boundary Valve Position	RXB	B2	None	AQ	ANSI/ANS 57.2-1983 with additions, clarifications, and exceptions of RG 1.13	N/A	III
• Heat Exchangers • Reactor Pool Cooling Pumps • Strainers • Valves (not listed above) - MOV, Air operated, Check, Manual, Relief	RXB	B2	None	None	None	D	III
• Instrumentation (not listed above) - Flow, Position, Pressure, Temperature • Orifices	RXB	B2	None	None	None	N/A	III
Instrumentation - Temperature (PAM D Variable)	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
PSCS, Pool Surge Control System							
• RXB Penetrations - Piping • Pool Penetrations - Piping	RXB RXB	B2	None	AQ-S	None	D	II
Tank Vent RE	Yard	B2	None	AQ	ANSI N42.18-2004	N/A	III
All other components	RXB, Yard	B2	None	None	None	D	III
UHS, Ultimate Heat Sink							
UHS Pool (water only; also see RXB and RBCM below)	RXB	A1	N/A	Q	None	N/A	N/A
Pool Level Instruments	RXB	B2	None	AQ-S	• IEEE 497-2002 with CORR 1 • NRC Order EA-12-051 • NEI 12-02 • NEI 12-06 (Order EA-12-049)	N/A	I
Water M/U Line	RXB	B2	None	AQ-S	• NRC Order EA-12-051 • NEI 12-02	C	I
PLDS, Pool Leakage Detection System							
All components	RXB	B2	None	None	None	D	III
CES, Containment Evacuation System							
Vacuum Pump Suction Pressure Indicators	RXB	B2	None	AQ-S	None	N/A	I
All other components (except as listed below)	RXB	B2	None	AQ	Quality Group D	D	III
CES instrumentation (except as listed below)	RXB	B2	None	None	N/A	N/A	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
Radiation Monitor	RXB	B2	None	AQ	• ANSI N42.18-2004 • ANSI/HPS N13.1-2011 • Table 1 of SRP 11.5 • Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	N/A	III
Sample Vessel Radiation Transmitter	RXB	B2	None	AQ	• ANSI N42.18-2004 • Table 1 of SRP 11.5	N/A	III
Gas Discharge Radiation Transmitter	RXB	B2	None	AQ	• ANSI/HPS N13.1-2011 • Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	N/A	III
• PSS Sample Panel Inlet and Outlet Isolation Valves • Vacuum Pump Bypass Valve	RXB	B2	None	AQ	Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	D	III
• Charcoal Pre-Filter • Charcoal Filter • Discharge Filter	RXB	B2	None	AQ	• RG 1.140	D	III
• Containment Service Air Pressure Valve • Sample Vessel Drain Sampler	RXB	B2	None	None	None	D	III
CFDS, Containment Flooding And Drain System							
All components (except as listed below)	RXB	B2	None	None	None	D	III
CFD Module Post Accident Monitoring Return Valves	RXB	B2	None	AQ	Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	D	III
Radiation Transmitter	RXB	B2	None	AQ	ANSI N42.18-2004	N/A	III
RCCWS, Reactor Component Cooling Water System							
All components (except as listed below)	RXB	B2	None	None	None	D	III
Radioactivity Transmitters for: • RCCW CE Vacuum Pumps and Condensers • RCCW CVC NRHXs and PSS Coolers • RCCW PSS Cooling Water TCU	RXB	B2	None	AQ	ANSI N42.18-2004	N/A	III
RCCWS instrumentation	RXB	B2	None	None	None	N/A	III
PSS, Process Sampling System							
All components (except as listed below)	RXB, TGB	B2	None	None	None	N/A	III
Reactor coolant discharge sample line isolation valve	RXB	B2	None	AQ	ANSI N13.1	D	III
Primary sampling system analysis panel	RXB	B2	None	AQ	ANSI N13.1	N/A	III
• Containment evacuation system sample line isolation valve	RXB	B2	None	AQ	• ANSI N13.1 • Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	D	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
• Containment sampling system sample panel	RXB	B2	None	AQ	• ANSI N13.1 • RG 1.7 • Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	D	III
• Primary sampling system sample cooler cooling water chillers	RXB	B2	None	AQ	Quality Group D	D	III
• Combined polisher effluents sample line isolation valve • Condensate polisher sample line isolation valves • Condensate pump discharge sample line isolation valve • Condenser hotwell sample line isolation valve • Feedwater sample line isolation valves • Main Steam bypass sample line isolation valves • Main steam sample line isolation valves	TGB	B2	None	None	None	D	III
MSS, Main Steam System							
• Start-up Isolation Valves • RXB Steam Traps	RXB	B2	None	AQ-S	None	D	I
• Secondary Main Steam Isolation Valves (Note 6) • Secondary Main Steam Isolation Bypass Valves (Note 6)	RXB	B2	None	AQ-S	• Technical Specification Surveillance for operability and inservice testing. • Valve Leak Detection	D	I
• Secondary Main Steam Isolation Bypass Valve Close and Open Position Indicators • Secondary Main Steam Isolation Valve Close and Open Position Indicators	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• Auxiliary Steam Supply Valve • Auxiliary Steam Warm-up Valve • Main Steam Safety Valves • Main Steam Vent Valve • N2 Injection Isolation Valves • Steam Sample Panel Isolation Valve • TGB Steam Traps	TGB TGB Yard TGB RXB TGB TGB	B2	None	None	None	D	III
• Main Steam Flow Transmitters • Main Steam Radiation Monitors	RXB, TGB	B2	None	AQ	• IEEE 497-2002 with CORR 1 • ANSI N42.18-2004 (Radiation Monitors)	N/A	III
• Main Steam Pressure Transmitters • Main Steam Temperature Elements	RXB, TGB	B2	None	AQ	None	N/A	III
All other components	RXB, TGB	B2	None	None	None	N/A	III
FWS, Condensate and Feedwater System							
All components (except as listed below)	TGB, RXB	B2	None	None	None	N/A	III
Feedwater Regulating Valve A/B (Note 6)	RXB	B2	None	AQ-S	Technical Specification Surveillance for operability and inservice testing.	D	I
Feedwater Supply Check Valve (Note 6)	RXB	B2	None	AQ-S	Inservice Testing	D	I
Feedwater Regulating Valve Accumulators	RXB	B2	None	AQ	Technical Specification Surveillance for operability and inservice testing.	D	III
Feedwater Regulating Valve A/B Limit Switch	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• Condensate Storage Tank (located adjacent to TCB) • Condensate Storage Tank Makeup Level Control Valve	Yard	B2	None	None	None	D	III
Steam Generator Differential Pressure Transmitter	RXB	B2	None	AQ	None	N/A	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
• Condensate Header Emergency Rejection Level Control Valve • Condensate Header Normal Rejection Level Control Valve • Condensate Polishing Rinse Recycle Pump Skid • Condensate Polishing System Inlet Thermal Well • Condensate Pump Liquid Seal Flow Orifice A/B/C • Condensate Pump Redundant Minimum Flow Protection valve • Condensate Pumps A/B/C • Condensate Filters A/B • Condensate Polishing Skid • Gland Steam Condenser Outlet Thermal Well • Condensate Strainers A/B • Feedwater Bypass Manual Valve • Feedwater Header Temperature Thermal Well • Feedwater Main Condenser • Feedwater Pumps A/B/C • Feedwater Pumps Minimum Flow Protection Control Valve A/B/C • Gland Steam Condenser Bypass Manual Valve • Long Cycle Cleanup AOV and Flow Control Valve • LP, IP, & HP Feedwater Heater • LP, IP, & HP FWH Inlet Thermal Well • LP, IP, & HP FWH Outlet Temperature Thermal Well • LP, IP, & HP FWH Outlet Thermal Well • LP/IP Feedwater Heater Bypass Manual Valve • Main Condenser Emergency Makeup Level Control Valve • Main Condenser Normal Makeup Level Control Valve • Main Condenser Thermal Well • PSS Sampler (Isolock) • Short Cycle Cleanup Flow Control Valve • Sparging Steam Control Valve	TGB	B2	None	None	None	D	III
FWTS, Feedwater Treatment							
All components (except as listed below)	TGB	B2	None	None	No	D	III
CPRS, Condensate Polisher Resin Regeneration System							
All components	TGB	B2	None	None	• NEI 97-06 • EPRI PWR Secondary Water Chemistry Guidelines, Rev 7	D	III
HVDS, (Feedwater) Heater Vents and Drains System							
All components	TGB	B2	None	None	None	D	III
CHWS, Chilled Water System							
All components	RXB, CRB, CUB, RWB	B2	None	None	None	N/A	III
ABS, Auxiliary Boiler System							
• High Pressure and Low Pressure Aux Boiler skids • High Pressure and Low Pressure Aux Boiler Condensate Tanks • High Pressure and Low Pressure Chemical Injection Packages • High Pressure Aux Boiler Flash Tank	ABB	B2	None	None	No	D	III
Radioactivity Instruments	RXB, ABB	B2	None	AQ	ANSI N42.18-2004	N/A	III
CARS, Condenser Air Removal System							
All components (except as listed below)	TGB	B2	None	None	None	D	III
• Effluent Radiation Element • Effluent Radiation Transmitter • Discharge Flow Transmitter	TGB	B2	None	AQ	IEEE 497-2002 with CORR 1	N/A	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
TGS, Turbine Generator System							
All components (except as listed below)	TGB	B2	None	None	None	N/A	III
TG Gland Seal Exhauster Radiation Monitor	TGB	B2	None	AQ	ANSI N42.18-2004	N/A	III
TLOSS, Turbine Lube Oil Storage System							
All components	TGB	B2	None	None	None	N/A	III
CPS, Cathodic Protection System							
Cathodic Protection System	RXB, RWB, TGB, ANB, CRB, DGB, CUB, FWB, ABB, Yard, Other minor buildings	B2	None	None	None	N/A	III
CWS, Circulating Water System							
All components (except as listed below)	TGB, Yard	B2	None	None	None	D	III
CWS pump bay and cooling tower basin level instrumentation	TGB, Yard	B2	None	None	None	N/A	III
SCWS, Site Cooling Water System							
All components (except as listed below)	RXB, CUB, TGB, ABB, Yard	B2	None	None	None	D	III
SCWS Instrumentation (except as listed below)	RXB, CUB, TGB, ABB, Yard	B2	None	None	None	N/A	III
Letdown line rad monitor	Yard	B2	None	AQ	ANSI N42.18-2004	N/A	III
PWS, Potable Water System							
All components (except as listed below)	Various	B2	None	None	None	N/A	III
Supply and return piping from the CRE penetration (includes only the isolation devices (loop seals) and the piping between the loop seals and the outer wall of the CRE)	CRB	B2	None	None	None	N/A	II
UWS, Utility Water System							
All components (except as listed below)	Yard, RWB, FWB, RXB, TGB, CRB, ANB, CUB	B2	None	None	None	N/A	III
• Wastewater effluent discharge portion of UWS • Discharge Basin • Local Grab Sample Line • Letdown Line	Yard	B2	None	None	None	D	III
Letdown Line Rad Monitor	Yard	B2	None	AQ	ANSI N42.18-2004	N/A	III
DWS, Demineralized Water System							
All components (except as listed below)	Yard, RWB, RXB, ANB, CRB, ABB, TGB, CUB	B2	None	None	None	D	III
Radiation indication instruments for DWS headers	RXB	B2	None	None	None	N/A	III
NDS, Nitrogen Distribution System							
All components	Yard, RWB	B2	None	None	None	N/A	III
SAS, Service Air System							
All components	CUB, ANB, RXB, TGB, RWB	B2	None	None	None	N/A	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
IAS, Instrument and Control Air System							
All components	CUB, RWB, RXB, TGB, SCB, DGB, ANB	B2	None	None	None	N/A	III
TBVS, Turbine Building HVAC System							
All components	TGB	B2	None	None	None	N/A	III
SBVS, Security Building HVAC System							
All components	SCB	B2	None	None	None	N/A	III
DGBVS, Diesel Generator HVAC System							
All components	DGB	B2	None	None	None	N/A	III
ABVS, Annex Building HVAC System							
All components	ANB, RWB	B2	None	None	None	N/A	III
FPS, Fire Protection System							
All components	CRB, RXB, TGB, RWB, SCB, ANB, DGB, ATB, FWB, WHB, CUB	B2	None	AQ	RG 1.189	N/A	III
BPDS, BOP Drain System							
All components (except as listed below)	TGB, CRB, CUB, DGB, ABB, FWB, Yard	B2	None	AQ	RG 1.26	D	III
• Instrumentation • Radiation Monitor	TGB, CRB, CUB, DGB, ABB, FWB, Yard	B2	None	None	None	N/A	III
EHVS, 13.8 KV and SWYD System							
All components	TGB, Yard, Switchyard	B2	None	None	None	N/A	III
EMVS, Medium Voltage AC Electrical Distribution System							
All components	TGB, RXB	B2	None	None	None	N/A	III
ELVS, Low Voltage AC Electrical Distribution System							
B6000 series Motor Control Centers	RXB, CRB	B2	None	AQ	None	N/A	III
• Motor Control Center, non-B6000 • Station Service Transformers for B6000 and non-B6000 MCCs • Load Centers (SWG) for B6000 and non-B6000 MCCs	RXB, CRB, TGB, RWB, SCB, ANB, ATB, CUB, Switchyard	B2	None	None	None	N/A	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
EDSS, Highly Reliable DC Power System							
• Channel A, Channel C, and Common Division I Components: - DC Bus - Switchgear - Batteries 1 and 2 - Battery Chargers 1 and 2 - Transfer Switches 1 and 2 • Channel B, Channel D, and Common Division II Components: - DC Bus - Switchgear - Batteries 1 and 2 - Battery Chargers 1 and 2 - Transfer Switches 1 and 2 • EDSS-C, Cabling • EDSS-C, Fusible Disconnects • EDSS-MS, Cabling • EDSS-MS, Fusible Disconnects	RXB, CRB	B2	None	AQ-S	• 10 CFR 50.55a(1) • 10 CFR 50.55a(h) • IEEE Std. 603-1991 • Environmental Qualification • Independence • Single Failure Criterion • Common-Cause Failure • Location of Indicators and Controls • Multi-Unit Station Considerations	N/A	I
• Channel A, Channel C, and Common Division I Components: - Battery Charger Ammeters 1 and 2 - Battery Monitors 1 and 2 - DC Bus Ground Fault Relay - DC Bus Overvoltage Relay - DC Bus Undervoltage Relay • Channel B, Channel D, and Common Division II Components: - Battery Charger Ammeters 1 and 2 - Battery Monitors 1 and 2 - DC Bus Ground Fault Relay - DC Bus Overvoltage Relay - DC Bus Undervoltage Relay	RXB, CRB	B2	None	AQ-S	• 10 CFR 50.55a(1) • 10 CFR 50.55a(h) • IEEE Std. 603-1991 • Environmental Qualification • Independence • Single Failure Criterion • Common-Cause Failure • Location of Indicators and Controls • Multi-Unit Station Considerations	N/A	I
Channel A, Channel B, Channel C, Channel D, Common Division I, and Common Division II DC Bus Voltmeters	RXB, CRB	B2	None	AQ-S	• 10 CFR 50.55a(1) • 10 CFR 50.55a(h) • IEEE Std. 603-1991 • Environmental Qualification • Independence • Single Failure Criterion • Common-Cause Failure • Location of Indicators and Controls • Multi-Unit Station Considerations • IEEE 497-2002 with CORR 1	N/A	I
EDNS, Normal DC Power System							
All components	RXB, CRB, RWB, TGB, Yard	B2	None	None	None	N/A	III
BPSS, Backup Power Supply System							
All components (except as listed below)	DGB, RXB, Yard	B2	None	AQ-S	None	N/A	II
Auxiliary AC Power Supply	Yard	B2	None	None	None	N/A	III
PLS, Plant Lighting System							
All components (except as listed below)	All Buildings	B2	None	None	None	N/A	III
Main Control Room DC emergency lighting (including fixtures, cables, and lighting boards)	CRB	B2	None	AQ	• Powered from highly-reliable DC power distribution system • Environmental Qualification	N/A	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
GLPS, Grounding and Lightning Protection System							
All components	RXB, TGB, RWB, SCB, ANB, DGB, ATB, CUB, FWB, CRB	B2	None	None	None	N/A	III
SPS, Security Power System							
All components	Various	B2	None	None	None	N/A	III
MPS, Module Protection System							
All components (except as listed below)	RXB, CRB	A1	N/A	Q	None	N/A	I
• Division I and Division II Engineered Safety Features Actuation System: - Equipment Interface Modules for Secondary MSIVs, Secondary MSIV Bypass Isolation Valves and Feedwater Regulating Valves for Containment Isolation and DHRS Actuation • Manual LTOP Actuation Switch • Separation Group A, B, C, and D: - Safety Function Module and associated Maintenance Switch for LTOP function	RXB, CRB	A2	N/A	Q	None	N/A	I
• Separation Group A - Safety Function Module: - Feedwater Indication and Control - Leak Detection into Containment • Separation Group B and C - Safety Function Module for PAM indication functions • Separation Group D - Safety Function Module: - Leak Detection into Containment	RXB	B2	None	AQ-S	• IEEE 497-2002 with CORR 1 • EMI/RFI • Environmental Qualification • Power from Vital Instrument Bus • 10 CFR 50.55a(1) • 10 CFR 50.55a(h) • IEEE Std. 603-1991 • Independence • Single Failure Criterion • Common-Cause Failure • Location of Indicators and Controls • Multi-Unit Station Considerations	N/A	I
• 24-Hour Timers for PAM-only Mode • Division I and Division II: - Engineered Safety Features Actuation System - Equipment Interface Module for low AC voltage to battery chargers function - Engineered Safety Features Actuation System Monitoring and Indication Bus, Communication Module - MPS Gateway - Reactor Trip System Monitoring and Indication Bus - Communication Module • Separation Group A, B, C, and D: - Monitoring and Indication Bus - Communication Module • Separation Group B and C - Safety Function Modules for PAM indication functions	RXB	B2	None	AQ-S	• IEEE 497-2002 with CORR 1 • EMI/RFI • Environmental Qualification • Power from Vital Instrument Bus	N/A	I
Division I and II Maintenance Workstations	RXB	B2	None	AQ-S	None	N/A	II
NMS, Neutron Monitoring System							
• Excore Neutron Detectors • Excore Separation Group A/B/C/D - Power Isolation, Conversion and Monitoring Devices • Excore Signal conditioning and processing equipment	RXB	A1	N/A	Q	None	N/A	I
• Flood Highly Sensitive Neutron Detectors (for CNV flooding events) • Flood Signal conditioning and processing equipment (for CNV flooding events)	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• Refuel Neutron Detectors (for refueling) • Refuel Signal conditioning and processing equipment (for refueling)	RXB	B2	None	AQ-S	None	N/A	II
SDIS, Safety Display and Indication System							
All components	CRB	B2	None	AQ-S	• IEEE 497-2002 with CORR 1 • EMI/RFI • Power from Vital Instrument Bus	N/A	I

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
MCS, Module Control System							
• RSS HMI • MCR HMI • MCS Domain Controller (Green) • MCS Domain Controller (Yellow)	RXB, CRB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	II
• Gateway from MPS • Gateway to PCS	RXB, CRB	B2	None	AQ-S	None	N/A	II
• Cabinets (PAM E Variables) • Controllers (PAM E Variables) • I/O Modules (PAM E Variables)	RXB, CRB, TGB	B2	None	AQ	IEEE 497-2002 with CORR 1	N/A	III
• Controllers (other than above) • I/O Modules (other than above)	RXB, CRB, TGB	B2	None	AQ	None	N/A	III
ICIS, In-Core Instrumentation System							
In-core instrument string sheath	RXB	A2	N/A	Q	None	A	I
In-core instrument string/ temperature sensors	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
In-core instrument string/ flux sensors	RXB	B2	None	AQ-S	None	N/A	I
Signal Conditioning and Processing Electronics	RXB	B2	None	AQ-S	None	N/A	II
PCS, Plant Control System							
• Controllers • I/O Modules	CRB, RXB, TGB, RWB	B2	None	AQ	• Backup diesel powered • Analyzed for seismic qualification	N/A	III
• Controllers for RSS indication • I/O Modules for RSS indication	RXB	B2	None	AQ	IEEE 497-2002 with CORR 1	N/A	III
• Cabinets • PCS Domain Controller (Green) • PCS Domain Controller (Yellow) • RSS HMI • MCR HMI	CRB, RXB, TGB, RWB	B2	None	AQ	• IEEE 497-2002 with CORR 1 • Backup diesel powered • Analyzed for seismic qualification	N/A	III
• Gateway from MCS X • Gateway from PPS • RWBCR HMI	CRB, RXB, RWB	B2	None	None	None	N/A	III
PPS, Plant Protection System							
• Division I and Division II: - Monitoring and Indication Bus Communication Modules - Division I Safety Function Module for Spent Fuel Pool and Reactor Pool Level Indication - Equipment Interface Modules: • CRH Air Supply Isolation Valve • CRH Pressure Relief Isolation Valve • CRV General Exhaust Damper • CRV Return Air Damper • CRV Smoke Purge Exhaust Damper • CRV Supply Air Damper • Division I and Division II Safety Function Module for EDSS-C Bus Voltage Indication	CRB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
Division I and Division II: • ELVS Voltage Sensors • Manual CRH Actuation Switches	CRB	B2	None	AQ-S	None	N/A	I
Division I and Division II Safety Function Module for CRE Air Flow Delivery Indication	CRB	B2	None	AQ-S	None	N/A	I

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
Division I and Division II: • CTB Communication Module • Enable Nonsafety Control Switch • Hard-Wired Module • Scheduling and Bypass Modules • Safety Function Modules for CRV Post-filter Radiation Sensor • Safety Function Module for CRV Post-filter Radiation Sensor Trip/Bypass Switches	CRB	B2	None	AQ-S	RG 1.78	N/A	I
Division I and Division II: • CRV Outside Air Isolation Damper Equipment Interface Module • Manual Outside Air Isolation Actuation Switch • Safety Function Module for CRV Toxic Gas Sensor • Safety Function Module for CRV Toxic Gas Sensor Trip/Bypass Switch	CRB	B2	None	AQ-S	RG 1.78	N/A	I
Division I and Division II Maintenance Workstations	CRB	B2	None	AQ-S	None	N/A	II
RMS, Radiation Monitoring System							
RM system that monitors PAM B & C variables	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
Radiation monitors that monitors Type E variables	RXB, TGB	B2	None	AQ	IEEE 497-2002 with CORR 1	N/A	III
Area airborne radiation monitors that monitors Type E Variable	CRB, RXB	B2	None	AQ	• IEEE 497-2002 with CORR 1 • ANSI/HPS N13.1-2011	N/A	III
Area airborne radiation monitors in: • Annex Building • Radioactive Waste Building • Reactor Building	ANB, RWB, RXB	B2	None	AQ	ANSI/HPS N13.1-2011	N/A	III
Radiation monitors in: • Annex Building • Control Building • Radioactive Waste Building • Reactor Building • Turbine Buildings	ANB, CRB, RWB, RXB, TGB	B2	None	AQ	None	N/A	III
RXB, Reactor Building							
Reactor Building (includes interior walls and floor forming UHS pool)	Yard	A1	N/A	Q	None	N/A	I
RBC, Reactor Building Cranes							
Reactor Building Crane	RXB	B1	None	AQ-S	ASME NOG-1	N/A	I
Module Lifting Adapter	RXB	B1	None	AQ-S	ANSI N14.6	N/A	N/A
Traveling Jib Crane	RXB	B2	None	N/A	None	N/A	II
Wet Hoist	RXB	B2	None	AQ	ASME NOG-1	N/A	N/A
RBCM, Reactor Building Components							
Over-Pressurization Vents (OPV)	RXB	A2	None	Q	None	C(d)	I
• UHS Pool Liner and Dry Dock Liner Dry Dock Gate support stainless steel plates at plate-to-liner weld locations	RXB	B2	None	AQ-S	ANSI/ANS 57.2-1983 with additions, clarifications, and exceptions of RG 1.13	N/A	I
Bioshield	RXB	B2	None	AQ-S	EQ requirements to GDC 4 and 23	N/A	II
Reactor Building Equipment Door	RXB	B2	None	AQ-S	None	N/A	II
Dry Dock Gate	RXB	B2	None	AQ-S	None	N/A	II
• Dry Dock Gate Closure instrumentation • Reactor Building Equipment Door Condition Instrumentation	RXB	B2	None	None	None	N/A	III
[[TGB, Turbine Generator Building]]							
Turbine Generator Building	Yard	B2	None	None	None	N/A	III
[[TBC, Turbine Building Cranes]]							
Turbine Building Cranes	TGB	B2	None	None	None	N/A	III
RWB, Radioactive Waste Building							
Radioactive Waste Building	Yard	B2	None	AQ	None	RW-IIa	II, RW-IIa

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
[[SCB, Security Buildings (Guardhouse)]]							
• Security Building • Vehicle inspection sally port	Yard	B2	None	None	None	N/A	III
[[ANB, Annex Building]]							
Annex Building	Yard	B2	None	None	None	N/A	III
[[DGB, Diesel Generator Building]]							
Diesel Generator Building	Yard	B2	None	None	None	N/A	III
[[CUB, Central Utility Building]]							
Central Utility Building	Yard	B2	None	None	None	N/A	III
[[FWB, Firewater Building]]							
Firewater Building	Yard	B2	None	None	None	N/A	III
CRB, Control Building							
CRB Structure at EL 120'-0" and below (except as discussed below).	Yard	A1	N/A	Q	None	N/A	I
• CRB Structure above EL 120'-0" • Inside the CRB elevator shaft and two stairwells, full height of structure • CRB Fire Protection Vestibule (on East Side of CRB)	Yard	B2	None	AQ-S	None	N/A	II
MEMS, Metrology and Environmental Monitoring System							
All components	Yard, CRB	B2	None	AQ	IEEE 497-2002 with CORR 1	N/A	III
COMS, Communication Systems							
All components	Yard for collection of data CRB for display of results	B2	None	None	None	N/A	III
SMS, Seismic Monitoring System							
All components	RXB, CRB	B2	None	AQ-S	None	N/A	I

Note 1: Acronyms used in this table are listed in Table 1.1-1.

Note 2: QA Program applicability codes are as follows:

- Q = indicates quality assurance requirements of 10 CFR 50 Appendix B are applicable in accordance with the quality assurance program (see Section 17.5).
- AQ = indicates that pertinent augmented quality assurance requirements for nonsafety-related SSCs are applied to ensure that the function is accomplished when needed based on that functionality's regulatory requirements. Note that in meeting regulatory guidance, codes, and standards, those applicable SSCs may also have quality assurance requirements invoked by said guidance (e.g., RG 1.26, RG 1.143, IEEE 497, RG 1.189).
- AQ-S = indicates that the pertinent requirements of 10 CFR 50 Appendix B are applicable to nonsafety-related SSC classified as Seismic Category I or Seismic Category II in accordance with the quality assurance program.
- None = indicates no specific QA program or augmented quality requirements are applicable.

Note 3: Additional augmented design requirements, such as the application of a Quality Group, radwaste safety, or seismic classification, to nonsafety-related SSC are reflected in the columns Quality Group/Safety Classification and Seismic Classification, where applicable.

Note 4: See Section 3.2.2.1 through Section 3.2.2.4 for the applicable codes and standards for each RG 1.26 Quality Group designation A, B, C, and D. A Quality Group classification per RG 1.26 is not applicable to supports or instrumentation. See Section 3.2.1.4 for a description of RG 1.143 classifications for RW-IIa, RW-IIb, and RW-IIc.

Note 5: Where SSC (or portions thereof) as determined in the as-built plant which are identified as Seismic Category III in this table could, as the result of a seismic event, adversely affect Seismic Category I SSC or result in incapacitating injury to occupants of the control room, they are categorized as Seismic Category II consistent with Section 3.2.1.2 and analyzed as described in Section 3.7.3.8.

Note 6: Provides nonsafety-related backup isolation to a safety-related isolation device. See FSAR Sections 3.9.6.5, 15.0.0.6.6 and Table 3.9-17.

Note 7: Includes all subcomponents of the reactor vessel internals upper riser assembly with the exception of the bellows lateral seismic restraining structure and bellows vertical expansion structure which are listed separately.

3.3 Wind and Tornado Loadings

The design includes three structures that are evaluated for wind and tornado loadings: the Seismic Category I Reactor Building (RXB) and Control Building (CRB) [the CRB is Seismic Category II above elevation 120' and in the areas below 120' defined in Section 1.2.2.2] and the Seismic Category II Radioactive Waste Building (RWB). The RXB, CRB and RWB are enclosed structures. This section describes the design approach for severe and extreme wind loads on these structures. Section 3.8.4 discusses the design of the Seismic Category I Structures.

The Seismic Category II RWB is also classified as RW-IIa (High Hazard) in accordance with Regulatory Guide (RG) 1.143, Rev. 2, "Design Guidance For Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." The RWB is designed using the same wind, tornado and hurricane loads as specified for as the Seismic Category I structures. This meets or exceeds the wind load specified in Table 2 of RG 1.143, Rev. 2. This regulatory guide directs the use of ASCE 7-95 for wind loads. However, ASCE 7-05 (Reference 3.3-1) is used for wind loads in this design. Similarly, the tornado missiles from RG 1.76, Rev.1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," are used rather than the tornado missiles identified in Table 2 of RG 1.143, Rev. 2.

In addition, other structures, systems, and components that have the potential to interact with the Seismic Category I buildings are evaluated to demonstrate they do not adversely affect the RXB or Seismic Category I portions of the CRB. This is described in Section 3.3.3.

The design complies with General Design Criteria 2 and 4 in that structures, systems, and components are designed to withstand the most severe effects of natural phenomena wind, hurricane, and tornadoes without loss of the capability to perform their safety functions. This is achieved by establishing design parameters that are representative of a reasonable number of potential plant site locations in the United States. Design parameters for severe wind loads are provided in Section 3.3.1.1 and design parameters for extreme wind loads are provided in Section 3.3.2.1.

The RWB has been evaluated for severe and extreme wind loads using the methodology in Section 3.3.1.2 and Section 3.3.2.2 and can withstand the severe and extreme winds.

3.3.1 Severe Wind Loadings

3.3.1.1 Design Parameters for Severe Wind

The design basis severe wind is a 3-second gust at 33 feet above ground for exposure category C. The wind speed (V_w) is 145 mph. The wind speed is increased by an importance factor of 1.15 for the design of the RXB, CRB, and RWB. These design parameters are based upon ASCE/SEI 7-05.

3.3.1.2 Determination of Severe Wind Forces

The maximum velocity pressure (q_z) based on the applicable maximum wind speed (V_w) is calculated in conformance with ASCE/SEI 7-05 (Reference 3.3-1), Equation 6-15, as follows:

$q_z=0.00256\ K_z\ K_{zt}\ K_d\ V_w^2\ I\ (lb/ft^2)$

where,

K_z = velocity pressure exposure coefficient evaluated at height "z", as defined in ASCE/SEI 7-05, Table 6-3, but not less than 0.87. For simplicity and conservatism, z is assumed to be the building height,

K_{zt} = topographic factor equal to 1.0,

K_d = wind directionality factor equal to 1.0,

V_w = maximum wind speed equal to 145 mph, and

I = importance factor equal to 1.15 for the RXB, CRB, and RWB.

Design wind loads on the RXB, CRB, and RWB are determined in conformance with ASCE/SEI 7-05 (Reference 3.3-1), Equation 6-17:

$p=qGC_p - q_i\ (GC_{pi})\ (lb/ft^2)$

where,

G = gust factor equal to 0.85,

C_p = external pressure coefficient equal to 1.0,

GC_{pi} = internal pressure coefficient equal to 0.18,

q = velocity pressure, and

q_i = internal velocity pressure.

3.3.2 Extreme Wind Loads (Tornado and Hurricane Loads)

3.3.2.1 Design Parameters for Extreme Winds

Tornado wind loads include loads caused by the tornado wind pressure, tornado atmospheric pressure change effect, and tornado-generated missile impact. Hurricane wind loads include loads due to the hurricane wind pressure and hurricane-generated missiles.

The parameters for the design basis tornado are the most severe tornado parameters postulated for the contiguous United States as identified in RG 1.76, Rev. 1.

- Maximum wind speed .... 230 mph
- Translational speed .... 46 mph

- Maximum rotational speed .... 184 mph
- Radius of maximum rotational speed...... 150 ft
- Pressure drop............................. 1.2 psi
- Rate of pressure drop .... 0.5 psi/s

The wind speed for the design basis hurricane is the highest wind speed postulated in Regulatory Position 1 of RG 1.221, Rev. 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," which occurs in Figure 2 of RG 1.221, Rev. 0.

- Maximum wind speed .... 290 mph

Refer to Section 3.5 for a description of hurricane and tornado wind-generated missiles.

3.3.2.2 Determination of Tornado and Hurricane Forces

Tornado and hurricane wind velocities are converted into effective pressure loads in accordance with ASCE/SEI 7-05 (Reference 3.3-1), Equation 6-15, as follows:

$q_z = 0.00256\ K_z\ K_{zt}\ K_d\ V_w^2\ I\ (lb/ft^2)$

where,

K_z = velocity pressure exposure coefficient evaluated at height "z", as defined in with ASCE/SEI 7-05, Table 6-3, but not less than 0.87. (For tornados, wind speed is not assumed to vary with height.) For simplicity and conservatism, z is assumed to be the building height.

K_{zt} = topographic factor equal to 1.0,

K_d = wind directionality factor equal to 1.0,

V_w = maximum wind speed (mph) (For tornadoes, V_w is the resultant of the maximum rotational speed and the translational speed), and

I = importance factor equal to 1.15 for the RXB, CRB, and RWB.

Extreme wind loads on the RXB, CRB, and RWB are determined in conformance with ASCE/SEI 7-05, Equation 6-17:

$p = qGC_p - q_i\ (GC_{pi})\ (lb/ft^2)$

where,

G = gust factor equal to 0.85,

C_p = external pressure coefficient equal to 1.0,

GC_{pi} = internal pressure coefficient equal to 0.18 for the hurricane,

q = velocity pressure, and

q_i = internal velocity pressure.

Internal pressure from the tornado is the design parameter for maximum pressure drop.

3.3.2.3 Combination of Forces

The most adverse of the following combinations are considered for the total hurricane or tornado load:

$$W_t = W_p$$

$$W_t = W_w + 0.5\,W_p + W_m$$

where,

W_t = total load,

W_w = load from wind effect,

W_p = load from tornado atmospheric pressure change effect ($W_p = 0$ for hurricanes), and

W_m = load from missile impact effect.

COL Item 3.3-1: A COL applicant that references the NuScale Power Plant design will confirm that nearby structures exposed to severe and extreme (tornado and hurricane) wind loads will not collapse and adversely affect the Reactor Building or Seismic Category I portion of the Control Building.

3.3.3 References

3.3-1 American Society of Civil Engineers/Structural Engineering Institute, "Minimum Design Loads for Buildings and Other Structures," ASCE/SEI 7-05, Reston, VA.

3.4 Water Level (Flood) Design

Flooding of a nuclear power plant can come from internal sources - piping ruptures, tank failures or the actuation of fire suppression systems, or from external sources - flooding from nearby water bodies or precipitation. Section 3.4.1 evaluates flooding effects of discharged fluid resulting from the high and moderate energy line breaks and cracks; from fire-fighting activities; and from postulated failures of non-seismic and non-tornado protected piping, tanks, and vessels outside the structures. In the absence of final pipe routing information, the flooding hazards are representative of the flooding hazards expected throughout the plant.

The design satisfies General Design Criterion 4 in that the structures, systems, and components (SSC) are designed to withstand the effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents without loss of the capability to perform their safety functions.

The design also satisfies General Design Criterion 2 in that SSC accommodate the effects of natural phenomena, including floods, without losing the ability to perform their safety function. Section 3.4.2 addresses flooding from natural phenomena.

Dynamic effects from pipe rupture are addressed in Section 3.6. Environmental effects are addressed in Section 3.11. Loads on Seismic Category I and other structures are addressed in Section 3.8.

3.4.1 Internal Flood Protection for Onsite Equipment Failures

Internal flooding analyses were performed in the Reactor Building (RXB) and the Control Building (CRB) to confirm that flooding from postulated failures of tanks and piping or actuation of fire suppression systems does not cause the loss of equipment required to: (a) maintain the integrity of the reactor coolant pressure boundary for any module, (b) shut down the reactor for any module and maintain it in a safe shutdown condition, or (c) prevent or mitigate the consequences of accidents which could result in unacceptable offsite radiological consequences. These SSC are collectively identified as "equipment subject to flood protection."

Table 3.4-2 identifies the rooms that contain SSC that have safety-related or risk-significant attributes that are subject to flood protection. The flooding analysis considers areas and rooms that contain these SSC, not the specific SSC themselves. Safety-related cable is either routed above the flood level or qualified for submergence. Rooms where cable is the only safety-related SSC are not included. Mitigation of flooding in the identified rooms will be accomplished by, for example, watertight or water resistant doors, elevating equipment above the flood level, enclosing or qualifying equipment for submersion, or other similar type of flood protection.

The internal flooding analysis is conducted on a level-by-level and room-by-room basis for the Seismic Category I RXB and CRB for the postulated flooding events.

The RXB and CRB flooding analysis consists of the following steps:

- identification of potential flooding sources
- identification of rooms/areas that contain equipment subject to flood protection

- estimation of flood depth in the identified rooms/areas
- determination of the need for protection and mitigation measures for rooms containing equipment subject to flood protection

3.4.1.1 Assumptions used in the Flooding Analyses

Unless a stress analysis has been performed to identify potential break locations or eliminate the piping from consideration of potential breaks, high and moderate energy piping greater than 2 inches nominal diameter are assumed to have a full circumferential break in any room or area where they pass. The design operational pressure/flow rate is used to estimate leakage flow rates. The total quantity of fluid released is consistent with the action necessary to isolate the line. The following assumptions are used for isolation times:

For the CRB:

- Thirty minutes are assumed between leak initiation and leak isolation (the CRB is continuously occupied).

For the RXB:

- Thirty minutes are assumed between initiation of a leak and detection by any means (except for the main steam line which automatically isolates).
- Ten minutes are assumed between leak detection and isolation.

Fire suppression activities are also a potential flooding source. The discharge of the fire suppression system for the RXB and CRB is assumed to be 700 gpm and 550 gpm, respectively. These estimates are based on the automatic fire suppression flow rate of 0.3 gpm/ft^2 over a 1,500 ft^2 area for the RXB and 0.2 gpm/ft^2 over a 1,500 ft^2 area for the CRB based on the occupancy categories of NFPA 13 (Reference 3.4-1) with the addition of 250 gpm for manual hose flow (NFPA 14, Reference 3.4-2). The fire suppression duration is assumed to be two hours for the RXB and 60 minutes for the CRB based on the occupancy categories of NFPA 13.

The following assumptions are used to determine flood water volumes in rooms and areas within the RXB and CRB:

- Floor drains and sump pumps are not credited for reducing flood water volume during the event.
- Backflow through floor drains is not considered. It is assumed to be bounded by the direct flooding pathways. Floor drains are discussed in Section 9.3.3.
- Interior doors, unless specified as a watertight/waterproof door, are assumed to fail open or provide a high leak flow rate between rooms.
- In areas with multiple sources, each source is considered separately.

3.4.1.2 Reactor Building Flooding Analysis

There are multiple flooding sources in the RXB. The sources are discussed below, and the water sources and volumes are listed in Table 3.4-1.

- The 12-inch fire protection main lines enter the RXB through pipe shrouds located on the north and south side of the RXB at elevation 100’-0”. This header distributes fire protection water to the fire suppression sprinkler system on each RXB elevation. A break in the fire protection line can provide up to 2500 gpm from the pipe rupture. The water from the rupture is assumed to be released for 40 minutes.
- Fire suppression activities consisting of area sprinklers and operating fire hoses with a flowrate of 700 gpm total (450 gpm + 250 gpm respectively), are assumed to provide flooding water for two hours.
- Reactor Building HVAC system chilled water cooling coil piping (from Site Cooling Water) has a flow of 1,000 gpm that is assumed to provide flood water for 40 minutes.
- The site cooling water system header piping into the RXB at elevation 100' has a flow of 5,000 gpm that is assumed to provide flood water for 40 minutes.
- Demineralized water system and utility water system has a flowrate of 300 gpm. The pipe rupture is assumed to provide floodwater for 40 minutes.
- Main steam line break has such a small time frame between the break and pipe isolation (five seconds) that the condensed steam from the break will not cause an internal flood.
- Feedwater line break has a flow of 600 gpm that is assumed to provide flood water for 40 minutes.
- The spent fuel pool cooling and reactor pool cooling inlet and outlet piping are routed from elevation 85' to elevation 50'. A break in either piping line on elevation 75' or elevation 50' could drain 158,900 ft^3 and result in a flood height of 9'-4 ¾" on elevation 50' and 14'-7 ½" on elevation 75'. Each of the rooms that contain SSC subject to flood prevention either have flood doors or the equipment in the rooms are designed or protected for submergence.
- Chemical volume control system (CVCS) line break has a flow of 90 gpm that is assumed to provide flood water for 40 minutes.
- Pool surge control system line break has a maximum flow of 2747 gpm that is assumed to provide flood water for 40 minutes.
- Auxiliary boiler system has maximum break flow of 80 gpm and that is assumed to provide flood water for 40 minutes.

3.4.1.2.1 Flooding at Elevation 125'-0"

Flooding of this elevation results from a fire suppression system actuation or a site cooling water pipe break. The electrical and mechanical equipment rooms on this elevation contain SSC that are subject to flood protection. Water level on this elevation is predicted to be less than four inches. Individual rooms subject to flood protection are shown in Table 3.4-2.

3.4.1.2.2 Flooding at Elevation 100'-0"

Flooding of this elevation can be caused by fire suppression system actuation or a feedwater line break. The feedwater line break produces the highest water level of approximately 48 inches. Individual rooms subject to flood protection are shown in Table 3.4-2.

3.4.1.2.3 Flooding at Elevation 86'-0"

A fire suppression system actuation in the hallways provides flooding water for this elevation. However, the metal floor grating in the hallways allows the flood water to drain to elevation 75'-0".

3.4.1.2.4 Flooding at Elevation 75'-0"

Elevation 75'-0" of the RXB contains the remote shutdown station and other electrical equipment rooms that house SSC that are subject to flood protection. Grating in elevation 86'-0" hallway floors drains flood water from that elevation to elevation 75'-0" hallways. However, fire suppression activities in the elevation 75′-0″ hallways produces the highest flood level of approximately 23 inches. Individual rooms containing equipment subject to flood protection have smaller flood levels. Individual rooms subject to flood protection are shown Table 3.4-2.

3.4.1.2.5 Flooding at Elevation 62'-0"

Miscellaneous mechanical equipment rooms are located on elevation 62′-0″. There are no SSC subject to flood protection located at this elevation.

3.4.1.2.6 Flooding at Elevation 50'-0"

Elevation 50'-0" contains CVCS equipment, demineralized water valves, and miscellaneous mechanical and electrical equipment rooms. Fire suppression activities in the hallways produces the highest flood level of approximately 16.5 inches.

3.4.1.2.7 Flooding at Elevation 35'-8"

Elevation 35'-8" contains CVCS pump rooms and miscellaneous mechanical equipment rooms. There are no SSC subject to flood protection located at this elevation.

3.4.1.2.8 Flooding at Elevation 24'-0"

Elevation 24'-0" contains CVCS filters and ion exchangers and miscellaneous mechanical equipment rooms. There are no SSC subject to flood protection located on this elevation.

3.4.1.2.9 Containment Flooding Analysis

Containment is flooded as part of normal shutdown, and may also be flooded as part of accident mitigation as described in Chapter 15. Therefore, there is no equipment subject to flood protection inside containment and no containment flooding analysis is necessary.

3.4.1.3 Control Building Flooding Analysis

There are four potential flooding sources in the CRB. The sources are discussed below, and the water volumes and sources are listed in Table 3.4-1.

- The 6-inch fire protection main line enters the CRB through the fire riser room between the 100' and 120' floor level. From this header, the pipe distributes fire protection water to the fire suppression sprinkler system located on each CRB elevation. A break in the fire protection line can provide up to 2,225 gpm from the pipe rupture. The water from the rupture is assumed to be released for 30 minutes.
- The 4-inch chilled water supply provides water to the HVAC system on elevation 120' of the CRB, and has a flow of 226 gpm that is assumed to provide flood water for 30 minutes.
- The 2-inch potable water supply pipe provides potable water to floor elevation 76' 6" and elevation 100'. Though this line is not considered a large pipe, its routing through the CRB poses a flooding risk. The system has a flow of 50 gpm that is assumed to provide flood water for 30 minutes.
- Fire suppression activities consisting of area sprinkler and operation fire hoses with a flow rate of 550 gpm (300 gpm + 250 gpm, respectively), are assumed to provide flooding water for one hour.

3.4.1.3.1 Flooding at Elevation 120'-0"

Elevation 120'-0" contains HVAC and miscellaneous mechanical equipment. There are no SSC subject to flood protection located at this elevation.

3.4.1.3.2 Flooding at Elevation 100'-0"

Flooding at the 100'-0" elevation could occur from a break in the potable water system, a break in the fire suppression riser, or from fire-fighting activities. There are no SSC subject to flood protection at elevation 100'-0".

The fire riser room is located outside the main building next to the vestibule. The fire riser is a potential flooding source in the CRB. However, the water from the riser will flow into the vestibule and out to the environment or into the main hallway and down the stairwells and will have no impact on elevation 100’-0”.

3.4.1.3.3 Flooding at Elevation 76'-6"

The main control room is located on elevation 76'-6". This room contains equipment subject to flood protection. Flooding could occur from actuation of the sprinkler system in an adjacent hallway or from a break in the potable water line that is routed which is in rooms connected to the hallway. Due to the small volume of water from a potable water system line break, sprinkler actuation is the dominant flooding source. Firefighting activities in the adjacent rooms could result in a flood depth of approximately 17.5 inches.

3.4.1.3.4 Flooding at Elevation 63'-3"

Elevation 63'-3” contains electrical equipment and utility rooms. There are no SSC subject to flood protection located at this elevation.

3.4.1.3.5 Flooding at Elevation 50'-0"

Elevation 50'-0" contains electrical equipment, air bottles, and utility rooms. There are no SSC that are subject to flood protection at this elevation.

3.4.1.4 Flooding Outside the Reactor and Control Buildings

Flooding of the RXB or CRB caused by external sources does not occur. The design external flood level is established as less than 99' elevation (one foot below the baseline plant elevation (top of concrete) at 100'-0"). The finished grade at the building perimeter of the RXB and CRB is approximately 6 inches below the top of concrete elevation, except at the CRB tunnel and a truck ramp on the south side of the Radwaste Building.

Water from tanks and piping that are non-seismic and non-tornado/hurricane protected is a potential flooding source outside the buildings. [[However, there are no large tanks or water sources near the entrances to the CRB and RXB.]] The site is graded to transport water away from these buildings. Therefore, failure of equipment outside the CRB and RXB cannot cause internal flooding.

3.4.1.5 Site Specific Analysis

COL Item 3.4-1: A COL applicant that references the NuScale Power plant design certification will confirm the final location of structures, systems, and components subject to flood protection and final routing of piping.

COL Item 3.4-2: A COL applicant that references the NuScale Power plant design certification will develop the on-site program addressing the key points of flood mitigation. The key points to this program include the procedures for mitigating internal flooding events; the equipment list of structures, systems, and components subject to flood protection in each plant area; and providing assurance that the program reliably mitigates flooding to the identified structures, systems, and components.

COL Item 3.4-3: A COL applicant that references the NuScale Power plant design certification will develop an inspection and maintenance program to ensure that each water-tight door, penetration seal, or other “degradable” measure remains capable of performing its intended function.

COL Item 3.4-4: A COL applicant that references the NuScale Power plant design certification will confirm that site-specific tanks or water sources are placed in locations where they cannot cause flooding in the Reactor Building or Control Building.

3.4.2 Protection of Structures Against Flood from External Sources

The design includes the two Seismic Category I structures: the RXB and the CRB. The Radioactive Waste Building (RWB) is Seismic Category II and does not contain any equipment subject to flood protection. There are no other safety-related structures in the design.

3.4.2.1 Probable Maximum Flood

The design is the equivalent of a "Dry Site" as defined in Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," Rev. 1. The Seismic Category I structures are protected from external floods and groundwater by establishing the following design parameters:

- The probable maximum flood elevation (including wave action) of the design is one foot below the baseline plant elevation (100'-0).
- The maximum groundwater elevation for the design is two feet below the baseline plant elevation.
- With the exceptions of the subgrade CRB tunnel and a truck ramp on the south side of the Radwaste Building, the finished grade for all building structures is approximately six inches below the baseline plant elevation. The yard is graded with a minimum slope of 1.5 percent away from these structures.

The below grade portions of the Seismic Category I structures provide protection for the safety-related and risk-significant SSC from groundwater intrusion by utilizing the following design features:

- the portions of the buildings that are below grade consider the use of waterstops and waterproofing
- exterior below grade wall or floor penetrations have watertight seals
- waterproofing and dampproofing systems, if used, are applied per the International Building Code Section 1805 (Reference 3.4-3)
- waterproofing and dampproofing materials, if used in horizontal applications, will have a coefficient of static friction equal to or greater than the design parameter established in Table 2.0-1 for all interfaces between the basemat and soil.

The design does not use a permanent dewatering system.

COL Item 3.4-5: A COL applicant that references the NuScale Power Plant design certification will determine the extent of waterproofing and dampproofing needed for the underground portion of the Reactor Building and Control Building based on site-specific conditions. Additionally, a COL applicant will provide the specified design life for waterstops, waterproofing, damp proofing, and watertight seals. If the design life is less than the operating life of the plant, the COL applicant will describe how continued protection will be ensured.

COL Item 3.4-7: A COL applicant that references the NuScale Power Plant design certification will determine the extent of waterproofing and damp proofing needed to prevent groundwater and foreign material intrusion into the expansion gap between the end of the tunnel between the Reactor Building and the Control Building, and the corresponding Reactor Building connecting walls.

The NuScale Power Plant design establishes a design basis flood level (including wave action) of one foot below the baseline top of concrete elevation at the ground level floor. Therefore, there are no dynamic flood loads on the RXB and CRB. The lateral hydrostatic pressures on the structures due to the design flood level, as well as ground water and soil pressure, are factored into the structural design as static and dynamic loads discussed in Section 3.8.4.3.3.

3.4.2.2 Probable Maximum Precipitation

The design utilizes bounding parameters for both rain and snow. The rainfall rate for roof design is 19.4 inches per hour and 6.3 inches for a 5 minute period and the design static roof load because of snow is 50 pounds per square foot. The extreme snow load is 75 pounds per square foot.

The roofs of the RXB and CRB prevent the undesirable buildup of standing water in conformance with Regulatory Guide 1.102 as described below:

- The RXB has a gabled roof, with the sloping portions to the north and south. There are no parapets on the top, flat section.
- The CRB roof is a sloped steel structure with scuppers in the parapet designed to allow rainfall to drain off the roof. An additional drainage pipe limits the average water depth on the CRB roof to a maximum of four inches.

The bounding rain and snow loads are used in the structural analysis described in Section 3.8.4.

3.4.2.3 Interaction of Non-Seismic Category I Structures with Seismic Category I Structures

Nearby structures are assessed, or analyzed if necessary, to ensure that there is no credible potential for interactions that could adversely affect the Seismic Category I RXB and CRB. Figure 1.2-2 provides a site plan showing the plant layout. The non-Seismic Category I structures that are adjacent to the Seismic Category I RXB and CRB are:

- RWB (Seismic Category II) adjacent to RXB
- CRB above elevation 120' (Seismic Category II), above Seismic Category I CRB and adjacent to RXB
- [[North and south Turbine Generator Buildings (Seismic Category III), adjacent to RXB]]
- [[Central Utilities Building (Seismic Category III), adjacent to CRB]]
- [[Annex Building (Seismic Category III), adjacent to RXB]]

The Seismic Category II portion of the CRB was analyzed along with the Seismic Category I portion of the structure and shown to be capable of withstanding the effects of the probable maximum precipitation.

The RWB has been evaluated and shown to be capable of withstanding the effects of the probable maximum precipitation.

COL Item 3.4-6: A COL applicant that references the NuScale Power Plant design certification will confirm that nearby structures exposed to external flooding will not collapse and adversely affect the Reactor Building or Seismic Category I portion of the Control Building.

3.4.3 References

3.4-1 National Fire Protection Association, "Standard for the Installation of Sprinkler Systems," NFPA 13, 2016 edition, Quincy, MA.

3.4-2 National Fire Protection Association, "Standard for Installation of Standpipe and Hose Systems," NFPA 14, 2016 edition, Quincy, MA.

3.4-3 International Code Council, "Dampproofing and Waterproofing," International Building Code Section 1805, Lenexa, KS, 2015.

Table 3.4-1: Flooding Sources in the Reactor Building and Control Building

Building	Description	Pipe Size (in)	Flow (gpm)	Isolation time (min)	Volume of liquid (gal)	Approximate Volume of liquid (ft^3)
CRB	Fire suppression riser	6	2,225	30	66,750	8,900
	Fire suppression activities	N/A	550	60	33,000	4,400
	Chilled water to HVAC	4	226	30	6,780	900
	Potable water	2	50	30	1500	200
RXB	Fire suppression riser	12	2500	40	100,000	13,400
	Fire suppression activities	N/A	700	120	84,000	11,200
	Main steam	12	77,000	0.0833	6,420	860
	Feedwater	8	600	40	24,000	3,200
	Site cooling water support for HVAC	18	1000	40	40,000	5,400
	Site cooling water header	32	5,000	40	200,000	26,700
	Demineralized water	2	300	40	12,000	1,600
	Auxiliary boiler	6	80	40	3200	400
	CVCS	2-1/2	90	40	3600	500
	Pool surge control system	10	2747	40	110,000	14,700
	Spent fuel pool/reactor pool cooling	10	---	---	1,188,600	158,900

Table 3.4-2: Flood Levels for Rooms Containing Systems, Structures, and Components Subject to Flood Protection (Without Mitigation)

Building	Elevation	Room	Flood depth (in)	Function
RXB	{{ Withheld - See Part 9 }}	010-507	11.25	Mechanical equipment area
		010-509	11.25	Mechanical equipment area
	{{ Withheld - See Part 9 }}	010-411	36.75	Steam gallery
		010-418	48.0	Steam gallery
	{{ Withheld - See Part 9 }}	none		
	{{ Withheld - See Part 9 }}	010-207	17.75	Remote shutdown room
		010-209	22.75	Battery room
		010-210	22.75	Battery room
		010-211	22.75	I/O cabinet room
		010-212	22.75	Battery room
		010-213	22.75	Battery room
		010-214	22.75	Battery room
		010-215	22.75	Battery room
		010-216	22.75	I/O cabinet room
		010-217	22.75	Battery room
		010-218	22.75	Battery room
		010-220	22.75	Battery room
		010-221	22.75	Battery room
		010-222	22.75	I/O cabinet room
		010-223	22.75	Battery room
		010-224	22.75	Battery room
		010-225	22.75	Battery room
		010-226	22.75	Battery room
		010-227	22.75	I/O cabinet room
		010-228	22.75	Battery room
		010-229	22.75	Battery room
		010-230	22.75	Battery room
		010-231	22.75	Battery room
		010-232	22.75	I/O cabinet room
		010-233	22.75	Battery room
		010-234	22.75	Battery room
		010-235	22.75	Battery room
		010-236	22.75	Battery room
		010-237	22.75	I/O cabinet room
		010-238	22.75	Battery room
		010-239	22.75	Battery room
		010-244	23.25	Battery room
		010-245	23.25	Battery room
		010-246	23.25	I/O cabinet room
		010-247	23.25	Battery room
		010-248	23.25	Battery room
		010-249	23.25	Battery room
		010-250	23.25	Battery room
		010-251	23.25	I/O cabinet room
		010-252	23.25	Battery room
		010-253	23.25	Battery room
		010-254	23.25	Battery room
		010-255	23.25	Battery room
		010-256	23.25	I/O cabinet room

Table 3.4-2: Flood Levels for Rooms Containing Systems, Structures, and Components Subject to Flood Protection (Without Mitigation) (Continued)

Building	Elevation	Room	Flood depth (in)	Function
		010-257	23.25	Battery room
		010-258	23.25	Battery room
		010-259	23.25	Battery room
		010-260	23.25	Battery room
		010-261	23.25	I/O cabinet room
		010-262	23.25	Battery room
		010-263	23.25	Battery room
		010-265	23.25	Battery room
		010-266	23.25	Battery room
		010-267	23.25	I/O cabinet room
		010-268	23.25	Battery room
		010-269	23.25	Battery room
		010-270	23.25	Battery room
		010-271	23.25	Battery room
		010-272	23.25	I/O cabinet room
		010-273	23.25	Battery room
		010-274	23.25	Battery room
	{{ Withheld - See Part 9 }}	none		
		010-107	15.00	Mechanical equipment area
	{{ Withheld - See Part 9 }}	010-114	16.00	Mechanical equipment area
		010-125	16.5	Mechanical equipment area
		010-134	15.25	Mechanical equipment area
	{{ Withheld - See Part 9 }}	none		
	{{ Withheld - See Part 9 }}	none		
CRB	{{ Withheld - See Part 9 }}	none		
	{{ Withheld - See Part 9 }}	none		
	{{ Withheld - See Part 9 }}	170-100	17.5	Main control room
	{{ Withheld - See Part 9 }}	none		
	{{ Withheld - See Part 9 }}	none		

3.5 Missile Protection

Protection from external missiles is accomplished by locating SSC that require missile protection inside the Seismic Category I Reactor Building (RXB) or Control Building (CRB), or in the Seismic Category II Radioactive Waste Building (RWB).

The design complies with General Design Criteria (GDC) 2 and GDC 4 in that structures, systems, and components (SSC) are designed to accommodate the effects of internally and externally generated missiles without losing the ability to perform their safety function.

The Seismic Category II RWB is also classified as RW-IIa in accordance with Regulatory Guide (RG) 1.143, "Design Guidance For Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Rev. 2. The RWB is designed for the same external missiles as the Seismic Category I structures. This meets or exceeds the design criteria for missiles specified in Table 2 of RG 1.143, Rev. 2.

Inside the buildings, missile protection is provided by

- providing design features to prevent the generation of missiles.
- orienting or physically separating potential missile sources away from equipment subject to missile protection.
- providing local shields and barriers for equipment subject to missile protection.

Safety-related SSC and those risk-significant SSC that have a safety function that would be relied upon following the missile producing event are potential missile targets. These structures, systems, and components are located inside the RXB and CRB. Table 3.2-1 lists SSC, their safety classification, and their risk significance.

3.5.1 Missile Selection and Description

The following potential missile generating sources are considered:

- internally generated missiles (outside containment) (Section 3.5.1.1)
- internally generated missiles (inside containment) (Section 3.5.1.2)
- turbine missiles (Section 3.5.1.3)
- missiles generated by tornadoes and extreme winds (Section 3.5.1.4)
- site proximity missiles (except aircraft) (Section 3.5.1.5)
- aircraft hazards (Section 3.5.1.6)

Missile generation is assumed to occur during all operating conditions.

After a potential missile has been identified, its statistical significance is determined in accordance with the following.

1) If the probability of occurrence of the missile (P_1) is determined to be less than 10^{-7} per year, the missile is dismissed from further consideration because it is not statistically significant.

2) If (P_1) is greater than 10^{-7} per year, its probability of impacting each safety-related or risk-significant target (P_2) is determined. If the combined probability is less than 10^{-7} per year, the missile and target combination is not considered statistically significant and is dismissed from further consideration.

3) If the product of (P_1) and (P_2) is greater than 10^{-7} per year, the probability for damage to the target (P_3) is assessed. If the combined probability is less than 10^{-7} per year, the missile and target combination is not considered statistically significant and is dismissed.

4) If the product of (P_1), (P_2) and (P_3) is greater than 10^{-7} per year, barriers or other measures are taken to protect the SSC.

3.5.1.1 Internally Generated Missiles (Outside Containment)

Internally generated missiles are missiles from plant equipment or processes. Missiles can be generated from pressurized systems and components, from rotating equipment, from explosions, or from improperly secured equipment. However, not all potential missiles are credible. The following provides discussion on when missiles do not need to be considered credible ($P_1 < 10^{-7}$).

3.5.1.1.1 Pressurized Systems

Moderate and low energy systems have insufficient stored energy to generate a missile. As such, the probability of missile occurrence (P_1) from systems with operating pressures less than 275 psig is considered to be less than 10^{-7} (i.e., not credible).

Although high energy piping failures could result in dynamic effects, they do not form missiles as such because the whipping section remains attached to the remainder of the pipe. Section 3.6 addresses the dynamic effects associated with pipe breaks. Therefore, potential missiles from high energy piping are the attached components: valves, fasteners, thermowells, and instrumentation.

Missiles from piping or valves designed in accordance with ASME Section III, (Reference 3.5-1) and maintained in accordance with an ASME Section XI (Reference 3.5-2) inspection program are not considered credible.

Bolted bonnet valves and pressure-seal bonnet valves constructed in accordance with ASME Section III, ASME B16.34, or to an equivalent consensus standard are not considered credible missiles. The use of consensus standards provides reasonable assurance that the components are designed, manufactured and constructed in a manner that demonstrates a high level of quality (e.g., material, design, fabrication, examination, testing). The use of ASME B16.34 and other recognized industry Codes and Standards provides reasonable assurance that the valve maintains its structural integrity during normal and upset conditions and that bolted bonnet valves and pressure-seal bonnet valves cannot become credible missiles.

Valve stems are not considered as credible missiles if at least one feature (in addition to the stem threads) is included in their design to prevent ejection. Valve stems with back seats are prevented from becoming missiles by this feature. In addition, the valve stems of valves with power actuators, such as air- or motor-operated valves, are effectively restrained by the valve actuator.

Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and thus are not considered as credible missiles.

Thermowells and similar fittings attached to piping or pressurized equipment by welding are not considered as credible missiles. The completed joint has greater design strength than the parent metal. Such a design makes missile formation not credible.

Instrumentation such as pressure, level, and flow transmitters and associated piping and tubing are not considered as credible missiles. The quantity of high energy fluid in these instruments is limited and will not result in the generation of missiles. The connecting piping and tubing is made up using welded joints or compression fittings for the tubing. Tubing is small diameter and has only a small amount of stored energy.

3.5.1.1.2 Pressurized Cylinders

Industrial compressed gas cylinders and tanks are used for the control room habitability system. In addition, smaller portable tanks or bottles used for the chemical and volume control system and maintenance activities may also be stored within the buildings. Cylinders, bottles, or tanks containing highly pressurized gas are considered missile sources unless appropriately secured.

The control room habitability system air bottles are mounted in Seismic Category I racks to ensure that each air bottle is contained and does not become a missile. Plates at the end of each bottle retain horizontal movement and pipe straps are installed to prevent vertical movement.

Procedures developed in accordance with Section 13.5.2.2 ensure that portable pressurized gas cylinders or bottles are moved to a location where they are not a potential hazard to equipment subject to missile protection, or seismically restrained to prevent them from becoming missiles.

3.5.1.1.3 Rotating Equipment

The plant design has limited rotating equipment. There are no reactor coolant pumps, turbine driven pumps, or other large rotating components inside the safety-related structures. The main turbine generators are outside of the RXB and are discussed in Section 3.5.1.3.

Catastrophic failure of rotating equipment such as fans and compressors leading to the generation of missiles is not considered credible. These components are designed to preclude having sufficient energy to move the masses of their rotating parts through the housings in which they are contained. In addition, material characteristics, inspections, quality control during fabrication and erection, and prudent operation as applied to the particular component reduce the likelihood of missile generation.

3.5.1.1.4 Explosions

The battery compartments in the CRB and RXB are ventilated to preclude the possibility of hydrogen accumulation. In addition, the design incorporates valve-regulated lead acid batteries which reduce the hydrogen production in battery rooms compared to vented lead acid batteries. Therefore, a hydrogen explosion in a battery compartment is not a credible missile source. The RWB does not contain any battery compartments.

3.5.1.1.5 Gravitational Missiles

Structures, systems, and components which could fall and impact or adversely affect safety-related or risk-significant SSC are classified as Seismic Category II (Table 3.2-1). Seismic Category II equipment is mounted to ensure there is no adverse interaction between Seismic Category 1 SSC and Seismic Category II SSC as described in Section 3.2.1.2. These structures, systems, and components are not considered credible missiles.

Section 9.1.5 provides an evaluation of the reactor building crane and the module assembly equipment. Due to the significance of a drop of a NuScale Power Module, safety features are designed into these devices as described in Section 9.1.5. Therefore, these devices are not a credible missile source.

Procedures developed in accordance with Section 13.5.2.2 ensure that hoisting or lifting activities address movement of heavy loads above safety-related and risk-significant SSC. Control of heavy loads eliminates drops as credible missile sources.

Unsecured equipment is a potential gravitational missile. Procedures developed in accordance with Section 13.5.2.2 ensure that maintenance equipment, both equipment brought into the building to perform maintenance, and equipment undergoing maintenance located in the RXB or CRB, are seismically restrained to prevent them from becoming missiles, removed from the building, or moved to a location where they are not a potential hazard. Control of unsecured equipment eliminates falling equipment as credible missile sources.

3.5.1.2 Internally Generated Missiles (Inside Containment)

There are no credible missiles inside containment.

The NPM uses a steel containment that encapsulates the reactor pressure vessel (RPV). There is no rotating equipment inside containment, and all pressurized components are ASME Class 1 or 2 and therefore not credible missile sources as discussed in Section 3.5.1.1.1.

A control rod drive mechanism (CRDM) housing failure, sufficient to create a missile from a piece of the housing or to allow a control rod to be ejected rapidly from the core, is non-credible. The CRDM housing is a Class 1 appurtenance per ASME Section III.

3.5.1.3 Turbine Missiles

The turbine generator building layout in relation to the overall site layout is shown on Figure 1.2-2. The turbine generator rotor shafts are physically oriented such that the RXB, CRB, and RWB are within the turbine trajectory hazard, thereby making the turbines unfavorably oriented with respect to the NPMs, as defined by RG 1.115, Revision 2. Appendix A of RG 1.115, Rev. 2 identifies SSC requiring protection from turbine missiles. The SSC that require protection from turbine missiles (high-trajectory and low-trajectory turbine rotor and blade fragments) are located in either the RXB or the CRB. The SSC located in the RXB and below grade in the CRB are classified as A1 or A2 (per Section 3.2) and are considered essential SSC as defined in Appendix A of RG 1.115. Table 3.2-1 provides a complete listing of these SSC.

Section 10.2.2 and Table 10.2-1 provide details regarding the type of turbine to be used in the NuScale design. Using the design and material specifications that appear in Section 10.2.2 and Table 10.2-1, the turbine missiles selected for evaluation included:

- A turbine blade weighing 32.6 lbs with an equivalent diameter of 1.41 inches, and a velocity of 1150 ft/s.
- A turbine blade with a rotor fragment weighing 52.6 lbs with a rotor width of 4.5 inches, and a velocity of 1461 ft/s.
- Half of the last stage of the turbine rotor weighing 3079 lbs that is 48 inches in diameter by 12 inches wide, and a velocity of 512 ft/s.

COL Item 3.5-1: A COL applicant that references the NuScale Power Plant design certification will demonstrate that the site-specific turbine missile parameters are bounded by the DC analysis, or provide a missile analysis using the site-specific turbine generator parameters to demonstrate that barriers adequately protect essential SSC from turbine missiles.

COL Item 3.5-2: A COL applicant that references the NuScale Power Plant design certification will address the effect of turbine missiles from nearby or co-located facilities.

3.5.1.4 Missiles Generated by Tornadoes and Extreme Winds

Hurricane and tornado generated missiles are evaluated in the design of safety-related structures and risk-significant SSC outside those structures. The missiles used in the evaluation are assumed to be capable of striking in all directions and conform to the Region I missile spectrums presented in Table 2 of RG 1.76, Rev. 1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants" for tornado missiles and Table 1 and Table 2 of RG 1.221, Rev. 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," for hurricane missiles. These spectra are based on the design basis tornado and hurricane defined in Section 3.3.2 and represent probability of exceedance events of 1×10^{-7} per year for most potential sites.

The selected missiles include

- A massive high-kinetic-energy missile that deforms on impact, such as an automobile.

  The "automobile" missile is 16.4 feet by 6.6 feet by 4.3 feet with a weight of 4000 lbs. and a C_DA/m (drag coefficient x projected area/mass) of 0.0343 ft^2/lb.

  This missile has a horizontal velocity of 135 ft/s and a vertical velocity of 91 ft/s in a tornado; and corresponding velocities of 307 ft/s and 85 ft/s, respectively, in a hurricane.

  The automobile missile is considered capable of impact at all altitudes less than 30 ft above all grade levels within 1/2 mile of the plant structures.
- A rigid missile that tests penetration resistance, such as a six-inch diameter Schedule 40 pipe.

  The "pipe" missile is 6.625 inch diameter by 15 feet long with a weight of 287 lbs. and a C_DA/m of 0.0212 ft^2/lb.

  This missile has a horizontal velocity of 135 ft/s and a vertical velocity of 91 ft/s in a tornado; and corresponding velocities of 251 ft/s and 85 ft/s, respectively, in a hurricane.
- A one-inch diameter solid steel sphere to test the configuration of openings in protective barriers.

  The "sphere" missile is 1 inch in diameter with a weight of 0.147 lbs. and a C_DA/m of 0.0166 ft^2/lb.

  This missile has a horizontal velocity of 26 ft/s and a vertical velocity of 18 ft/s in a tornado; and corresponding velocities of 225 ft/s and 85 ft/s, respectively, in a hurricane.

These missile parameters are key design parameters and are provided in Table 2.0-1.

3.5.1.5 Site Proximity Missiles (Except Aircraft)

As described in Section 2.2, the NuScale Power Plant certified design does not postulate any hazards from nearby industrial, transportation or military facilities. Therefore, there are no proximity missiles.

3.5.1.6 Aircraft Hazards

As described in Section 2.2, the NuScale Power Plant certified design does not postulate any hazards from nearby industrial, transportation or military facilities. Therefore, there are no design basis Aircraft Hazards. Discussion of the beyond design basis Aircraft Impact Assessment is provided in Section 19.5.

3.5.2 Structures, Systems, and Components to be Protected from External Missiles

All safety-related and risk-significant SSC that must be protected from external missiles are located inside the seismic Category I RXB and Seismic Category I portions of the CRB. The concrete walls and roof of the RXB and the CRB below the 30 ft above plant grade threshold are designed to withstand all design basis missiles discussed in Section 3.5.1.3 and Section 3.5.1.4. The portions of the RXB and the CRB that are above 30 ft plant elevation have not been analyzed to withstand the design basis automobile missile, but are resistant to the other design basis missiles discussed in Section 3.5.1.4. Section 3.8 provides additional information for the design of RXB and CRB.

COL Item 3.5-3: A COL applicant that references the NuScale Power Plant design certification will confirm that automobile missiles cannot be generated within a 0.5-mile radius of safety-related structures, systems, and components and risk-significant structures, systems, and components requiring missile protection that would lead to impact higher than 30 feet above plant grade. Additionally, if automobile missiles impact at higher than 30 feet above plant grade, the COL applicant will evaluate and show that the missiles will not compromise safety-related and risk-significant structures, systems, and components.

The RXB and CRB meet the requirements of the RG 1.13, Rev. 2, "Spent Fuel Storage Facility Design Basis", RG 1.117, Rev. 2, "Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants," and RG 1.221, Revision 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants" for protection of SSC from wind, tornado and hurricane missiles.

The RXB and CRB have been credited to withstand turbine missiles.

COL Item 3.5-4: A COL Applicant that references the NuScale Power Plant design certification will evaluate site-specific hazards for external events that may produce more energetic missiles than the design basis missiles defined in FSAR Tier 2, Section 3.5.1.4.

3.5.3 Barrier Design Procedures

In the design, there are a limited number of potential internal missiles and a limited number of targets. If a missile/target combination is determined to be statistically significant (i.e., the product of (P_1), (P_2) and (P_3) is greater than 10^{-7} per year), barriers are installed.

Safety-related and risk-significant SSC are protected from missiles by ensuring the barriers have sufficient thickness to prevent penetration and spalling, perforation, and scabbing that could challenge the SSC. Missile barriers are designed to withstand local and overall effects of missile impact loadings. The barrier design procedures discussed below may be used for both internal and external missiles.

3.5.3.1 Local Damage Prediction

The prediction of local damage in the impact area depends on the basic material of construction of the structure or barrier (i.e., concrete, steel, or composite). The analysis approach for each basic type of material is presented separately. It is assumed that the missile impacts normal to the plane of the wall on a minimum impact area.

3.5.3.1.1 Concrete Barriers

Concrete missile barriers are evaluated for the effects of missile impact resulting in penetration, perforation, and scabbing of the concrete using the Modified National Defense Research Committee (NDRC) formulas discussed in "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," (Reference 3.5-3) as described in the following paragraphs. Concrete barrier thicknesses calculated using the equations in this section for perforation and scabbing are increased by 20 percent. The NDRC formulas were not used for determining penetration distance for the design basis turbine blade and blade with rotor fragment missiles. Instead, a finite element analysis was used because the Modified NDRC equations are based on the assumption that the missile and target are essentially non-deformable, which is not appropriate for a turbine blade which will deform. Using NDRC equations for deformable missiles over-predicts the penetration distance. After the penetration distance is determined with the finite element model, the Modified NDRC formulas are used to determine perforation and scabbing thicknesses.

Concrete thicknesses to preclude perforation or scabbing from the design basis hurricane and tornado pipe and sphere missiles have been calculated for the 5000 psi and 7000 psi concrete used for the RXB, CRB and RWB external walls and roof using the below equations. The design basis hurricane and tornado automobile missile is incapable of producing significant local damage; therefore, it is not considered. The same is true for the design basis turbine missile of half of the last stage of the turbine rotor. The wind and tornado missile results are tabulated in Table 3.5-1. The RXB has five foot thick outer walls and a four foot thick roof. The missile protected portions of the CRB have three foot thick exterior walls and roof, consisting of a concrete slab with a steel cover, and the RWB has exterior walls that are two feet thick above grade and has a one foot thick roof. The local results for the design basis turbine missile are presented in Table 3.5-2 through Table 3.5-4.

Additional design characteristics of the RXB and the CRB are provided in Section 3B.2. The RWB exterior walls are 5000 psi concrete reinforced with a minimum of #8 reinforcing bars on 12-inch centers.

3.5.3.1.1.1 Penetration and Spalling Equations

The depth of missile penetration, x, is calculated using the following formulas:

$$x = \left[4KNWd\left\langle\frac{V}{1000d}\right\rangle^{1.8}\right]^{0.5} \text{ for } \frac{x}{d} \leq 2.0 \qquad \text{Eq. 3.5-1}$$

$$x = KNW\left(\frac{V}{1000d}\right)^{1.8} + d \text{ for } \frac{x}{d} \geq 2.0 \qquad \text{Eq. 3.5-2}$$

where,

x = penetration depth, in,

W = missile weight, lb,

d = effective missile diameter, in,

N = Missile shape factor:

- flat nosed bodies = 0.72,
- blunt nosed bodies = 0.84,
- average bullet nose (spherical end) = 1.00,
- very sharp nosed bodies = 1.14,

V = Velocity, ft/sec,

K = $180/(\sqrt{f'_c})$, and

f'_c = concrete compressive strength (lb/in2).

3.5.3.1.1.2 Perforation Equations

The relationship for perforation thickness, t_p (inches), and penetration depth, x, is determined from the following formulas:

$$t_p/d = 3.19(x/d) - 0.718(x/d)^2 \text{ for } (x/d) < 1.35$$

$$t_p/d = 1.32 + 1.24(x/d) \text{ for } 1.35 \le (x/d) \le 13.5$$

3.5.3.1.1.3 Scabbing Equations

The relationship for scabbing thickness, t_s (inches), and penetration depth, x, is determined from the following formulas:

$$t_s/d = 7.91(x/d) - 5.06(x/d)^2 \text{ for } (x/d) < 0.65$$

$$t_s/d = 2.12 + 1.36(x/d) \text{ for } 0.65 \le (x/d) \le 11.7$$

3.5.3.1.2 Steel Barriers

There are no steel missile barriers used in the design.

3.5.3.1.3 Composite Barriers

The design does not use composite barriers.

3.5.3.2 Overall Damage Prediction

For predicting overall damage, a dynamic impulse load concentrated at the impact area is determined and applied as a forcing function to determine the structural response.

The forcing functions to determine the structural responses are derived using EPRI NP440, "Full Scale Tornado Missile Impact Tests," (Reference 3.5-9) for the triangular impulse formulation of the design basis steel pipe missile. BC-TOP-9A, Rev. 2, "Design of Structures for Missile Impact," (Reference 3.5-8) is used for the design basis automobile missile and design basis half of the last stage turbine rotor. The solid sphere missile and turbine blades are too small to affect the structural response of the RXB and the CRB and were not evaluated for their contribution to overall structural response.

The automobile missile forcing functions are applied to the building models in selected locations using the horizontal impact loads since they are higher than the vertical loads. The results are addressed in Section 3.8.4.

The weights and velocity of an automobile missile and half of the last stage rotor are similar. Equating the turbine rotor to an equivalent static force resulted in the following:

For the RXB, the flexural demand to capacity ratio (DCR) is 0.18, shear DCR is 0.16, and wall deflection is 0.02 inches.

For the CRB, the flexural DCR is 0.81, shear DCR is 1.46, and deflection is 0.25 inches. All values are based on the exterior wall only. It is observed that the CRB exterior wall is not sufficient to prevent a shear failure that results from the impact of a turbine rotor missile. However, the penetration opening that could be developed is smaller than the size of a personnel door and after impact the normal operating loads will redistribute to the redundant structural members adjacent to the impact location to prevent further damage. In addition, it is anticipated that the exterior wall will suffice to reduce the velocity by a significant margin. The remaining turbine missile will then be contained by the three-foot-thick concrete grade-level floor; this conclusion is determined by inspection due to the significant loss of energy in the missile, and the fact that the strike will occur at not less than a 45 degree angle (which has a substantial influence on the penetration depth). The combination of the exterior wall and grade-level floor serve adequately to protect essential SSCs, which are located below grade in the CRB.

Finite element analyses of the automobile missile has shown to have an insignificant effect on the global response of either structure. Given the similarity of the turbine missile to the automobile missile, the analysis is considered valid for evaluating the effect from the turbine missile. The base reactions, joint displacements, and deformed shape of both the RXB and CRB support this conclusion.

Design for impulsive and impactive loads is in accordance with ACI 349 "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," (Reference 3.5-6) for concrete structures and AISC N690 "Specification for Safety-Related Steel Structures for Nuclear Facilities," (Reference 3.5-7) for steel structures except for the modifications listed below.

Stress and strain limits for the missile impact equivalent static load comply with applicable codes and RG 1.142, Rev. 2 "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," and the limits on ductility of steel structures are given as noted below.

Concrete

Structural concrete members designed to resist missile impact are designed for flexural, shear, spalling, scabbing, and perforation effects using the equivalent static load obtained for the evaluation of structural response.

The permissible ductility for beams, walls, and slabs subjected to impulsive or impactive loads, if flexure controls the design, is in accordance with Section F.3.3 of ACI-349.

In Section F.3.5 of ACI-349, the permissible ductility ratio (μ), when a concrete structure is subjected to a pressure pulse due to compartment pressurization, is as follows, based on RG 1.142:

1) for the structure as a whole, $\mu \leq 1.0$

2) for localized area in the structure (ductility in flexure), $\mu \leq 3.0$

In Section F.3.7 of ACI-349 where shear controls the design, the permissible ductility ratio is as follows, based on RG 1.142:

1) when shear is carried by concrete alone, $\mu \leq 1.0$

2) when shear is carried by combination of concrete and stirrups or bent bar, $\mu \leq 1.3$

3) when shear is carried completely by stirrups, $\mu \leq 3.0$

In Section F.3.8 of ACI-349, the maximum permissible ductility ratio in flexure is as follows, based on RG 1.142.

1) When the compressive load is greater than 0.1 $f'_c A_g$ or one-third of that which would produce balanced conditions, whichever is smaller, the maximum permissible ductility ratio should be 1.0.

2) When the compressive load is less than 0.1 $f'_c A_g$ or one-third of that which would produce balanced conditions, whichever is smaller, the permissible ductility ratio should be as given in F.3.3 or F.3.4 of ACI-349.

3) The permissible ductility ratio should vary linearly from 1.0 to that given in F.3.3 or F.3.4 of ACI-349 for condition between specified in 1 and 2.

Steel

Structural steel members designed to resist missile impact are designed for flexural, shear, buckling and perforation effects using the equivalent static load obtained for the evaluation of structural response.

Based on Section NB3.15 of AISC N690, the following ductility factors (μ) from Table NB3.1 are used.

1) For steel tension members, $\mu \leq \frac{0.25\varepsilon_{\mu}}{\varepsilon_y} \leq \frac{0.1}{\varepsilon_y}$

   a) ε_u = strain corresponding to elongation at failure (rupture)

   b) ε_y =strain corresponding to yield stress

2) For structural steel flexural members:

   a) Open sections (W, S, WT, etc.), $\mu \leq 10$

   b) Closed sections (pipe, box, etc.), $\mu \leq 20$

   c) Members where shear governs design $\mu \leq 5$

3) Structural steel columns, $\mu = 0.225/(F_y/F_e)\ \varepsilon_{st}/\varepsilon_y$ (not to exceed 10)

   a) $F_e = \pi^2 E/(KL_e/r)^2$

   b) F_y = yield strength of steel member

   c) ε_{st} = strain corresponding to the onset of strain hardening

In determining an appropriate equivalent static load for (Y_r), (Y_j) and (Y_m), elasto-plastic behavior may be assumed with permissible ductility ratios as long as deflections do not result in loss of function of any safety-related system.

Section 3.8 provides additional information on loading combinations and analysis methods for the RXB and CRB.

3.5.4 References

3.5-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Rules for Construction of Nuclear Facility Components," 2013 edition with no Addenda (subject to the conditions specified in paragraph (b)(1) of section 50.55a), Section III, New York, NY.

3.5-2 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2013 edition with no Addenda (subject to the conditions specified in paragraph (b)(2) of section 50.55a), Section XI, New York, NY.

3.5-3 Kennedy, R.P., "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," Nuclear Engineering and Designs, (1976) 37:2, 183-203.

3.5-4 Cottrell, W.B., and A.W. Savolainen, "U.S. Reactor Containment Technology," Volume 1, Chapter 6, ORNL NSIC-5, Oak Ridge National Laboratory, Oak Ridge, TN, 1965.

3.5-5 Russel, C.R., Reactor Safeguards, MacMillian, New York, 1962.

3.5-6 American Concrete Institute, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," ACI 349-06, Farmington Hills, MI.

3.5-7 American Institute of Steel Construction, "Specification for Safety-Related Steel Structures for Nuclear Facilities," AISC N690, 2012, Chicago, IL.

3.5-8 Bechtel Power Corporation, "Design of Structures for Missile Impact," BC-TOP-9A, Rev. 2, San Francisco, CA, September 1974.

3.5-9 Electric Power Research Institute, "Full Scale Tornado Missile Impact Tests," EPRI NP440, Palo Alto, CA July 1977.

Table 3.5-1: Concrete Thickness to Preclude Missile Penetration, Perforation, or Scabbing

Direction	Missile	N	W (lbs)	D (in.)	V (ft/s)	Concrete Strength (psi)	Penetration Distance (in.)	Perforation Distance (in.)	Thickness to Preclude Scabbing	Building	Wall/Roof Thickness (in.)
horizontal	pipe	0.84	287	6.625	251	7000	6.2	18.6	23.7	RXB	60
						5000	6.7	19.8	27.8 from EC-F170-3650, Rev 1	CRB	36
										RWB	24
	sphere	1.00	0.147	1	224	7000	0.3	1.1	2.3	RXB	60
						5000	0.3	1.1	2.4	CRB	36
										RWB	24
vertical	pipe	0.84	287	6.625	91	5000	2.7	9.4	18.9	RXB	48
										CRB	36
										RWB	12
	sphere	1.00	0.147	1	85	5000	0.1	0.5	1.2	RXB	48
										CRB	36
										RWB	12

Table 3.5-2: Summary of Barrier Thickness for Turbine Missile Penetration

Overspeed	Velocity (mph)	Penetration (inch)	Required Barrier Thickness (inch)
120%	747	17.0	20.4
140%	872	21.5	25.8
160%	996	24.0	28.8
180%	1121	26.0	31.2
190%	1183	25.5	30.6
200%	1245	26.0	31.2
210%	1308	28.5	34.2
220%	1370	28.5	34.2

Table 3.5-3: Summary of Barrier Thickness for Turbine Missile Perforation

Overspeed	x Penetration FEA results (inch)	d* Missile Diameter (inch)	x/d*	t_p (inch)	Required Barrier Thickness (inch)
120%	17.0	1.41	12.1	22.9	27.5
		3	5.7	25.	30.3
140%	21.5	1.41	15.2	28.5	34.2
		3	7.2	30.6	36.7
160%	24.0	1.41	17.0	31.6	37.9
		3	8.0	33.7	40.5
180%	26.0	1.41	18.4	34.1	40.9
		3	8.7	36.2	43.4
190%	25.5	1.41	18.1	33.5	40.2
		3	8.5	35.6	42.7
200%	26.0	1.41	18.4	34.1	40.9
		3	8.7	36.2	43.4
210%	28.5	1.41	20.2	37.2	44.6
		3	9.5	39.3	47.2
220%	28.5	1.41	20.2	37.2	44.6
		3	9.5	39.3	47.2

d* = conservative equivalent diameter used in perforation and scabbing equations

Table 3.5-4: Summary of Barrier Thickness for Turbine Missile Scabbing

Overspeed	x Penetration FEA results (inch)	d* Missile Diameter (inch)	x/d*	t_p (inch)	Required Barrier Thickness (inch)
120%	17.0	1.41	12.1	26.1	31.3
		3	5.7	29.5	35.4
140%	21.5	1.41	15.2	32.2	38.7
		3	7.2	35.6	42.7
160%	24.0	1.41	17.0	35.6	42.8
		3	8.0	39.0	46.8
180%	26.0	1.41	18.4	38.3	46.0
		3	8.7	41.7	50.1
190%	25.5	1.41	18.1	37.7	45.2
		3	8.5	41.0	49.2
200%	26.0	1.41	18.4	38.3	46.0
		3	8.7	41.7	50.1
210%	28.5	1.41	20.2	41.7	50.1
		3	9.5	45.1	54.1
220%	28.5	1.41	20.2	41.7	50.1
		3	9.5	45.1	54.1

d* = conservative equivalent diameter used in perforation and scabbing equations

3.6 Protection against Dynamic Effects Associated with Postulated Rupture of Piping

This section describes the design bases and measures needed to protect safety-related and essential systems and components inside and outside containment against the effects of postulated pipe rupture. Figure 3.6-1 is a flowchart depicting the steps in the process for evaluation of potential line breaks. The NuScale methodology applicable to identification and assessment of pipe rupture hazards addresses determination of postulated rupture locations, characteristics of ruptures, and assessment of the possible dynamic and external effects of ruptures. Details of the analyses are provided in the Pipe Rupture Hazards Analysis Technical Report (Reference 3.6-21).

Pipe rupture protection is provided according to the requirements of 10 CFR 50, Appendix A, General Design Criterion 4. In the event of a high- or moderate-energy pipe rupture within the NuScale Power Module (NPM), adequate protection is provided so that safety-related and essential structures, systems, and components (SSC) are not unacceptably affected. Essential systems and components are those required to shut down the reactor and mitigate the consequences of the postulated piping rupture. Nonsafety-related systems are not required to be protected from the dynamic and environmental effects associated with the postulated rupture of piping except as necessary to preclude adverse effect on an essential system. In addition, although neither safety-related nor essential, the post-accident monitoring (PAM) functionality provided by various portions of the instrumentation and control (I&C) systems is protected.

The criteria used to evaluate pipe rupture protection are generally consistent with NRC guidelines including those in the Standard Review Plan Section 3.6.1, Section 3.6.2, and Section 3.6.3, NUREG-1061, Vol. 3, and applicable Branch Technical Positions (BTPs), as discussed within this section.

Section 3.6.1 identifies the high- and moderate-energy lines that have a potential to affect safety-related and essential SSC, and describes the approaches used in the NuScale Power Plant design for protection of these SSC. Section 3.6.2 describes the analytical methodology used to determine break locations, identifies postulated breaks, and discusses the consequences of those breaks and the effect on SSC functionality. Section 3.6.3 describes the leak-before-break (LBB) analysis for applicable piping systems inside containment. Section 3.6.4 discusses the analysis of non-LBB high- and moderate-energy piping.

3.6.1 Plant Design for Protection against Postulated Piping Ruptures in Fluid Systems

General Design Criterion (GDC) 4 requires that SSC be 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 (LOCAs). This includes both environmental effects (temperature changes, pressure changes, humidity changes, and flooding) from line breaks and leakage cracks and dynamic effects (blast, pressurization, pipe whip and jet impingement) that may result from high-energy line breaks (HELB).

Plant designers are provided with options to address GDC 4 for pipe ruptures. These options are as follows:

- On a limited basis, portions of pipe may be excluded from postulating breaks and cracks provided they meet criteria regarding the design arrangement, stress and fatigue limits, and a high level of inservice inspection (ISI). The criteria for this exclusion are provided in BTP 3-4, "Fluid System Piping in Containment Penetration Areas," Section B.A.(ii).
- Systems that can demonstrate a low probability of rupture prior to the detection of a leak may be excluded from HELB dynamic effect considerations. This is referred to as LBB analysis and is discussed in SRP 3.6.3. LBB is applied to high-energy piping systems having well-characterized loading conditions and load combinations. This method is an acceptable design approach provided that plant design and specific analyses have indicated a low probability of rupture from damage mechanisms such as water hammer, steam hammer, stress corrosion cracking, and fatigue.
- For high- and moderate-energy systems that cannot be fully excluded using criteria of BTP 3-4 Section B.A.(ii) or LBB, line breaks and leakage cracks are postulated. The criteria for the specific locations for the postulated breaks are provided in BTP 3-4 (e.g., Section B.A.(iii)). In general, locations meeting certain stress, fatigue and design requirements may be excluded and are not required to be postulated to rupture. Other locations, such as terminal ends or high-stress locations, must be postulated to rupture. At break locations, the piping systems are located such that there is no safety-related or essential equipment in the area (i.e., separation), or safety-related and essential SSC are shown to be protected from exposure to break effects or otherwise not unacceptably affected.

The piping systems that must be considered include the ASME Section III Class 1, 2, 3, and ASME B31.1, high-energy and moderate-energy systems located inside and outside of the containment vessel (CNV). Table 3.6-1 identifies the high- and moderate-energy piping systems and associated plant locations. Breaks and leakage cracks need not be postulated in high- and moderate-energy lines that are NPS 1 and smaller.

High-energy lines are evaluated for both line breaks and through-wall leakage cracks. Line breaks include both circumferential (complete rupture around the circumference of the pipe) and longitudinal breaks (rupture of the pipe along its axis). Line breaks are analyzed for dynamic and environmental effects. Through-wall leakage cracks are analyzed for flooding and environmental effects.

Moderate-energy lines are evaluated for through-wall leakage cracks and analyzed for flooding and environmental effects.

Additionally, the environmental effects of nonmechanistic breaks of main steam system (MSS) and feedwater system (FWS) piping in the containment penetration area are evaluated.

Locations having the greatest effect on essential equipment are chosen for evaluation of impacts.

Flooding is discussed in Section 3.4. Environmental effects are discussed in Section 3.11. Analysis of subcompartment pressurization effects within the CNV are discussed in Appendix 3A.

3.6.1.1 Identification of High- and Moderate-Energy Piping Systems

High-energy fluid systems include those systems or portions of systems where either of the following conditions is met:

- the maximum operating temperature exceeds 200 degrees F, or
- the maximum operating pressure exceeds 275 psig

Moderate-energy fluid systems include systems or portions of systems pressurized above atmospheric pressure during normal plant conditions but do not meet the criteria for high-energy systems. Moderate-energy fluid systems are those systems where both of the following conditions are met: (a) the maximum operating temperature is 200 degrees F or less, and (b) the maximum operating pressure is 275 psig or less. In addition, piping systems that exceed 200 degrees F or 275 psig for 2 percent or less of the time during which the system is in operation or that experience high-energy pressures or temperatures for less than 1 percent of the plant operation time are also considered moderate-energy.

Table 3.6-1 provides a list of high- and moderate-energy piping systems and identifies the areas where the systems are located. The areas of the plant that contain high- and moderate-energy lines, or safety-related and essential SSC are considered in six groups. Each is discussed in a separate section.

- inside the CNV (Section 3.6.1.1.1)
- outside the CNV (under the bioshield) (Section 3.6.1.1.2)
- in the Reactor Building (RXB), (outside the bioshield) (Section 3.6.1.1.3)
- in the Control Building (CRB) (Section 3.6.1.1.4)
- in the Radioactive Waste Building (RWB) (Section 3.6.1.1.5)
- onsite (outside the buildings) (Section 3.6.1.1.6)

Table 3.6-1 identifies the largest piping line size and the highest normal operating pressure and temperature of the fluid system to assign an energy classification. The energy classification and line size do not necessarily correspond to the same region of the fluid system.

While Table 3.6-1 provides a listing of the high- and moderate-energy systems outside of the NPM, the piping line size and energy classification may vary from these maximum values at the postulated rupture location. COL Item 3.6-1 requires that the COL applicant confirm the content of Table 3.6-1 following the performance of the balance of plant PRHA, or update it accordingly.

Figure 6.6-1 shows the high- and moderate-energy lines that interface with the CNV. Generally, the portions of these lines from the NPM disconnect flanges up to and including the CNV penetration are considered to be part of the containment system (CNTS). Inside the CNV, the lines are considered to be part of a different system. The main steam and feedwater lines are part of the steam generator system (SGS) inside containment. The chemical and volume control system (CVCS) lines are part of the reactor coolant system (RCS) inside the CNV, and include the RCS injection, RCS discharage, pressurizer (PZR) spray supply, and reactor pressure vessel (RPV) high point degasification lines. The reactor component cooling water system (RCCWS) supply and return lines are part of the control rod drive system (CRDS) inside the CNV.

The decay heat removal system (DHRS) piping is a high-energy system only associated with the NPM.

The containment flooding and drain system (CFDS) is a single open pipe inside containment that is normally isolated and not pressurized during operation. This line is moderate-energy based on the amount of time in use. This line is identified as the CNTS flooding and drain line both inside and outside the CNV.

Generally, in this Section a particular portion of piping is referred to by its functional name (e.g., main steam, RCCWS) regardless of whether that portion is inside the CNV, a part of the CNTS, or outside the NPM.

3.6.1.1.1 Inside the Containment Vessel

The high-energy lines inside the CNV are: main steam, feedwater, RCS injection, RCS discharge, high point degasification, PZR spray supply and DHRS condensate return. There are two moderate-energy lines inside the CNV, the RCCWS supply and return lines and the CFDS line (See Table 3.6-1). The ECCS includes several small hydraulic lines inside containment that run between the ECCS valves, the Trip/Reset valves, and the RCS injection line. These high-energy ECCS lines are excluded from consideration as they are smaller than NPS 1.

3.6.1.1.2 Outside the Containment Vessel (Under the Bioshield)

The high-energy lines (main steam, feedwater, RCS injection, RCS discharge, high point degasification, PZR spray supply and DHRS) and the moderate-energy lines (CRDS, CFDS, and the containment evacuation system (CES)) continue outside containment to the NPM disconnect flange (See Table 3.6-1).

The DHRS steam line connects to the MSS line outside containment, immediately upstream of the MSS containment isolation valve and leads to the DHRS condenser and then to the DHRS condensate return lines. Although not normally in use, this entire system is pressurized during NPM operation.

3.6.1.1.3 In the Reactor Building (Outside the Bioshield)

Within the RXB, but outside the area under the bioshield, the high-energy lines include the MSS, FWS, and CVCS lines, and additional high-energy lines associated with the auxiliary boiler and process sampling system (PSS) (See Table 3.6-1). Based on limited operating time, the auxiliary boiler lines are considered moderate-energy. Based on the nominal diameter of the PSS lines, breaks do not need to be postulated.

The high-energy MSS and FWS lines exit the reactor pool through the North and South reactor pool walls, cross a mechanical equipment area (pipe gallery) and exit the RXB.

Once they exit the area under the bioshield, the high-energy CVCS lines run vertically downward in a pipe chase to the CVCS heat exchanger rooms at elevation 50' 0'' and associated CVCS rooms at Elevations 24′ 0″ and 35' 6". The pipe chase can be seen on the general arrangement drawings in Section 1.2.

Moderate-energy lines are routed throughout the RXB (See Table 3.6-1).

3.6.1.1.4 In the Control Building

There are no high-energy lines in the CRB. There are three moderate-energy lines: fire protection, chilled water, and potable water (See Table 3.6-1).

3.6.1.1.5 In the Radioactive Waste Building

There are no high-energy lines in the RWB. There are two moderate-energy lines: fire protection and liquid radioactive waste management (See Table 3.6-1).

3.6.1.1.6 Onsite (outside the buildings)

Outside of the RXB and CRB there are three high-energy lines: MSS, FWS, and extraction steam, and multiple moderate-energy lines (See Table 3.6-1).

There is no safety-related or essential equipment in the area outside of the RXB or CRB. Final routing of piping outside of the RXB, CRB, and RWB is the responsibility of the COL applicant.

COL Item 3.6-1: A COL applicant that references the NuScale Power Plant design certification will complete the routing of piping systems outside of the containment vessel and the area under the bioshield, identify the location of high- and moderate-energy lines, and update Table 3.6-1 as necessary. This activity includes the performance of associated final piping stress analyses, design and qualification of associated piping supports, evaluation of subcompartment pressurization effects (if applicable), and completion of the Balance of Plant Pipe Rupture Hazards Analysis, including the design and evaluation of pipe whip/jet impingement mitigation devices as required. This includes an evaluation and disposition of multi-module impacts in common pipe galleries.

3.6.1.2 Identification of Safety-Related and Essential Structures, Systems, and Components

By design, the NuScale Power Plant only has a small number of safety-related and essential SSC. These SSC are primarily associated with the NPM, either inside the CNV or mounted on the top of the CNV head.

Shutdown of the reactor requires the following systems be protected from HELB:

- RCS
- module protection system (MPS)
- neutron monitoring system

- SGS
- CVCS
- control rod assembly and the CRDS
- CNTS
- DHRS
- emergency core cooling system (ECCS)
- ultimate heat sink / reactor pool

Of these, only the CNTS, DHRS, ECCS, and ultimate heat sink/reactor pool are needed following reactor shutdown. In addition, PAM functionality for Type B and C variables (there are no Type A variables) is protected.

3.6.1.3 Characteristics of the NuScale Design

The NuScale design is an integral, multi-unit, small modular reactor for which safety is provided by passive features without the need for safety-related electrical power. Because NRC regulatory guidance for HELB is premised on the existing fleet of large light water reactors with reactor coolant loops and active safety features, instances exist where the current NRC pipe rupture guidance is not a direct fit. In many cases, the NRC has not issued a Design-Specific Review Standard to address what is directly applicable for the NuScale design.

Specific examples of relevant design differences are:

- The response to HELB for a NuScale plant requires neither electric power nor injection of additional cooling water.
- The NPMs are mostly submerged in a large pool of water that serves as the ultimate heat sink and does not require replenishment for design-basis events.
- Design-basis accidents do not require operator actions or re-establishing electric power for long-term cooling.
- Piping is small compared to the large reactors for which regulatory guidance was initially developed.
- Active safety-related components (e.g., ECCS valves, DHRS actuation valves, and containment isolation valves (CIVs)) are shown to operate during refueling. As part of the start-up sequence for an NPM, each of the safety-related ECCS, DHRS, and containment isolation valves is repositioned. These system line-up activities provide assurance the safety-related valves are operable.
- The NPM containment is a pressure vessel designed and constructed to ASME Code Section III Class 1 requirements versus a building in conventional LWRs.
- Piping of the NPM, including secondary system piping, is made of corrosion-resistant stainless steel.
- MSS and FWS piping inside the containment boundary and under the bioshield is designed to RCS design pressure and temperature.
- MSS and FWS piping inside the CNV meets LBB criteria.

- HELB inside the CNV are limited to NPS 2 piping.
- The length of piping in which breaks must be postulated is minimal and the size of high-energy piping is small compared to current design reactors.
- The NPM containment is operated at a vacuum.
- Equipment and piping inside the NPM containment are not covered by insulation. This is important for multiple reasons:
  - Jet impingement does not dislodge insulation that could lead to blockage of long-term-cooling recirculation.
  - Detection of small leakage cracks is not impeded by retention of moisture in insulation.
  - The bare piping is readily inspectable during refueling, because insulation does not need to be removed to observe deposits, discoloration, or other signs of degradation.
  - Corrosive substances (e.g., chlorides) cannot be trapped and held in contact with the piping surface.
- Safety-related and essential components inside the NPM containment are qualified to be functional after exposure to saturated steam at containment design pressure of at least 1000 psia, requiring designs that are robust.
- The small NPM containment results in congestion that makes difficult the addition of traditional piping restraints and the separation of essential components from break locations, but whipping pipes in turn have a limited range of motion before encountering an obstacle.
- Containment isolation valves are outside of containment. Where two valves in series are required (i.e., for containment penetrations governed by GDC 55 and 56), both are in a single-piece valve body (i.e., no piping or welds between CIVs, precluding breaks in between). Also, the lines directly connected to the primary system or the containment have only a single piping weld in the area between the containment wall and the CIV.
- The RCS-connected lines (i.e., CVCS), except for the normally isolated RPV high-point degasification line, have check (or excess flow check) valves immediately outside the CIVs to preserve reactor coolant inventory in case of LOCAs outside containment.
- Containment pressure suppression is not required, and there are no sprays that introduce chemical additives.
- During a refueling, the NPM is disconnected from supporting systems by removal of piping spools, transported by crane to a refueling location, and disassembled. This provides access for inspection to portions of the plant not normally accessible.
- Up to 12 NPMs are operating at the same time and in proximity, so the potential for a rupture in a system of one module to affect others is considered.
- The plant main control room is in a separate building that does not contain high energy piping systems.

- Dynamic effects of postulated HELB on multiple modules are evaluated, and protection for PAM capability and reliable DC power is provided by separation in different compartments within the building.

These unique characteristics affect choices about the means to address HELB.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

This section describes the criteria and methods used to postulate break and leakage crack locations in high-energy and moderate-energy piping inside and outside containment, the methodology used to define potential blast effects, the thrust at the postulated break location, potential pipe whip, the jet impingement loading on adjacent essential safety-related SSC and subcompartment pressurization resulting from fluid blowdown.

General Design Criterion 4 requires that SSC important to safety both accommodate the effects of, and are compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. In the event of a high-energy or moderate-energy pipe rupture within the plant, GDC 4 requires that adequate protection is provided so that essential SSC are not impacted unacceptably by the adverse effects of the rupture. Nonsafety-related systems are not required to be protected from the dynamic and environmental effects associated with the postulated rupture of piping. Compliance with GDC 4 is demonstrated through conformance with the criteria of BTP 3-4 as described in Section 3.6.2.1.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

Branch Technical Position 3-4 provides guidance on the selection of the break locations within a piping system. The types of breaks postulated in high-energy lines include circumferential breaks in fluid system piping greater than 1 inch NPS; longitudinal breaks in fluid system piping that is 4-inch NPS and greater, and leakage cracks in fluid system piping greater than 1-inch NPS. Leakage cracks are also postulated in moderate-energy lines.

The pipe break criteria of BTP 3-4 include the requirement that breaks be postulated at terminal ends. The definition of a terminal end, consistent with BTP 3-4, is the extremity of a piping run that connects to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection on a main piping run is a terminal end for the branch run, except where the branch run is classified as part of a main run in the stress analysis or is shown to have a significant effect on the main run behavior. In piping runs that are maintained pressurized during normal plant conditions for a portion of the run (i.e., up to the first normally closed valve), a terminal end is the piping connection to this closed valve.

General Design Criterion 4 allows dynamic effects associated with postulated pipe ruptures to be 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. This is referred to as LBB analyses. This is discussed in Section 3.6.3. Similarly, breaks and leakage cracks may be excluded within the containment penetration area if criteria of BTP 3-4 B.A.(ii) are met.

3.6.2.1.1 Pipe Breaks Inside the Containment Vessel

The CIVs are outside the containment. A break inside the CNV does not lead to containment bypass. Therefore, there is no containment penetration area inside the CNV and BTP 3-4 B.A.(ii) does not apply. Due to the tight configuration and the concentration of safety-related and essential SSC inside the CNV, dynamic effects of pipe breaks are assessed for specific locations. The following strategies are employed for HELB inside containment:

- The main steam and feedwater lines meet the criteria for LBB (see Section 3.6.3). Therefore, circumferential and longitudinal breaks are not postulated for dynamic effects for the MSS and FWS lines inside containment.
- The RCS injection, RCS discharge, PZR spray supply, and high-point degasification lines inside containment are NPS 2, Schedule 160, ASME Class 1 stainless steel pipes. Due to their size, longitudinal breaks are not postulated. Circumferential breaks are postulated in accordance with BTP 3-4 Section B.A.(iii)(1). Breaks in Class 1 high-energy piping systems are postulated at the following locations:
  a) terminal ends (defined in Section 3.6.2.1)
  b) intermediate locations where the maximum stress range exceeds 2.4 S_m as calculated by equation (10) and either equation (12) or (13) of NB-3653 of Section III of the ASME Boiler and Pressure Vessel Code.
  c) intermediate locations where the cumulative usage factor exceeds 0.1, unless environmentally assisted fatigue is considered in which case the cumulative usage factor exceeds 0.4.
- The DHRS condensate lines inside containment run from each feedwater line, just upstream of the feed plenum, to the containment upper cylindrical shell penetration. These lines are NPS 2 ASME Class 2. Due to their size, longitudinal breaks are not postulated. Circumferential breaks are postulated in accordance with BTP 3-4 Section B.A.(iii)(2). Breaks in Class 2 high energy piping systems are postulated at the following locations:
  a) terminal ends (defined in Section 3.6.2.1)
  b) at intermediate locations where stresses are calculated by the sum of equations (9) and (10) in NC-3653 of Section III of the ASME Boiler and Pressure Vessel Code to exceed 0.8 times the sum of the stress limits given in NC/ND-3653.

The RCCWS and CFDS lines are moderate-energy. Moderate-energy lines are subject only to through-wall leakage cracks and the resultant environmental consequences of localized flooding and increased temperature, pressure, and humidity (Section 3.6.1.2). The environmental effects of postulated moderate-energy leakage cracks are bounded by the accident conditions inside the CNV. As a result, leakage cracks are not evaluated further for the RCCWS and CFDS lines inside containment.

Final stress analysis is performed concurrent with fabrication of the first NPM. The postulated break locations based upon the current analysis are listed in Table 3.6-2.

ITAAC A07, Pipe Break Hazards Protective Features Verification, was established to confirm that the final pipe rupture hazards analysis demonstrates the acceptability of the dynamic and environmental effects associated with postulated ruptures in high-energy and moderate-energy piping systems within the NPM.

3.6.2.1.2 Pipe Breaks Outside the Containment Vessel (under the bioshield)

The CIVs for the RCS injection, RCS discharge, PZR spray supply, and RPV high-point degasification lines are each dual, independent valves in a single body that is welded directly to an Alloy 690 safe-end that is welded to the respective nozzle on the CNV head. These lines, except for the normally isolated RPV high-point degasification line, also have a check (injection and spray) or excess flow check (discharge) valve welded directly to the CIV. The feedwater system CIV is similar, except there is a single isolation valve (in accordance with GDC 57) with a check valve as the outboard valve in the single piece body.

The MSS lines each have a single CIV. Between the CNV safe end and the valve body are two tee fittings to which the DHRS steam lines attach.

Outboard of the valves in each of these lines is a short piping segment welded to a flange used to connect the refueling pipe spools to the module.

The containment isolation valves are outside the containment. The containment penetration area is defined by regulatory guidance as the run of piping from the inside CIV to the outside CIV. This definition is not directly applicable to NuScale. Instead, NuScale has omitted piping inside the CNV, but includes the above described valves. In other words, the NuScale containment penetration area is limited to the section from the CNV safe-end-to-valve (or tee) weld out to and including the piping weld to the outermost of the CIV or check/excess flow check valve.

For welds in the containment penetration area, provisions of BTP 3-4 Section B.A.(ii) have been applied to preclude the need for breaks to be postulated, because they meet the design criteria of the Section III of the ASME Boiler and Pressure Vessel Code, Subarticle NE-1120 and the following seven criteria:

1) The ASME Class 1 piping (i.e., the four CVCS lines) is designed to satisfy the following stress and fatigue limits:

    a) The maximum stress range between any two load sets (including the zero load set) calculated by equation (10) in Section III of the ASME Boiler and Pressure Vessel Code, NB-3653 does not exceed 2.4 S_m.

       Or, if the calculated maximum stress range of equation (10) exceeds 2.4 S_m, the stress ranges calculated by both equation (12) and equation (13) in Section III of the ASME Boiler and Pressure Vessel Code, NB-3653 meet the limit of 2.4 S_m.

b) The cumulative usage factor is less than 0.1 unless environmentally assisted fatigue is considered in which case the cumulative usage factor is less than 0.4.

c) The maximum stress, as calculated by equation (9) in Section III of the ASME Boiler and Pressure Vessel Code, NB-3652 under the loadings resulting from a postulated piping rupture beyond these portions of piping, does not exceed 2.25 S_m and 1.8 S_y.

The ASME Class 2 main steam and feedwater piping from the safe end to the weld outboard of the body holding the CIV and check valve is designed to satisfy the following stress limits:

a) The maximum stress ranges as calculated by the sum of equations (9) and (10) in Paragraph NC-3653, Section III of the ASME Boiler and Pressure Vessel Code, do not exceed 0.8(1.8 S_h + S_A).

b) The maximum stress, as calculated by Section III of the ASME Boiler and Pressure Vessel Code, paragraph NC-3653 equation (9) under the loadings resulting from a postulated piping rupture of fluid system piping beyond these portions of piping, does not exceed 2.25 S_h and 1.8 S_y.

2) There are no welded attachments for pipe supports.

3) There is a minimum number of circumferential and no longitudinal welds in these lines in the containment penetration area.

4) The length of the piping is the minimum practical (the total containment penetration piping length for 12 NPMs is less than a typical large pressurized water reactor).

5) There are no pipe anchors or restraints.

6) Guard pipes are not used.

7) The piping welds are included in the ISI program as described in Section 6.6, and the NPS 2 welds including and inboard of those of the pipe to outer nozzle welds of the check and excess flow check valves and CIVs are 100 percent volumetrically inspected, in addition to surface inspections as required by the ASME Boiler and Pressure Vessel Code Section XI.

Outboard of the containment isolation valves and check/excess flow check valves, the CVCS NPS 2, Schedule 160, RCS discharge, RCS injection, PZR spray supply, and high point degasification lines are ASME Class 3 lines to the first spool piece used to disconnect the NPM from the permanent piping. The spool piece and subsequent piping are also ASME Class 3 to the junction of an additional valve (or check valve) in each line, and subsequently become ASME B31.1 after that last valve. At the first spool piece breakaway flange, the four lines become part of the CVCS. Remaining piping under the bioshield, including the refueling pipe spools, is designed to comply with BTP 3-4 Rev. 2 Paragraph B.A.(iii) to preclude breaks at intermediate locations by limiting stresses calculated by the sum of equations (9) and (10) in NC/ND-3653 of Section III of the ASME Boiler and Pressure Vessel Code to not exceed 0.8 times the sum of the stress limits given in NC/ ND-3653.

Final stress analysis is performed concurrent with fabrication of the first NPM. Based on designing to meet the criteria from BTP 3-4, no breaks in the NPM bay outside the CNV (under the bioshield) are postulated. However, nonmechanistic breaks in MSS and FWS lines in the containment penetration area and leakage cracks are considered.

Decay Heat Removal System Lines

The DHRS is a closed loop system outside of the CNV that is entirely associated with a single NPM. Each NPM has two independent DHRS trains. Each train is associated with an independent steam generator (SG). The only active components in the DHRS are the DHRS actuation valves. The DHRS also relies on the MSS and FWS containment isolation valves to provide a closed loop system when it is activated. The DHRS is used to respond to transients including HELB outside containment. It is not used for normal shutdown, though the DHRS actuation valves are opened to allow slight circulation during wet layup of the SG. There is no flow through the DHRS system during normal operation. The DHRS is attached to the MSS line between the CNV and the MSS CIV. This portion of DHRS has two parallel actuation valves that are normally closed. These two lines join into a single line that supplies the passive condenser. Each DHRS condenser is attached to the outside of the CNV. The condenser is designed as an ASME Class 2 component. A NPS 2 line exits the bottom of each DHRS condenser and penetrates the CNV. This line connects to the feedwater system inside containment. During operation, the DHRS is pressurized from the feedwater line. See Section 5.4.3 for additional discussion about the DHRS.

Breaks are not postulated in the DHRS piping outside containment in accordance with in BTP 3-4, B.A.(ii). Subject to certain design provisions, NRC guidance allows breaks associated with high-energy fluid systems piping in containment penetration areas to be excluded from the design basis. Though the DHRS piping extends beyond what would traditionally be considered a containment penetration area, this approach is chosen because the DHRS cannot be isolated from the CNV as there are no isolation valves.

Breaks are not postulated in this segment of piping because it meets the design criteria for break exclusion in a containment penetration area (see Section 3.6.2.1.2). Although the DHRS condenser is manufactured from piping products, it is considered a major component and not a piping system; thus breaks are not postulated.

COL Item 3.6-2: A COL applicant that references the NuScale Power Plant design certification will verify that the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the containment vessel (under the bioshield) is applicable. If changes are required, the COL applicant will update the pipe rupture hazards analysis, design additional protection features as necessary, and update Table 3.6-2.

3.6.2.1.2.1 Non-mechanistic Secondary Line Breaks in Containment Penetration Area

BTP 3-3 B.1 (a)(1) specifies:

"Even though portions of the main steam and feedwater lines meet the break exclusion requirements of item 2.A(ii) of BTP 3-4, they should be separated from essential equipment. Designers are cautioned to avoid concentrating essential equipment in the break exclusion zone. Essential equipment must be protected from the environmental effects of an assumed non-mechanistic longitudinal break of the main steam and feedwater lines. Each assumed non-mechanistic longitudinal break should have a cross sectional area of at least one square foot and should be postulated to occur at a location that has the greatest effect on essential equipment."

For the NuScale design, the following considerations apply:

- MSS and FWS piping is the largest, high energy piping near the containment boundary
- The lines have a single CIV outside containment in accordance with GDC 57 for lines closed inside containment
- MSS and FWS piping is usually made of less corrosion resistant material than used for the NuScale design. MSS and FWS piping in many pressurized water reactors is carbon or low alloy steel, which has greater susceptibility to degradation than stainless steel.

Analyzing for non-mechanistic ruptures provides assurance that multiple essential SSCs are capable of withstanding the effects of a limited piping failure should one occur. In the NuScale plant, the dual CIVs are located outside the containment and exposed to the same environmental conditions, which makes protection against unexpected ruptures particularly important. However, the NuScale design has the following characteristics that make non-mechanistic ruptures low risk:

- The essential SSCs in vicinity of MSS and FWS piping in the containment penetration area are CIVs, DHRS valves, and instrumentation cables and sensors.
- Unlike some safety-related valves in other plant designs that use motor-operators, the NuScale CIVs are hydraulically held open against pneumatic pressure from an accumulator and shut upon a loss of power or a failure of the hydraulic line. The DHRS actuation valves similarly fail open.
- Failure of MSS and FWS piping is unlikely because:
  - Piping in the containment penetration area is made of stainless steel.
  - The physical length of MSS and FWS piping in the containment penetration area is zero (i.e., there are only valves and fittings).
  - MSS and FWS piping has a design pressure and temperature of 2100 psia and 625 degrees F, respectively, equivalent to the RCS piping.

The flow area of 1 ft^2 specified in BTP 3-3 for a non-mechanistic, longitudinal break is disproportionately large for a small modular reactor with small pipe sizes. NuScale MSS piping is NPS 12 Schedule 120 and FWS piping is NPS 4 and NPS 5 Schedule 120 in the containment penetration area. For those piping sizes, a 1 ft^2 flow area exceeds the area for a full circumferential rupture, which is physically unrealistic.

For the NuScale design, non-mechanistic breaks of MSS and FWS piping in the containment penetration area are evaluated, after consideration of the design differences from larger LWR plants. Comparing the typical PWR pipe MSS flow area to that of NuScale (NPS 30 to 38 vs NPS 12) yields a ratio of one-eighth to one twelfth. On this basis, NuScale analyzes for environmental effects of an MSS non-mechanistic break with an area of 12 in^2, versus 1 ft^2 (144 in^2). The non-mechanistic FWS break size applied for the NuScale design (NPS 4 and NPS 5) is 5.87 in^2.

The volume under the bioshield is small; roughly a cube 20 feet on a side. Therefore, even though only leakage cracks are required to be considered outside the containment penetration area, analysis is performed for a 12 in^2 MSS break at the highest point of the pipe run, resulting in a conservative pressure and temperature profile over time for environmental qualification and bounding breaks occurring in any section of the piping under the bioshield.

3.6.2.1.2.2 Break Exclusion

BTP 3-4 B.A.(iii) identifies specific criteria for which ruptures need not be considered from the containment wall to and including the inboard or outboard isolation valves (usually referred to as the containment penetration area "break exclusion zone"). The concept was necessary due to constraints on ability to cope with breaks between the CIVs. Should a break occur between the CIVs followed by a single failure of a CIV, then containment bypass could occur. To preclude bypass, criteria were developed to ensure that the probability of a piping failure was sufficiently low to make it implausible.

The NuScale plant has both CIVs in a single valve body. There are no break locations between the valves. However, the weld between the valve body and the CNV safe end is equivalent to those to which break exclusion applies. Therefore, NuScale has extended this boundary outside the CNV to include:

- The outboard weld at the CIV
- The outboard check or excess flow check valve nozzle weld in pressurizer spray, injection, and discharge lines
- DHRS piping welds outside the CNV

Accordingly, the guidance of BTP 3-4 B.A.(ii) is used in piping design to ensure that breaks and leakage cracks can be excluded in the containment penetration area. BTP 3-3 non-mechanistic breaks of MSS and FWS piping are also addressed. The remaining high energy piping under the bioshield applies

BTP 3-4 B.A.(iii) for ruptures and (v) for leakage cracks. Figure 3.6-33 is a representation (not all lines shown) of application of the BTP 3-4 guidance on break location and size, as applied in the NPM bay and the RXB.

The length of piping and number of welds inside the NuScale CNV is limited. For the NuScale design, no primary or secondary piping other than about 160 feet of DHRS piping is within the break exclusion zone outside containment. The design pressure and temperature of MSS, FWS, and DHRS piping in the break exclusion zone is the same as for the RCS.

Break exclusion is not applied to any of the piping in the RXB outside of the bioshield.

3.6.2.1.2.3 Leakage Cracks

Leakage cracks are excluded in containment penetration areas where the criteria of BTP 3-4 B.A.(ii) are satisfied.

For areas outside the containment penetration area, per BTP 3-4 Paragraph B.A.(v), leakage cracks are postulated unless specific criteria are met. For Class 2 piping, the acceptance criterion is for the calculated stress to not exceed 0.4 times the sum of stress limits given in Subarticles NC/ND-3635. BTP 3.4 B.C.(iii) specifies postulating leakage cracks with a flow area of one-half of a pipe diameter by one half pipe wall thickness in piping in the vicinity of essential SSCs, regardless of system.

3.6.2.1.3 Pipe Breaks in the Reactor Building (outside the Bioshield)

Within the NPM, there are a number of essential SSC that require protection and relatively small amounts of piping. Therefore, postulated pipe break locations within the NPM or in close proximity to the NPM (i.e., under the bioshield) are specifically addressed by analysis, as discussed in Section 3.6.1.3.

Beyond the NPM, there are fewer SSC that require protection and a large amount of high- and moderate-energy piping (See Table 3.6-1). The SSC that require protection are evaluated for effects of line breaks or are separated within compartments of the RXB from areas that contain piping. In addition, the building structure necessary to support the modules and to maintain the integrity of the pool (i.e., the ultimate heat sink) is evaluated.

Piping arrangements in the RXB have not been finalized yet. It is appropriate, therefore, for evaluation of potential rupture locations beyond the reactor pool bay wall, to identify the bounding dynamic effects of postulated breaks and then to determine if protection is required. The approach is to evaluate:

- blast, unconstrained pipe whip, and jet impingement caused by rupture of a main steam pipe.
- subcompartment pressurization, spray wetting, flooding, and other adverse environmental effects caused by main steam or CVCS breaks that are potentially limiting where they might occur in the building.

- multi-module impacts in common pipe galleries.

A break in a high-energy MSS or FWS line in the RXB (outside of the bioshield) could potentially cause breaks or leakage cracks in smaller diameter or pipe schedule lines of other NPMs, introducing an additional transient in a second NPM. Therefore, RXB MSS and FWS pipes must be arranged, and/or pipe whip restraints must be provided to prevent a collateral rupture, or pipe whip impact analysis must be performed to show that a collateral rupture does not occur. However, the effects of an MSS or FWS break are assumed to cause an MSS bypass line rupture in an adjacent module in order to determine bounding dynamic effects and to ensure that the RXB structure is adequate for beyond design basis interactions between adjoining modules. Once piping arrangements are finalized, the need for pipe whip restraints and barriers may be determined to avoid multi-module effects. This is addressed by the COL applicant as part of COL Item 3.6-3.

The CVCS lines in the RXB (outside the bioshield) are not co-located with essential SSC, with the exception of the RXB itself. Therefore, dynamic effects are addressed on a bounding basis and individual break locations are not specified. For flooding and environmental effects, as discussed in Sections 3.4 and 3.11 respectively, breaks are postulated to occur anywhere on the line.

COL Item 3.6-3: A COL applicant that references the NuScale Power Plant design certification will perform the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the reactor pool bay in the Reactor Building (RXB), and update Table 3.6-2 as appropriate. This includes an evaluation and disposition of multi-module impacts in common pipe galleries, and evaluations regarding subcompartment pressurization. The COL applicant will show that the analysis of RXB piping bounds the possible effects of ruptures for the routings of lines outside of the RXB or perform the pipe rupture hazards analysis of the high- and moderate-energy lines outside the buildings.

3.6.2.1.4 Pipe Breaks in the Control Building

There are no high-energy lines in the CRB. Flooding and environmental evaluations are described in Section 3.4 and 3.11, respectively.

3.6.2.1.5 Pipe Breaks in the Radioactive Waste Building

There are no high-energy lines or essential equipment in the RWB. Therefore, no breaks or leakage cracks are postulated.

3.6.2.1.6 Pipe Breaks Onsite (Outside the Buildings)

As discussed in Section 3.6.1.1.6, there are four high-energy lines outside of the RXB and CRB: MSS, FWS, auxiliary boiler, and extraction steam, and multiple moderate-energy lines (See Table 3.6-1). However, there is no essential equipment outside of the RXB or CRB. The routing of piping outside of the RXB, CRB, and RWB is the scope of the COL applicant.

COL Item 3.6-4: Not used.

3.6.2.1.7 Types of Breaks

The criteria used to determine the axial locations of postulated pipe breaks are described in Section 3.6.2.1.1, Section 3.6.2.1.2, and Section 3.6.2.1.3. At these locations, either a circumferential or longitudinal break, or both, are postulated according to the following criteria:

- For piping sizes larger than NPS 1, at piping terminal ends, a circumferential break only is postulated.
- For piping sizes larger than NPS 1 but smaller than NPS 4, at intermediate locations (i.e., not terminal ends), a circumferential break only is postulated.
- For piping sizes NPS 4 and larger, at intermediate locations (i.e., not terminal ends), both a circumferential and longitudinal break are postulated unless the location of the break is selected using stress analysis per the criteria given in Section 3.6.2.1.1, Section 3.6.2.1.2, and Section 3.6.2.1.3 and a further evaluation of the stress results is used to determine the break type as follows:
  - If the circumferential stress range is at least 1.5 times the axial stress range, only a longitudinal break need be postulated
  - If the axial stress range is at least 1.5 times the circumferential stress range, only a circumferential break need be postulated

Where circumferential breaks are postulated, the following assumptions are made:

- A circumferential break results in pipe severance and separation amounting to at least a one-diameter, lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (i.e., a plastic hinge not developed in the piping).
- Pipe movement is initiated in the direction of the jet reaction and whipping occurs in a plane defined by the piping geometry and configuration.

Where longitudinal breaks are postulated, the following assumptions are made:

- A longitudinal break results in an axial split without pipe severance. Splits are postulated to be oriented (but not concurrently) at two diametrically opposed circumferential locations such that the jet reactions cause out-of-plane bending of the piping configuration. Alternatively, a single split is assumed at the location of highest tensile stress as calculated by detailed stress analysis (e.g., finite element analysis).
- Pipe movement occurs in the direction of the jet reaction unless limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis.

Longitudinal cracks are not applicable in the CNV, because piping NPS 4 and larger meets LBB criteria. Also, longitudinal breaks are not considered under the bioshield, based on meeting criteria for not considering circumferential breaks. In the rest of the RXB, effects of longitudinal breaks (with break flow areas equal to the piping flow area) are bounded by circumferential breaks.

3.6.2.1.8 High- and Moderate-Energy Leakage Cracks

For high-energy lines, with the exception of those portions of piping exempted using the criteria contained in BTP 3-4 B.A(ii) as described in Section 3.6.2.1.2 and Section 3.6.2.7, leakage cracks are postulated at locations that result in the most severe environmental consequences unless otherwise selected by stress analysis. For lines where stress analysis has been performed, postulated leakage crack locations are determined according to the criteria in BTP 3-4 B.A(v) as follows:

- For ASME Code, Section III, Class 1 piping at axial locations where the calculated stress range by Eq. (10) in NB-3653 exceeds 1.2Sm.
- For ASME Code, Section III, Class 2 and 3 piping, or nonsafety-class (i.e., non-ASME Class 1, 2, or 3), at axial locations where the calculated stress equal to the sum of Eq. (9) and Eq. (10) in NC/ND-3653 exceeds 0.4 times the sum of the stress limits given in NC/ND-3653.

For moderate-energy lines, leakage cracks are not postulated inside the containment or outside the containment under the bioshield. Per BTP 3-4 Part B.B(iv), leakage cracks need not be postulated in moderate-energy piping located in an area in which a break in high-energy piping is postulated, provided such leakage cracks would not result in more limiting environmental conditions than the high-energy piping break. For the areas inside containment (described in Section 3.6.1.1.1) and outside containment under the bioshield (described in Section 3.6.1.1.2), the effects of leakage cracks in the moderate-energy RCCWS, CFDS, and CES, are bounded by breaks in high-energy lines. In other areas of the plant, ruptures of moderate-energy lines are assumed at locations that result in the most severe environmental consequences. Environmental conditions are based upon the leakage cracks of the worst case (typically largest or hottest) line in the proximity of safety-related SSC. For flooding analysis, full circumferential breaks in piping larger than NPS 2 in a room where they are located are used to evaluate flooding. Environmental effects are discussed in Section 3.11 and flooding analysis is described in Section 3.4.

Per BTP 3-4 C(iii)(1) leakage cracks in high- and moderate-energy lines need not be postulated in NPS 1 and smaller piping. Where leakage cracks are postulated in high- and moderate-energy lines, the following criteria from BTP 3-4 C(iii) are applied or are shown to be bounded:

- For high-energy piping, the leakage cracks should be postulated to be in the circumferential locations that result in the most severe environmental consequences. For moderate-energy piping, leakage cracks should be postulated at axial and circumferential locations that result in the most severe environmental consequences (per BTP 3-4 B(iii)(2)).
- Fluid flow from a leakage crack should be based on a circular opening of area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width. The flow from a leakage crack should be assumed to result in an environment that wets the unprotected components within the compartment with consequent flooding in the compartment and communicating compartments. Flooding effects should be determined on the

basis of a conservatively estimated time period necessary to effect corrective actions.

3.6.2.2 Effects of High- Energy Line Breaks

In accordance with SRP Section 3.6.2, the dynamic effects of postulated high-energy line break are evaluated using the methodology as described in this section.

3.6.2.2.1 Blast Effects

The potential for a blast wave to occur depends on the surrounding environment. Key factors include the timing of the break and the initial system thermodynamic conditions. The timing of opening of the break and the initial, intact system thermodynamic conditions also are key factors. Although pipe rupture times of less than a millisecond are unlikely, break opening time is assumed to be instantaneous, maximizing blast formation. The formation and effects of a blast wave caused by an HELB is evaluated using three-dimensional computational fluid dynamics (CFD) modeling that reflects the postulated break characteristics and NuScale plant geometry. The analysis is performed using ANSYS CFX.

The acceptability of using CFX for this purpose was demonstrated by performing verification and validation using eight test problems that exercised different capabilities of the code.

Key observations from this blast wave modeling are:

- A blast wave is weakly formed if the surrounding environment is at low pressure (less than 1 psia), as is the case inside the CNV. Buildup of pressure as blowdown progresses is not relevant, because the blast wave is a prompt and short-lived phenomenon.
- The severity of a blast depends on the amount of fluid that can escape within approximately one millisecond of break onset because the blast wave forms within that time.
- The NuScale high-energy, steam-filled lines are relatively small, which limits the severity of the blast pressure. The energy available to form the blast is less than one-twenty-fifth that for a typical large, light-water reactor.
- Blast waves are not significant for subcooled discharge, because liquid flashing occurs on time scales exceeding that of blast wave formation (Reference 3.6-22).
- A blast wave has well-defined and interrelated characteristics. For example, its peak pressure and speed decrease with distance from its origin.
- The pressure load applied by a blast wave is of short duration (i.e., an impulse load) and does not apply uniformly across large SSC at a given instant. Therefore, assuming the peak blast pressure is applied across the entire projected area of a component is inappropriate. The CFD analysis explicitly accounts for the time-varying pressure of the rapidly propagating blast wave.

- Reflection off surfaces can reinforce the pressure load, requiring consideration of plant-specific geometry. Angled or curved surfaces are loaded differently than a flat surface perpendicular to a line between the blast origin and surface. The pressures applied to surfaces by reflection can exceed the incoming wave pressure. For this reason, use of representative plant geometry is necessary. The CFD analysis includes the interaction of incident and reflected waves with each other and nearby surfaces, including how the shape and orientation of surfaces affect reflection.
- A small target has a lower peak pressure due to "clearing," which is a phenomenon where some of the blast overpressure is relieved by bleeding off around the edge of the target. Because of both pressure-relieving clearing and the short load duration as a supersonic blast wave moves over them, small structures are not exposed to significant loading. The only SSC in the CNV or RXB that are large are the structures (e.g., CNV, RPV, and RXB walls and floors). The CFD analysis considers clearing.
- Several locations and directions of CVCS breaks in the CNV and MSS breaks in RXB pipe gallery were modeled. These were selected to maximize blast pressure on nearby SSC (e.g., close to walls and corners) in order to bound final piping arrangements.

Blast analyses results show a maximum total force of 6000 lbf on a component in the CNV. The maximum total force is less than 10,000 lbf on a component and about 100,000 lbf on the five-foot-thick pool wall in the RXB pipe gallery (the wall load is spread over an area with a radius of about 100 inches, corresponding to an average pressure of less than 14 psig, compared to a concrete compressive strength of 5000 psia). These forces are impulse loads that last only a few milliseconds or less.

In summary, three-dimensional CFD analysis of blast wave formation in the CNV and RXB is performed using modeling assumptions that bound the pressurization effects that occur for HELB in the plant. Blast wave force time histories are calculated for nearby SSC. The results show:

- Peak forces are low and bounded by the jet thrust forces that subsequently develop. The values are low because NuScale HELB are relatively small diameter and deposit only a small amount of mass and energy in the time it takes for a blast wave to form. The forces inside the CNV are low because the initial low ambient pressure does not support formation of a significant blast wave.
- The forces of the passing shock wave are of very short duration.

Therefore, effects of HELB-induced blast waves in the NuScale plant are considered negligible. No damage to surrounding SSC occurs because these loads are small and brief.

3.6.2.2.2 Pipe Whip

The methodology for pipe whip includes determination of whether a pipe has sufficient energy to whip, whether a whipping pipe can actually contact a safety-significant target, whether the target is sufficiently robust to withstand the impact (qualitatively or by dynamic analysis), and evaluation of the consequences of an impact should the previous steps not obviate the possibility of damage.

The thrust force caused by release of fluid from a circumferential break of a high-energy piping system may cause the piping to rotate about a plastic hinge-point (e.g., pipe restraint, pipe anchor point) and possibly impact nearby SSC.

Inside the CNV, the largest pipe size subject to HELB conditions is NPS 2 and target SSC are robust [e.g., reactor vent valves (RVVs)]. High-energy piping systems larger than NPS 2 have been qualified for LBB inside the CNV. Outside the CNV, under the bioshield, piping satisfies the criteria of BTP 3-4 B.A.(ii) or (iii) to conclude that no breaks occur and that piping does not need to be evaluated for whip. However, nonmechanistic breaks of MSS and FWS lines and leakage cracks are considered. In the RXB outside the bioshield, MSS, FWS, and CVCS lines are subject to a postulated HELB, but there are only a limited number of SSC requiring protection. Also, Auxiliary Boiler System (ABS) line leakage cracks are evaluated.

The following considerations apply to evaluation of pipe whip:

- For piping meeting the criteria of break exclusion or LBB, pipe whip is not considered because dynamic effects of ruptures are excluded.
- If the end is an RPV or CNV safe end, whip does not occur because the safe end and its nozzle is short, stiff, straight, and restrained by the component.
- In accordance with SRP Section 3.6.2, a pipe struck by another pipe of equal or smaller diameter and schedule (i.e., wall thickness) does not break or crack. In the CNV where HELB are limited to NPS 2 Schedule 160 pipe, the RPV, CNV, ECCS valve bodies, and CRDMs are each equivalent to larger, thicker-walled pipe and, therefore, do not crack or break.
- Where pipe ruptures are postulated to occur, the distance is determined from the break location to the nearest restraint that limits the range and/or direction of the pipe whip.
- The jet thrust necessary to cause pipe whip is determined. The calculation of thrust and jet impingement forces considers no line restrictions (e.g., a flow limiter) between the pressure source and break location, but does consider the absence of energy reservoirs, as applicable (e.g., the high-point vent pipe in the CNV is normally isolated).
- If the jet thrust is insufficient to yield the pipe, then pipe whip at that break location is eliminated from further consideration.
- Pipe whip is considered to result in unrestrained motion of the pipe along a path governed by the hinge mechanism and the direction of the vector thrust of the break force. A maximum rotation of 90-degrees is assumed about a hinge. Pipe whip occurs in the plane defined by the piping geometry and configuration and initiates pipe movement in the direction of the jet reaction, as identified in BTP 3-3.

The RPV, CNV, CRDMs, and ECCS main valve bodies are robust structures with equivalent wall thicknesses in excess of the NPS 2 Schedule 160 pipe that may whip inside the CNV. The minimum wall thickness of these components is at least three times that of the postulated whipping pipe.

In view of the SRP 3.6.2 provision for impact of a pipe on like-size or larger pipe, the RPV, CNV, CRDMs, and ECCS valve bodies neither rupture nor crack if struck by a whipping NPS 2 Schedule 160 pipe in the CNV. Because of the disparity in the thickness of the walls, the whipping pipe kinetic energy is absorbed in the bending and crushing of the pipe itself. Functionality of components with moving parts (i.e., CRDMs and ECCS valves) following impact is addressed.

Postulated break locations are at the RPV (head for spray and high-point vent degasification lines, and side wall for injection and discharge lines) and CNV heads. The high-point vent line does not whip for a break at the RPV head, because the isolated line is filled with steam that immediately depressurizes as the break begins to open.

Ruptures above the NPM under the bioshield are excluded and there are no safety-related or essential components with whip range elsewhere in the RXB. However, the RXB structural integrity and, in particular, the integrity of the pool wall must be assessed, so pipe whip impact force on concrete surfaces is determined. After break-opening, the steady-state jet thrust force, F_b, is:

$$F_b = C_T P_o A_e \qquad \text{Eq. 3.6-1}$$

where,

F_b = Steady state thrust force at the break (lbf),

C_T = Thrust coefficient (unitless),

P_o = Internal system pressure (psia), and

A_e = Pipe flow area (in^2).

- Applying the jet thrust force at the distance to the nearest restraint or anchor point determines the force available to overcome the pipe plastic bending moment and accelerate the pipe. If there is sufficient energy for whip to occur, the energy to yield the pipe is not deducted from its kinetic energy. Pipe whip evaluations have determined that impact on targets in the CNV is unlikely. Quantitative pipe whip impact evaluation is performed only for concrete walls and other structures in the RXB.
- To determine the depth of penetration of the whipping pipe into an RXB wall, the Sandia formula developed by Young (Reference 3.6-20) is used. For an assumed pipe whip segment length and angle of travel that is bounding, whip of an MSS line (the highest-energy pipe in the RXB) causes a penetration of a depth of 4.1 inches, or about 7 percent of the minimum wall thickness of

importance. Additionally, the walls, ceilings, and floors are sufficiently thick that this impact does not cause spalling on the far side surface. Given their smaller pipe size, chemical and volume control system pipe impacts are even less damaging.

3.6.2.2.3 Jet Impingement

Target SSC in the path of jets issuing from postulated breaks are assessed for the load imparted by the jet. In industry testing, single-phase steam jets with upstream pressures of 1200 psia were found to cause damage to pipe insulation at a distance of up to 25 times the pipe exit diameter (i.e., L/D = 25). However, insulation is fragile as evident from Reference 3.6-16, which reports types of insulation suffering damage due to impingement pressures as low as 4 psig.

NUREG/CR-6808 (Reference 3.6-17) Table 3-1 provides the impingement pressures found in testing to cause damage to various types of piping insulation used in U.S. pressurized water reactors. The damage pressures range from 4 to 40 psi for fibrous insulation to a high of 190 psi for two types of reflective metal insulation. Insulation is more fragile than the uninsulated solid metal surfaces of SSC inside the CNV. Therefore, jet impingement pressures need to be considerably above 190 psi to be of concern. Impingement loads are only relevant for hard or relatively hard targets such as ECCS valve bodies, the CNV steel wall, and the RXB concrete structure. Impingement pressures must be substantial (above 190 psia) rather than the less than 4 psia needed to protect against dislodging insulation. As such, fewer uncertainties exist in predicting jet impingement effects on piping, and the relevant penetration distance is much shorter than 25 L/D.

Jet impingement testing was performed on electrical cable in support of the AP1000 assessment of debris generation. The conclusion was that cables at greater than or equal to 4 L/D from a jet simulating an AP1000 LOCA were not damaged. The results were given in terms of distance because of difficulty in accurately measuring impingement pressure. The NRC staff agreed with the conclusion. In Reference 3.6-18, the Advisory Committee on Reactor Safeguards (ACRS) also agreed, stating,

> "The recommended distance of four break diameters from a loss-of-coolant accident jet, at which unprotected cables would not be damaged, has been shown by testing to be sufficiently conservative to bound plant conditions with high likelihood."

Although the focus of this testing did not include cable functionality, inspection of test target cables showed no damage at greater than or equal to 4 L/D (with exception of one cable). The results were applicable only to the type of cables actually tested, but an AP1000 LOCA jet is considerably larger and higher energy than a NuScale NPS 2 HELB. Therefore, it is likely that even unprotected cable inside the CNV would survive jet impingement from an NPS 2 HELB provided its separation from the break exit exceeded 4 L/D, or 6.75 inches. NuScale cable to be used in the CNV is tested for survival under jet impingement.

For effects on concrete, MSS breaks are limiting and are assumed to occur within 2 L/D of a wall, with no reduction in jet pressure with distance from the break. The maximum force of the jet and its maximum pressure is that at the break exit, or 103,000 lbf and 630 psia, which is well within the minimum 5000 psi compressive strength of the concrete making up the five-foot thick wall. In addition, the effect of erosion is negligible.

An overview of the NuScale resistance to jet impingement is:

- The damage potential of the smaller-scale NuScale piping is reduced compared to large reactors:
  - Based on plant operating conditions and size of piping, thrust loads for NuScale line breaks are a fraction of those encountered in large LWRs (e.g., a NuScale 12-inch MSS line has about five percent of the total thrust force of a 38-inch MSS line break).
  - Main steam system HELB occurrence is limited to the RXB, because MSS breaks inside the CNV and under the bioshield are eliminated by LBB and break exclusion, respectively. Considering MSS steam density, flow rate driven by the system to ambient differential pressure, and the full break single-ended flow area, the NuScale MSS HELB mass and energy transfer is approximately five percent of that in other large LWRs.
- Jet reaction load and, if within the ZOI, potential jet impingement load is included in load combinations in accordance with FSAR Section 3.9 and Section 3.12.
- Damage to insulation on piping is not a concern:
  - In the CNV, no pipe or component thermal insulation is used.
  - Under the bioshield, no ruptures are postulated.
  - In the NPM outside the pool area, dislodged insulation has no effect on long-term NPM cooling.

Thus, allowable impingement pressure on SSC is considerably higher than that in large pressurized water reactors where insulation stripping is relevant.

- The maximum load imposed by the impinging jet is that of the thrust force of the broken pipe at the break exit.
  - Because only NPS 2 RCS pipes are locations of postulated breaks in the CNV, the load is limited to the maximum operating pressure times the flow area times the thrust coefficient (1.26 for steam and two-phase jets). The total load imposed by the jet is approximately 5220 lbf.
  - The applied load is adjusted by a target shape factor (e.g., 0.576 for a jet striking a cylinder normal to its axis) and by the cosine of the angle from perpendicular for the intersection of the jet and the target surface. These two adjustments reduce the imposed load to below 2000 lbf, or approximately two times the weight of a reactor recirculation valve.

  - Finally, the jet rapidly traverses the zone of influence (ZOI) caused by whip of the broken pipe, moving more than 100 ft/sec within a few degrees of motion. The RVVs are not directly in line with a location in which a whipping pipe could come to rest and are, therefore, exposed to the jet only transiently. The RVVs are approximately a foot in diameter, meaning that they are within the jet for a maximum of 0.01 of a second. Exposing a 1000 lbm, thick-walled, metal component to 2000 lbf for 0.01 of a second or less is a negligible load that can be omitted from load combinations that include dead weight and seismic accelerations of over 10 g.

- The impingement damage threshold of 190 psi is a sufficient measure of the structural integrity of components, but does not confirm functionality. Essential components inside the CNV are qualified for a CNV design condition of at least 1000 psia saturated steam. This exceeds the 190 psi impingement acceptance threshold of 190 psia by a factor of more than five and is sufficient basis to consider functionality after jet impingement to be demonstrated.

- Jet impingement on concrete is neither a pressure load nor an erosion concern.

Having addressed the resistance of the NuScale design to jet impingement damage, the HELB jet conditions must be determined. Three categories of jets are considered:

1) Liquid jets

2) Two-phase jets

3) Steam jets

As discussed for other effects, jet behavior and effects differ for the three areas of the plant:

- Inside the CNV: breaks are limited to NPS 2 RCS-connected and DHRS piping because the SGS piping meets LBB. Only a degasification line break is steam, however, the reverse flow from a pressurizer spray line break almost immediately turns to steam. Other breaks such as DHRS, the injection line, or spray line forward flow are two-phase.
- Under the bioshield: piping satisfies criteria that no postulated breaks occur.
- In the RXB, outside the bioshield: piping arrangements are not finalized, so break locations and jet directions are assumed to be throughout in the rooms containing high-energy piping. The piping is limited to NPS 12 and 4 MSS, NPS 6 FWS, and NPS 2 to 3 CVCS piping at various pressures and temperatures (see Table 3.6-1 and Table 3.6-4). Main steam system jets are steam only, whereas FWS and CVCS breaks are two-phase.

The concern for jet impingement that underlies regulatory guidance is the stripping of insulation with subsequent sump blockage as described in GSI-191. As noted above, the impingement damage threshold for NuScale is greater than 190 psig.

Liquid jets

Liquid jets are assumed to not expand (i.e., the cross section of the pipe rupture is maintained) and to not droop with distance (i.e., travel straight until impeded). Additionally, a 2.0 thrust coefficient is used for dynamic loading. The only areas subject to liquid jets are in the RXB where CVCS lower temperature, high-pressure piping is present. The essential SSCs in this area are the CVCS demineralized water makeup valves and RXB structure (liquid jets are considered to have less potential to damage concrete structure than steam jets, which are shown to be acceptable).

Two-phase jets

Two-phase jets are assessed using the methodology of NUREG/CR-2913 (Reference 3.6-19). A bounding approach is taken by identifying criteria for jet formation in order to avoid the need to analyze individual break locations in the CNV and RXB.

- In the CNV

  Although the low operating pressure of the CNV is a variation from the experimental and analytical basis of NUREG/CR-2913, the low ambient pressure results in faster expansion of the jet and is conservative when estimating loading.

  Only RCS-connected NPS 2 pipe breaks are evaluated (DHRS system pressure and temperature are lower at postulated break locations). The inputs needed for the NUREG/CR-2913 methodology are the system static thermodynamic conditions, as shown in Table 3.6-4.

  Following the methodology, the relevant graph of Appendix A of NUREG/CR-2913 is selected to obtain target pressure and total force on the target for appropriate values of P_0, ΔT_0, or X_0, and distance to the target in L/D. For the CVCS breaks in the CNV, the thermodynamic conditions are 48 degrees K subcooling and 67 bar. The appropriate graph is Figure A.39, which shows pressures at specific points downstream in L/D and radially from the jet centerline in r/D. At the origin of the plot is the jet centerline at the break exit plane, and the shaded area at the lower left is the jet core (the region that has not yet begun to interact with the environment and in which fluid striking a target would experience full recovery of the fluid stagnation pressure). The letters A through D refer to the key for pressure (letters E and beyond for pressures above 10 bar are not plotted because they exist only near the jet core). For example, a letter B indicates pressure is 2.5 bar at 4 L/D and 1.5 r/D.

  The jet core is the region immediately downstream of a break in which the target pressure is the full stagnation pressure. Reference 3.6-17, Section 3.3.1.1 states that this region is significant only for jets involving subcooled stagnation conditions. Figure A.39 of NUREG/CR-2913 shows that the jet core dissipates within 2 L/D or about 3.4 inches for a thermodynamic condition similar to a chemical volume and control system HELB. This is viewed as conservative. Reference 3.6-13 Section 3.5.3.B notes that Sandia (Reference 2.6.19) emphasizes the pipe exit core. The persistence of the core is attributed by Sandia to the time it takes for external pressure to penetrate the jet, and that the core length will always be longer than 0.5 D for subcooled and saturated water jets. Reference 3.6-13 notes, however, that test data is not consistent with the Sandia model, with only one or two test data sets exhibiting something like a liquid core while most data contradict the presence of a liquid core. Reference 3.6-13 concludes “If a liquid core exists, it seems to be much smaller than indicated by Sandia.”

At 2.5 L/D and 1 r/D, the single D point is a pressure of 10 bar (145 psig), below the NuScale damage threshold of 190 psig. Within 4 L/D or about 6.8 inches, the jet peak pressure has dropped to below 5.0 bar (72.5 psig). The A points representing 1.0 bar correspond to the edge of the jet. The jet persists beyond 7.5 L/D, which is indicative of the concern for fibrous insulation damage at pressures of 4 psig out to a 10 L/D penetration distance. For NuScale's design, pressures at about 2 L/D are low enough to cause no damage to the hard components.

Although the NUREG/CR-2913 figure can be used to determine the ZOI, the ZOI in the CNV is assumed to be in the forward-facing hemisphere because of the greater spreading angle in the low-pressure CNV and possible pipe whip.

- In the RXB

  Similarly, for chemical and volume control system HELB in the RXB, the generic approach of a universal ZOI allows for breaks at locations determined once pipe routing is finalized and for pipe whip. Based on the discussion that follows for steam jets, CVCS pressure loading, as shown in Figure A.39 of NUREG/CR-2913, is not damaging.

<u>Steam Jets</u>

- In the CNV

  For breaks inside the CNV, expansion of the jet into the low-pressure surroundings results in different behavior than is experienced for HELB. Wider jet spreading (a half-angle exceeding 60 degrees) is expected to occur, because the initially low air density of the CNV removes most of the resistance to jet expansion. The wider jet expands the ZOI, but substantially reduces the pressure and the penetration length, because the mass and energy of the jet are widely dispersed. Although pressure within the CNV increases with time, the pre-existing wide expansion of the jet persists because the jet is already established.

  For simplicity and because there are no rigid restraints at postulated break locations to constrain separation, circumferential breaks are assumed to be full separation. For circumferential breaks with full separation, it is assumed that an essential system or component is within the ZOI if it is located within the forward-facing hemisphere based on the original pipe orientation.

Applying the break exit pressure over this ZOI is an overestimation of the possible jet impingement loading. Therefore, the steam and two-phase jet pressure is assumed to decrease with distance proportional to the area of a jet that expands at a 30-degree half-angle to five pipe diameters and then at 10 degrees beyond that. A half-angle of 30 degrees is less than identified in the ANSI/ANS 58.2 Standard and in other jet analyses for expansion into surroundings at normal atmospheric pressure. Thus, the jet pressure is below the 190 psi threshold for component damage at 2.2 L/D (3.65 in.). Although the NRC has challenged the general applicability of the ANSI/ANS Standard 58.2 spreading model, a half-angle of approximately 45 degrees or more is usually used. As the jet spreads more rapidly into the low-density CNV atmosphere, a 30-degree assumption is sufficiently conservative to bound actual jet impingement pressures due to local variation within the jet.

As noted, the jet core is only significant for subcooled jets. Reference 3.6-19 Section 3.6 discusses the core length L_c as ½D, one half of the pipe diameter for saturated stagnation conditions. It also notes that the length L_c depends on the time it takes a pressure wave to travel from the outer edge of the nozzle (i.e., break) to the jet center. Figure 4.3 of Reference 3.6-19 shows that for zero degrees subcooling L_c=½D. Thus, even if a jet core existed for a steam jet, its influence would be dissipated within ½D, which is too close for a jet impingement force to be of concern compared to pipe whip impact.

- In the RXB

  Jet core length is not relevant for RXB breaks because full exit plane pressure is assumed. The distance between a break and a target SSC is not defined because RXB piping arrangements have not yet been finalized. To verify suitability of the design of the RXB, bounding HELB scenarios have been identified.

  The MSS lines are larger and contain more energy than other potential jet sources in the RXB. Demonstrating passing performance for MSS breaks provides confidence that final HELB analysis results are bounded. Therefore, a conservative approach is taken in which the jet impingement pressure is assumed to be the same as that at the break exit (i.e., no reduction for spreading with distance). For a main steam system HELB, the break exit pressure is 500 psia. Applying the thrust coefficient C_T of 1.26 yields a jet impingement pressure of 630 psi, or about one-eighth of the minimum compressive strength of the concrete and less than the previously discussed erosion that testing demonstrated is acceptable.

Jet impingement for HELB in the NuScale plant is therefore not a source of concern because of the lesser jet energies associated with the smaller size piping, and because of the high impingement pressure damage threshold associated with not needing to protect against insulation being dislodged.

3.6.2.2.4 Dynamic Amplification and Resonance of Impingement Jet

Based upon concerns raised by the ACRS in 2004, the NRC identified (SRP Section 3.6.2) that unsteadiness in free jets, especially supersonic jets, tends to propagate in the shear layer (i.e., the region with a large velocity gradient near the boundary of the jet) and induce time-varying oscillatory loads on obstacles in the flow path. The ACRS concern was that pressures and densities vary nonmonotonically with distance along the axis of a typical supersonic jet, feeding and interacting with shear layer unsteadiness. In addition, for a typical supersonic jet, interaction with obstructions could lead to backward-propagating transient shock and expansion waves that cause further unsteadiness in downstream shear layers.

The concern was that synchronization of the transient waves with the shear layer vortices emanating from the jet break could lead to amplification of the jet pressures and forces (a form of resonance) that is not considered in ANSI/ANS 58.2. Should the dynamic response of the neighboring structure also synchronize with the jet loading time scales, further amplification of the loading occurs, including at the source of the jet. General observations by investigators were that strong discrete frequency loads occur when the impingement surface is within 10 diameters of the jet opening, and that when resonance within the jet does occur, amplification of impingement loads might result.

The basis for this concern was research into such amplification of loads that occur in the interaction of the jet issuing from vertical and short take-off and landing aircraft and certain industrial gas jet applications. It causes vibration and fatigue damage to aircraft parts, jet deflectors, parts cleaned with gas jets, etc. This phenomenon has been studied extensively, with considerable work performed to mitigate its effects.

Experiments simulating HELB routinely evince random oscillations, but not resonance. For dynamic amplification and resonance to occur, a number of criteria must be met. These criteria are based on the research referenced in SRP Section 3.6.2 and similar work that identified the physical phenomena leading to resonance. These processes require a stable, axisymmetric jet impinging at a fixed distance perpendicular to a large, flat surface. The processes at work during a HELB have fundamental differences from those that occur in a jet with dry, noncondensable gas issuing from a smooth, fixed nozzle. These physical differences involve instability of the discharge, irregular discharge geometry, phase changes that suppress pressure changes, misalignment of jet and impingement target surface preventing establishment of a feedback loop, lack of an appropriately flat surface within a sufficiently close distance, and etc. If one of these criteria is not met, a resonance is implausible. In a HELB in the NuScale plant, none of the criteria is satisfied, precluding the formation of a resonance.

Specifically, each of the following characteristics of postulated HELB in a NuScale plant is sufficient to ensure a resonance does not occur.

- A whipping pipe either 1) comes to rest against an object that intercepts a portion of the jet and distorts its axisymmetry or 2) flutters, causing a variation in the jet impingement angle and separation that prevents establishing synchronization of the transient waves.

- The break exit is distorted because of tearing as the break opens, which eliminates axisymmetry.
- Jets in the CNV dissipate in a short distance. The plant geometry precludes the end of a break coming within 2 L/D of a suitable impingement surface. Beyond that distance, the jet has weakened too much for amplification to be a problem, even if it does occur.
- No suitable (i.e., even, flat) impingement surfaces exist within the CNV. Relevant SSC are curved, which redirects reflected acoustic energy away from the break exit.
- The presence of a steam/water mixture in the jet acts to dampen pressure oscillations, preventing amplification.
- Splashing from the jet (and the jet from the opposite end of the break) interferes with the stability of the jet.

3.6.2.2.5 Subcompartment Pressurization

In the CNV, pressurization from postulated HELB is bounded by ECCS initiation and no breaks for which dynamic effects must be considered are postulated under the bioshield.

For the RXB, bounding HELB scenarios have been identified based on the high-energy systems in the subject areas of the building. The largest mass and energy input considered is in the pipe gallery and involves a MSS HELB with pipe whip that causes a MSS NPS 4 bypass rupture.

For each scenario, the necessary vent path area to avoid high subcompartment pressure is identified and verified to be provided by the RXB design. The allowable differential pressure across building structural elements (e.g., walls) is set to ensure that building and reactor pool structural integrity is satisfactory.

3.6.2.3 Protection Methods

As discussed previously, methods employed in the NuScale design to address pipe ruptures vary by location and system.

- In the CNV, main steam and feedwater piping is designed to satisfy LBB. Reactor coolant system-connected intermediate piping locations are designed to satisfy criteria to avoid breaks, while terminal ends of RCS lines are analyzed for break effects.
- Above the NPM under the bioshield, breaks are excluded by identifying a design that satisfies criteria for break exclusion in the containment penetration areas or criteria to avoid breaks at the intermediate piping locations.
- In the RXB, the SSC requiring protection against rupture effects are generally separated in rooms not containing high- or moderate-energy piping, and bounding analysis is performed to ensure the structural integrity of the RXB itself.

The application of passive safety systems and the simplification of systems that remain eliminate both potential break locations and targets. Where breaks are postulated, the smaller-scale systems reduce the amount of energy available to drive blasts, pipe whip, and jet impingement. Short piping lengths, intervening obstacles, short jet reach, and hard targets resistant to damage lower the risk for interactions that could adversely affect the functionality of safety-related and essential SSC.

3.6.2.3.1 Restraints, Barriers, and Shields

Pipe whip restraints may be used to limit the motion of a broken pipe to prevent it from hitting an essential structure, system, or component. Protection for pipe whip and jet impingement is also available through barriers afforded by walls, floors, and other structures. Sufficiently large and robust SSC can also function as a pipe whip barrier or jet impingement shield.

3.6.2.3.1.1 Pipe Whip Restraints

Pipe whip restraints constrain movement of a broken pipe for purposes of preventing or limiting the severity of contact with essential SSC. Restraints installed only for purposes of controlled pipe whip are not ASME Code components; restraints that also serve a support function under normal or seismic conditions are designed to ASME criteria. The design criteria for pipe whip restraints are:

- Pipe whip restraints do not adversely affect structural margin of piping for other conditions.
  - Restraint design does not restrict thermal expansion and contraction.
  - The restraint design either: a) does not carry loads during normal operation or seismic events or b) the structural analysis includes a conservative load combination.
- Pipe whip restraints are located as close to the axis of the reaction thrust force as practicable. Pipe whip restraints are generally located so that a plastic hinge does not form in the pipe. If, due to physical limitations, pipe whip restraints are located so that a plastic hinge can form, the consequences of the whipping pipe and the jet impingement effect are further investigated. Lateral guides are provided where necessary to predict and control pipe motion. For further details, see the Pipe Rupture Hazards Analysis technical report TR-0818-61384.
- Generally, pipe whip restraints are designed and located with sufficient clearances between the pipe and the restraint, such that they do not interact and cause additional piping stresses. A design hot position gap is provided that allows maximum predicted thermal, seismic, and seismic anchor movement displacements to occur without interaction.
  - Exception to this general criterion may occur when a pipe support and restraint are incorporated into the same structural steel frame, or when a zero design gap is required. In these cases, the pipe whip restraint is included in the piping analysis and designed to the requirements of pipe support structures for all loads except pipe break, and designed to

the requirements of pipe whip restraints when pipe break loads are included.

- In general, the pipe whip restraints do not prevent access required to conduct inservice inspection examination of piping welds. When the location of the restraint makes the piping welds inaccessible for inservice inspection, a portion of the restraint is designed to be removable to provide accessibility.
- Analysis of pipe whip restraints
  - Is either dynamic or conservative static.
  - Static analysis includes
    - dynamic load factor of 2.0 to account for the initial pulse thrust force, unless a lower value is analytically justified
    - potential increase by a factor of 1.1 in loading due to rebound.
  - Loading combination includes dead weight, seismic, and the jet thrust reaction force
  - The criteria for analysis and design of pipe whip restraints for postulated pipe break effects are consistent with the guidelines in ANSI/ANS 58.2-1988.
  - Design is based on energy absorption principles by considering the elastic-plastic, strain-hardening behavior of the materials used.
  - Non-energy absorbing portions of the pipe whip restraints are designed to the requirements of AISC N690 Code.
  - Except in cases where calculations are performed to determine if a plastic hinge is formed, the energy absorbed by the ruptured pipe is assumed to be zero. That is, the thrust force developed goes directly into moving the broken pipe and is not reduced by the force required to bend the pipe.
  - In that a HELB is an accident (i.e., infrequent) event, pipe whip restraints are single use: allowed to deform provided the whipping pipe is restrained throughout the blowdown. Where structural members of a restraint are designed for elastic response, a dynamic increase factor is used.
  - Allowable strain in a pipe whip restraint is dependent on the type of restraint.
    - Stainless steel U-bar – this one-dimensional restraint consists of one or more U-shaped, upset-threaded rods or strips of stainless steel looped around the pipe but not in contact with the pipe. This allows unimpeded pipe motion during seismic and thermal movement of the pipe. At rupture, the pipe moves against the U-bars, absorbing the kinetic energy of pipe motion by yielding plastically.
    - Structural steel – this two-dimensional restraint is a stainless steel frame encircling the pipe that does not restrict pipe motion for

normal operation or earthquakes. Should a rupture occur, the pipe motion brings it into contact with the frame, absorbing the kinetic energy of the pipe by deforming plastically.

- Crushable material – if used, the allowable energy absorption of the material is 80 percent of its capacity based on dynamic testing performed at equivalent temperatures and at loading rates of ±50 percent of that determined by analysis.

  Note that a wall penetration may also serve as a two-dimensional pipe whip restraint, provided the wall has sufficient strength to resist the pipe load.

- Material properties are consistent with applicable code values, with strain-rate stress limits 10 percent above code or specification values, consistent with NRC guidance (SRP 3.6.2, III.2.A).

3.6.2.3.1.2 Pipe Whip Barriers

Standard Review Plan 3.6.2 identifies that an unrestrained, whipping pipe need not be assumed to cause ruptures or through-wall cracks in pipes of equal or larger NPS with equal or greater wall thickness. By extrapolation, a structure, system, or component made of metal of equivalent or better yield strength, equal or larger diameter, and equal or greater wall thickness does not only not leak or crack but also obstructs further travel of the whipping pipe, protecting SSC farther away from being struck.

The pipe whip load must be considered for inclusion in SSC load combinations to verify that the barrier is not displaced by pipe whip impact. For any structures added to serve as a barrier (or jet impingement shield), Seismic Category 1 loading is analyzed to confirm the structure does not fail and cause damage.

3.6.2.3.1.3 Jet Impingement Shields

NRC guidance does not have specific criteria for judging suitability of an SSC as a jet shield. Regarding impingement effects, if the following criteria are met, then the SSC is judged capable of serving as a shield without further evaluation:

- The diameter and wall thickness of the shield meet the criteria for a pipe whip barrier with a size equal or greater than that of the broken pipe.
- The barrier is of sufficient area and positioned to subtend a solid angle from the pipe break opening (considering potential pipe whip) that covers the essential SSC to be protected.
- The barrier is solid (without openings) to the extent that no direct line of sight exists from the break opening to the essential SSC. This criterion allows for some indirect passage of spray through an opening, but environmental qualification for pressurization and flooding demonstrates

functionality. The possibility of pipe whip affecting the location of the pipe break exit must be considered.

3.6.2.4 Guard Pipe Assembly Design Criteria

Guard pipes are not used.

3.6.2.5 Analytical Methods to Define Forcing Functions and Response Models

See Section 3.6.2.2.

3.6.2.6 Dynamic Analysis Methods to Verify Integrity and Operability

See Section 3.6.2.2.

3.6.2.7 Implementation of Criteria Dealing with Special Features

See Section 3.6.2.1.2.

Connection of Reactor Vent Valves and Reactor Recirculation Valves to the Reactor Vessel

In the NuScale design, each of three RVVs and two RRVs bolt directly to the reactor vessel. These five bolted-flange connections are classified as break exclusion areas. Because this configuration does not include a physical piping length, a majority of the BTP 3-4 B.A (ii) criteria do not apply. However, these BTP 3-4 B.A (ii) criteria generically involve design stress and fatigue limits and in-service inspection (ISI) guidelines, which are addressed for these bolted connections below.

Additionally, discussion is provided regarding threaded fastener design and leakage detection, to demonstrate that the probability of gross rupture is extremely low. The leakage detection systems along with in-service inspections provide assurance that potential failure mechanisms are detected before the onset of a catastrophic failure involving the fasteners of the bolted flange connections for the RRVs and RVVs, and therefore, that a break at this location need not be postulated.

<u>Design Stress and Fatigue Limits</u>

BTP 3-4 B.A(ii)(1) specifies more conservative stress and fatigue limits for ASME Class 1 piping in containment penetration areas than those required for piping by ASME Code, Section III, NB-3653. The bases for these more conservative limits include a desire to limit the stresses resulting from service loads (excluding those due to peak stresses) to within the material yield strength (i.e., elastic strains), and a concern that the cumulative usage factor calculation account for the possibility of a faulty design or improperly controlled fabrication, installation errors, and unexpected modes of operation, vibration, and other structural degradation mechanisms.

The RVV and RRV bolted connections are not classified as piping by their design specifications, and instead are classified as components designed to the rules of NB-3200. For the RVV and RRV bolt material (SB-637 UNS N07718), the design criteria given in NB-3230 for bolting provides greater margin against yielding due to service loads than do the rules of NB-3653 for typical piping system materials, even when considering the more restrictive limits of BTP 3-4 B.A(ii)(1). Therefore, the imposition of more conservative stress limits are not justified.

Additional limits on CUF are not justified because the risk of a faulty design and fabrication and installation errors for a flanged connection is low compared to that of a piping system. The possible degradation mechanisms applicable to Class 1 piping systems do not apply to the ECCS valve bolts. These considerations are addressed further below.

Faulty design is not a concern for the RVV and RRV flanges as the design features for these flanged connections that affect the stresses in the bolts are primarily the number and size of the bolts used, which are selected based on industry standards (ASME B16.5). The RVV and RRV flanged connections consist of Class 2500 NPS 5 and NPS 2 B16.5 flange configurations, respectively. ASME B16.5, "Pipe Flanges and Flanged Fittings," has a history of reliability. In addition to conforming to an industry standard design, detailed analysis is required to validate the design per ASME BPVC Section III, NB-3230, including a fatigue evaluation. The fatigue evaluation for these bolts utilizes the fatigue curve from ASME Section III, Division I, Mandatory Appendix I, Figure I-9.7. Figure I-9.7 was generated specifically for small diameter bolting made of SB-637 UNS N07718. Also, as required by NB-3230.3(c) for high strength bolting, a fatigue strength reduction factor of no less than 4.0 is applied to the bolts. The fatigue strength reduction factor specified for bolting further reduces the risk of a faulty design for the RVV and RRV bolting, as compared to ASME Class 1 piping systems.

To address fabrication concerns, additional surface and UT examinations, beyond the ASME code requirements for these components, have been specified to properly control fabrication. Bolts analyzed using NB-3232.3(b) have further requirements as stated in NB-3232.3(b)(2) and (3) that place controls on fabrication, by specifying both a minimum thread root radius and minimum radius between the head and shank, thus ensuring that the specified fatigue strength reduction factor used in the calculation of CUF is sufficiently conservative.

Unexpected modes of operation for piping systems in the nuclear industry generally involve thermal stratification, cycling, and striping. These situations do not apply to these valves. Unexpected vibration is another common concern, however, the RVVs and RRVs are within the scope of the NuScale Comprehensive Vibration Assessment Program (CVAP). As described in TR-0716-50439, "NuScale Comprehensive Vibration Assessment Program Technical Report," 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).

Other degradation mechanisms that have contributed to past piping failures and not already discussed are addressed below. Included is an explanation as to why these mechanisms are less likely to occur in the RVV and RRV valves than in a typical piping system.

- Corrosion - Not applicable as suitable materials have been selected and the bolts are not exposed to fluid.

- Erosion/ Flow Assisted Corrosion - Not applicable as there is no flow through these valves during normal operation and the bolts themselves are not exposed to fluid.
- Stress Corrosion Cracking (SCC) - Not applicable as suitable materials have been selected and the bolts themselves are not exposed to fluid.
- Water Hammer - Water hammer is not credible because there is no downstream piping and the valves discharge into a vacuum. Additionally, functional testing is performed for these valves including the dynamic effects of blowdown. Blowdown is classified as a service level B load in the ASME loading combinations for the valves, and therefore is included in the fatigue evaluations of the bolts.

In-Service Inspection

BTP 3-4 B.A(ii)(1) states that a 100% volumetric in-service examination of all pipe welds should be conducted during each inspection interval as defined in ASME Code, Section XI, IWA-2400. This requirement is addressed for the RVV and RRV bolting by providing augmented ISI requirements for these bolts that exceed the Code requirements. For in-service inspection, if the connection is disassembled during the interval, a UT inspection is performed on the bolts (Section 3.13.2). If the connection is not disassembled during the inspection interval, a volumetric inspection of the connection is performed in-place. Additionally, exceptions in the ASME code for flanged connections that allow only a sample of bolting to be inspected are not followed, and instead all flange bolts for all RVVs and RRVs are inspected during each inspection interval.

Threaded Fastener Design

The applicable guidelines and recommendations in NUREG-1339 have been adopted by NuScale. Lubricants containing molybdenum sulfide are prohibited for pressure-retaining bolted joints including the RVV and RRV joints. Of the degradation mechanisms listed in NUREG-1339, only SCC could potentially affect RVV and RRV bolted joints. Alloy 718 is highly resistant to SCC in borated water. To further improve Alloy 718 SCC resistance, the solution treatment temperature range prior to precipitation hardening treatment is restricted to 1800°F to 1850°F. Additionally, the RRV bolting is submerged in borated water only during refueling, at a much lower temperature than RCS operating temperature, further reducing SCC susceptibility. The RVV bolting materials are not submerged in borated water as part of any normal operating condition. Based on these considerations, SCC is unlikely for Alloy 718 studs for RVVs and RRVs. Threaded fastener design is discussed further in DCD Section 3.13.

Leakage Detection

FSAR Section 3.6.3 and FSAR Section 5.2.5 describe how the reactor coolant pressure boundary leakage detection systems conform to the sensitivity and response time recommended in Regulatory Guide 1.45, Revision 1. Leakage monitoring is provided by two means, the change in pressure within the CNV and collected condensate from the containment evacuation system. Even under a scenario where leakage occurs due to one or more postulated bolt breaks, containment leakage monitoring systems are sensitive to a leak rate as low as 0.01 lbm per minute (or ~0.001 gallon per minute). This is because the containment is a relatively small closed volume and is maintained at a pressure of less than 1 psia during normal operation. Compared to LBB leakage through other postulated cracks, the flange opening slit (if any) has a smoother flow surface (lower surface roughness compared to the crack morphology of fatigue cracks), and a straighter flow path that causes less pressure loss through the flow path in the Henry-Fauske's flow model. Therefore it is expected to result in a higher leak rate than through other postulated LBB fatigue cracks, when other conditions are similar. High containment pressure is also a safety actuation signal that initiates a reactor trip.

3.6.3 Leak-Before-Break Evaluation Procedures

General Design Criterion 4 includes a provision that the dynamic effects associated with postulated pipe ruptures may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. This analysis is called LBB. The LBB concept is based on the plant's ability to detect a leak in the piping components well before the onset of unstable crack growth.

For the NuScale Power Plant, the application of LBB is limited to the ASME Class 2 main steam and feedwater piping systems inside the CNV. The FWS piping analysis addresses significant feedwater cyclic transients and produces bounding loads for the ASME Class 2 piping with respect to LBB.

The methods and criteria to evaluate LBB are consistent with the guidance in Standard Review Plan 3.6.3 and NUREG-1061, Volume 3. Potential degradation mechanisms are described in Section 3.6.3.1; analysis for main steam and feedwater piping is provided in Section 3.6.3.4. Leak detection is discussed in Section 3.6.3.5.

3.6.3.1 Potential Degradation Mechanisms for Piping

In high-energy piping systems, environmental and operating material degradation could adversely affect the integrity of the system as well as the piping system LBB applicability. The application of LBB requires that the affected systems not be susceptible to environmental and operating degradation mechanisms such as erosion/corrosion, fatigue loads, stress corrosion cracking, creep damage, erosion damage, irradiation embrittlement or water hammer. These mechanisms are discussed below.

3.6.3.1.1 Erosion/Corrosion

Erosion/corrosion is a flow accelerated form of corrosion due to the breakdown of a protective oxide layer on the surface of the piping. Several instances of carbon steel pipe wall thinning due to erosion/corrosion have been documented, but there is no history of wall thinning due to erosion/corrosion of stainless steel piping at nuclear power plants. Austenitic stainless steel is resistant to wall thinning by erosion/corrosion.

The main steam and feedwater piping in the NPM is fabricated from SA-312 and SA-182 Type 304/304L (dual certified) austenitic stainless steel material and compatible austenitic stainless steel weld filler metals. The materials, in combination with water chemistry control, provide assurance that wall thinning by erosion-corrosion does not occur in the piping.

The secondary water chemistry monitoring and control program described in Section 10.3.5 ensures that chloride, oxygen, fluoride, and sulfate levels do not cause erosion/corrosion in austenitic stainless steel in the main steam and feedwater piping.

Stainless steel piping and components, such as letdown orifices, are potentially susceptible to erosion by cavitation under specific RCS flow conditions. Cavitation erosion has been observed in stainless steel piping in chemical and volume control systems of PWRs downstream of letdown orifices. Piping downstream of valves that significantly drop the pressure of the fluid in the system are also possible locations of cavitation erosion.

The main steam and feedwater piping inside the CNV do not have inline components that significantly decrease the pressure of the fluid in the piping in the direction of flow. Therefore, conditions conducive to fluid cavitation do not exist.

Based on the above discussion, erosion/corrosion induced wall thinning is not an issue for the main steam and feedwater piping subject to LBB.

3.6.3.1.2 Stress Corrosion Cracking

If any one of the following three conditions is not present, stress corrosion-cracking (SCC) does not take place. The three conditions are:

- There must be a corrosive environment.
- The material itself must be susceptible.
- Tensile stresses must be present in the material.

The main steam and feedwater piping is not susceptible to SCC because the piping is not exposed to a corrosive environment, the material is SCC resistant, and tensile stresses that could initiate SCC are not present.

The secondary water chemistry monitoring and control program described in Section 10.3.5 ensures that chloride, oxygen, fluoride, and sulfate levels do not cause SCC in austenitic stainless steel in the main steam and feedwater piping.

During reactor shutdown conditions, the outside surfaces of some piping inside the CNV are exposed to borated water. Minimizing the chloride levels in the water along with the low levels of oxygen in the water reduces the potential for SCC. The temperature of the water on the outside of the piping is maintained near room temperature, which prevents SCC initiation in conjunction with minimizing chlorides in solution. Water chemistry conditions during shutdown conditions are controlled to preclude SCC initiation from the outer surface of the piping, using water treatment methods discussed in Section 10.3.5.

SA-312 TP304/304L dual certified stainless steel is also resistant to SCC given adequate control of dissolved oxygen levels. The alloy contains 0.03 maximum weight percent carbon, which mitigates sensitization. The use of cold worked austenitic stainless steels is generally avoided; however, if used, the cold worked LBB pipes are followed by a solution annealing process.

Based on the above, the LBB piping is not susceptible to SCC.

3.6.3.1.3 Creep and Creep Fatigue

The design temperature for the MSS and FWS lines is 650 degrees F and normal operating temperatures are 585 degrees F and 300 degrees F respectively. Creep and creep fatigue are not a concern for austenitic steel piping below 800 degrees F. Because the design and operating temperatures of the piping systems are below these limits, creep and creep fatigue are not a concern.

3.6.3.1.4 Water Hammer/Steam Hammer

The potential for water hammer and relief valve discharge loads are considered and their effects minimized in the design of the main steam system. Utilizing drain pots, proper line sloping, and drain valves minimize this potential. The dynamic loads such as those caused by main steam isolation valve closure or Turbine Stop Valve closure due to water hammer and steam hammer are analyzed and accounted for in the design and analysis of the main steam piping. Therefore, the main steam piping is not susceptible to effects of water hammer.

The FWS and SG contain design features and operating procedures that minimize the potential for and effect of water hammer. The SG and FWS features are designed to minimize or eliminate the potential for water hammer in the steam generator FWS. The dynamic loads such as those caused by feedwater isolation valve closure and turbine trip due to water hammer are analyzed and accounted for in the design and analysis of the FWS piping. Therefore, the feedwater system LBB piping is not susceptible to water hammer.

The safe shutdown earthquake loading used for the LBB evaluations bounds the water hammer loading for both the feedwater lines and the main steam lines.

3.6.3.1.5 Fatigue

Low-cycle Fatigue

The main steam and feedwater piping inside the CNV is ASME Class 2. Class 2 piping systems incorporate stress range reduction factors in accordance with Subsection NC of Section III of the ASME BPVC to account for cyclic loading. The reduction factors mitigate the need for a detailed fatigue evaluation including the calculation of cumulative usage factors. This design requirement ensures the piping is not susceptible to low-cycle fatigue due to operational transients. Confirmation is to be provided in the pre-operational thermal expansion monitoring program.

High-cycle Fatigue

Main steam and feedwater piping design requirements also ensure the piping is not susceptible to high-cycle fatigue due to vibration. The main steam and feedwater lines are part of the NuScale Power Module and are included within the scope of the NuScale CVAP, see Section 3.9.2. 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.

3.6.3.1.6 Thermal Aging Embrittlement

No cast steel is used for the main steam and feedwater piping. Wrought austenitic stainless steel is used. This product form is not susceptible to thermal aging embrittlement at the maximum design temperature of the piping. To minimize thermal aging embrittlement in austenitic stainless steel welds, delta ferrite content is controlled using the methods in RG 1.31. Delta ferrite for austenitic stainless steel weld filler metals with low molybdenum content, such as Type 308/308L, is limited to 5FN to 20FN. Delta ferrite for austenitic stainless weld filler metals with higher molybdenum content, such as Type 316/316L, is limited to 5FN to 16FN.

3.6.3.1.7 Thermal Stratification

Thermal stratification in piping occurs when fluid at a significantly different temperature is introduced into a long horizontal run of piping. The main steam and feedwater lines inside the CNV do not have long horizontal runs and are therefore not susceptible to thermal stratification.

3.6.3.1.8 Irradiation Effects

The main steam and feedwater piping materials, including austenitic stainless steels and compatible stainless steel welds, are not susceptible to irradiation embrittlement at the radiation levels outside the reactor vessel.

The main steam and feedwater piping is not susceptible to Irradiation Assisted Stress Corrosion Cracking (IASCC) due to its low fluence. IASCC typically affects components such as core support structures in regions with high fluence, near the core and inside the reactor vessel. Because the main steam and feedwater piping is outside of the reactor vessel and above the core, the fluence is insufficient to be an IASCC concern.

3.6.3.1.9 Rupture from Indirect Causes

The main steam and feedwater lines subject to LBB analysis are located inside the CNV. Rupture by indirect causes (e.g., fires, missiles, or natural phenomena) is precluded by design.

- The NPM and the components inside the CNV are safety-related and Seismic Category I, this precludes adverse interactions from a seismic event.

- Also, being inside the CNV precludes fires, external missiles, or damage from moving heavy loads.
- There are no internal missile sources inside containment (see Section 3.5).
- Containment is flooded as part of the normal shutdown process, therefore flooding is considered in the design.

3.6.3.1.10 Cleavage Type Rupture

Cleavage type ruptures are not a concern for the main steam and feedwater lines. Austenitic stainless steel is highly ductile and resistant to cleavage type ruptures at system operating temperatures and the lower temperatures experienced during shutdown conditions.

3.6.3.2 Materials

The MSS and FWS piping is fabricated from SA-312 and SA-182 TP304/TP304L (dual certified) material.

Alloy 600 and weld metal Alloy 82/182 are not used in the NPM LBB piping discussed.

3.6.3.2.1 Geometry

The main steam piping is evaluated in six segments:

Section Geometry	Nominal Inside Diameter (in.)	Nominal Thickness t, (in.)
NPS 8, SCH 120 straight and curved pipe base metal	7.187	0.719
NPS 8, SCH 120 pipe-to-pipe weld	7.187	0.719
NPS 8, SCH 120 pipe-to-safe-end weld	7.187	0.719
NPS 12, SCH 120 straight and curved pipe base metal	10.75	1.000
NPS 12, SCH 120 pipe-to-safe-end weld	10.75	1.000
NPS 8, SCH 120 elbow base metal	7.187	0.719

The feedwater piping is evaluated in four segments:

Section Geometry	Nominal Inside Diameter (in.)	Nominal Thickness t, (in.)
NPS 5, SCH 120 straight and curved pipe base metal	4.563	0.500
NPS 5, SCH 120 pipe-to-pipe, pipe-to-tee, pipe-to-safe-end, tee-to-tee welds	4.563	0.500
NPS 4, SCH 120 straight and curved pipe base metal	3.624	0.438
NPS 4, SCH 120 pipe-to-tee pipe-to-safe-end welds	3.624	0.438

3.6.3.2.2 Operating Conditions and Load

The operating pressure and temperature for the MSS piping are 500 psia and 585 degrees F, respectively.

The operating pressure and temperature for the FWS piping are 525 psia and 300 degrees F, respectively.

3.6.3.2.3 Materials

The MSS piping base metal is made of SA-312 and SA-182 Grade TP304/TP304L (dual certified). The pipe-to-pipe weld and pipe-to-safe-end weld are both made with austenitic stainless steel weld filler material. The tensile material properties used in the analysis of MSS materials are either at 550 degrees F or 585 degrees F. It is acceptable to use material properties at 550 degrees F to approximate the material properties at the actual operating temperature (585 degrees F) because the variations in the material properties between these temperatures are insignificant.

The FWS piping base metal is made of SA-312 Grade TP304/TP304L. The pipe-to-pipe, pipe-to-safe-end, pipe-to-tee, tee-to-tee welds are made with austenitic stainless steel weld filler material. The tensile material properties used in the analysis of FWS materials are at 300 degrees F.

Only gas tungsten arc welding is used for main steam and feedwater piping subject to LBB qualification and the weld filler metals are limited to the following:

- SFA-5.9: ER308, ER308L, ER316, ER316L
- SFA-5.30: IN308, IN308L, IN316, IN316L

3.6.3.2.4 Tensile Material Properties

Material	σ_y (ksi)	σ_u (ksi)	E (ksi)	ε_o	α	n
Main Steam Piping						
SA-312 TP304	18.7(1)	63.4(1)	25450(1)	0.00073(5)	8.07(4)	3.80(4)
ER308L Weld	22.1(7)	75.0(2)	25450 (1)	0.00087 (5)	2.31(3)	3.28(3)
Feedwater Piping						
SA-312 TP304	22.4(1)	66.2(1)	27000(1)	0.00083(5)	2.411(3)	3.616(3)
ER308L Weld	25.4(6)	75.0(2)	27000(1)	0.00094(5)	2.126(3)	3.616(3)

Notes

(1) ASME Boiler and Pressure Vessel Code, Section II, Part D, 2013 Edition no Addenda.

(2) ASME Boiler and Pressure Vessel Code, Section II, Part C, 2013 Edition no Addenda.

(3) α, n are R-O Model coefficient and exponent evaluated by method for elastic plastic fracture analysis that determines the R-O parameters (α, n) from basic mechanical properties determined from the ASME Code.

(4) from Reference 3.6-10

(5) $\varepsilon_o = \sigma_y/E$

(6) The weld metal minimum yield strength is assumed to be 25.4 ksi at 300 degrees F. This value is obtained from the base metal yield strength ratioed up by the ratio of the weld metal minimum ultimate strength to the base metal minimum ultimate strength.

(7) The weld metal minimum yield strength is assumed to be 22.1 ksi at 575 degrees F. This value is obtained from the base metal yield strength ratioed up by the weld metal minimum ultimate strength to the base metal minimum strength.

3.6.3.2.5 Crack Morphology Parameters

For fatigue cracks in pipes, the crack morphology parameters are obtained from Tables 3.3 through 3.8 of NUREG/CR-6004, "Probabilistic Pipe Fracture Evaluations for Leak-Rate-Detection Applications," (Reference 3.6-10). The mean values are listed below:

Parameter (Units)	Mean Value
Global roughness (μinch)	1325
Local roughness (μinch)	317
Number of 90-degree turns ($inch^{-1}$)	64
Global path deviation	1.07
Global and local path deviation	1.33

3.6.3.3 Analysis Methodology

To ensure that an adequate margin exists for leak detection, the analysis assumes a leak rate 10 times larger than the minimum plant leak detection capability.

A margin of 2.0 on flaw size and a margin of 1.0 on load is used when using the algebraic sum load combination method as described in Section 3.6.3.3.1.1. Therefore, for a given flaw size that develops a detectable leakage with safety factor of 10, a fracture mechanics analysis is performed using twice the leakage flaw size to obtain a maximum allowable stress. The maximum allowable stress must be equal to or greater than the actual applied stress.

3.6.3.3.1 Load Combination Method

It is allowable to use either the absolute sum load combination method or the algebraic sum load combination method, which require different margins on the flaw size. Both load combination methods consider deadweight (DW), thermal expansion (TH), flow loads due to pressure (PR), safe shutdown earthquake (SSE) inertial and seismic anchor motion (SAM) loads.

3.6.3.3.1.1 Algebraic Sum Method

The axial force, F, and moment, M, can be algebraically summed if a margin factor S_M of 1.4 is applied for the applicable DW, TH, PR, SSE, and SAM loads.

$$F_{Combined} = S_M(|F_{DW} + F_{TH} + F_{PR}| + |F_{SSE}| + |F_{SAM}|) \quad \text{Eq. 3.6-2}$$

$$M_{i,\,Combined} = S_M(|M_{i,\,DW} + M_{i,\,TH} + M_{i,\,PR}| + |M_{i,\,SSE}| + |M_{i,\,SAM}|) \quad \text{Eq. 3.6-3}$$

Where F_{DW}, F_{TH}, F_{PR}, F_{SSE} and F_{SAM} are axial force (with a unit of lbf) due to deadweight, thermal expansion, internal pressure, SSE and SAM, respectively, and $M_{i,DW}$, $M_{i,TH}$, $M_{i,PR}$, $M_{i,SSE}$, and $M_{i,SAM}$ are moment (with a unit of in-lbf) due to deadweight, thermal expansion, internal pressure, SSE and SAM, respectively, for component i ($i = X, Y, Z$). S_M is the safety margin for load combination.

First, for the algebraic sum method of load combination, the margin S_M is set to 1.4. If the allowable flaw length from the flaw stability analysis is at least equal to the leakage size flaw, then the margin on load is met. Second, the margin S_M is set to 1.0 and if the allowable flaw length from the flaw stability analysis is at least twice the leakage size flaw, then the margin on flaw size is met.

3.6.3.3.1.2 Absolute Sum Method

The loads can also be combined based on individual absolute values as follows:

$$F_{Combined} = |F_{DW}| + |F_{TH}| + |F_{PR}| + |F_{SSE}| + |F_{SAM}| \quad \text{Eq. 3.6-4}$$

$$M_{i,\,Combined} = |M_{i,\,DW}| + |M_{i,\,TH}| + |M_{i,\,PR}| + |M_{i,\,SSE}| + |M_{i,\,SAM}| \quad \text{Eq. 3.6-5}$$

The total moment for the primary bending stress is calculated as square root of the sum of squares (SRSS):

$$M_{Combined} = \sqrt{M_{x\,,\,Combined}^2 + M_{y\,,\,Combined}^2 + M_{z\,,\,Combined}^2} \quad \text{Eq. 3.6-6}$$

For an absolute sum load combination method, the margin on the load S_M is set to 1.0. If the allowable flaw length from the flaw stability analysis is equal to at least twice the leakage size flaw, the margins on load and flaw size are met.

3.6.3.3.2 Piping Load Combination

For normal stress calculation, the algebraic sum is used for load combinations based on SRP 3.6.3 paragraph III.11(c)(iii). The normal operating axial force and moments are calculated by the following equations:

$$\begin{aligned} F &= F_{DW} + F_{TH} + F_{PR} \\ M_X &= (M_X)_{DW} + (M_X)_{TH} \\ M_Y &= (M_Y)_{DW} + (M_Y)_{TH} \\ M_Z &= (M_Z)_{DW} + (M_Z)_{TH} \end{aligned} \qquad \text{Eq. 3.6-7}$$

Where F_{DW}, F_{TH}, F_{PR}, $M_{i,DW}$ and $M_{i,TH}$ ($i = X, Y, Z$) are defined in Section 3.6.3.3.1.1.

The resultant moment is then calculated as the SRSS:

$$M = \sqrt{M_X^2 + M_Y^2 + M_Z^2} \qquad \text{Eq. 3.6-8}$$

For the maximum stress calculation, the maximum axial force and moments are:

$$\begin{aligned} F &= |F_{DW}| + |F_{PR}| + |F_{SSE}| \\ M_X &= |(M_X)_{DW}| + |(M_X)_{SSE}| \\ M_Y &= |(M_Y)_{DW}| + |(M_Y)_{SSE}| \\ M_Z &= |(M_Z)_{DW}| + |(M_Z)_{SSE}| \end{aligned} \qquad \text{Eq. 3.6-9}$$

Where $M_{i,SSE}$ ($i = X, Y, Z$) are defined in Section 3.6.3.3.1.1.

The resultant moment is then calculated as the SRSS:

$$M = \sqrt{M_X^2 + M_Y^2 + M_Z^2} \qquad \text{Eq. 3.6-10}$$

In the above equations, the moment due to the internal pressure is not included although it is included in Eq. 3.6-3 and Eq. 3.6-5, because the moment due to internal pressure is negligible. For limit load analysis, the thermal expansion and SAM loads are not included in Eq. 3.6-51 because they are secondary loads.

The stresses due to axial loads and moments are then calculated by:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \qquad \text{Eq. 3.6-11}$$

where,

A = cross-sectional area,

Z = section modulus,

M = moment, and

F = axial force.

3.6.3.3.3 Leak Rate and Leakage Flaw Size Calculation

3.6.3.3.3.1 Elastic-Plastic Fracture Mechanics Methods

The first step of the leakage rate calculation is to determine the crack opening area, based on elastic-plastic fracture mechanics methods. Although finite element method and computational fracture mechanics can be used to calculate crack opening displacement and crack opening area, it is computationally inefficient when applied for LBB, because many iterations may be needed to find the crack size and the crack opening displacement to produce a detectable leakage rate, or bounding analysis curves may need to be developed. The GE/EPRI method (Reference 3.6-14) is used in this LBB calculation because it is easier to implement and is validated by experimental data.

The GE/EPRI method was developed for three loading conditions: pure tension, pure bending, and combined tension and bending. The crack opening displacement includes an elastic portion and a perfectly-plastic portion based on a Ramberg-Osgood (R-O) material model in Eq. 3.6-12.

$$\frac{\varepsilon}{\varepsilon_0} = \frac{\sigma}{\sigma_0} + \alpha\left(\frac{\sigma}{\sigma_0}\right)^n \qquad \text{Eq. 3.6-12}$$

where,

ε = true strain,

ε_0 = reference strain (given by $\frac{\sigma_0}{E}$),

E = Young's modulus (psi),

σ = true stress (psi),

σ_0 = reference stress (the ASME Code-specified 0.2 percent offset yield strength σ_y in this calculation) (psi), and

α, n = R-O model coefficient and exponent.

3.6.3.3.3.1.1 Crack Opening Displacement for Through-Wall Cracks in Cylinders under Remote Bending

In the linear elastic range, the elastic crack opening displacement δ_e of the total mouth opening displacement δ of a pipe, as illustrated in Figure 3.6-19, due to a remote bending stress can be expressed as:

$$\delta_e = \frac{4\sigma_B a}{E} V_1^B\left(\frac{a}{b}, \frac{R}{t}\right) \qquad \text{Eq. 3.6-13}$$

where,

$$a = R_m\theta \text{ = half crack length at the mean radius,} \qquad \text{Eq. 3.6-14}$$

$$b = \pi R_m \text{ = half pipe circumference,} \qquad \text{Eq. 3.6-15}$$

θ = half crack angle in radians,

R = mean pipe radius, $R = R_m$,

E = modulus of elasticity.

$$\sigma_B = \frac{MR}{I} \text{ = remote bending stress} \qquad \text{Eq. 3.6-16}$$

M = remote bending moment.

$$I = \frac{1}{4}\pi\left(R_0^4 - R_i^4\right) \cong \pi R^3 t \text{ = area moment of inertia} \qquad \text{Eq. 3.6-17}$$

R_0, R_i = pipe outer and inner radius,

t = pipe wall thickness, and

V_1^B = influence function for elastic crack opening displacement under bending, given as tabulated values for various crack sizes and pipe geometries in Table 6-5 of Reference 3.6-2 for straight pipe, and in Tables F.1 and F.2 of Reference 3.6-5 for elbows.

It is noted that $\frac{a}{b} = \frac{\theta}{\pi}$, so they are used interchangeably.

The plastic portion of crack opening displacement is expressed as:

$$\delta_p = \alpha\varepsilon_0 a H_2^B\left(\frac{a}{b}, n, \frac{R}{t}\right)\left(\frac{M}{M_0}\right)^n \qquad \text{Eq. 3.6-18}$$

where,

α, n = R-O model coefficient and exponent, and

H_2^B = influence function for plastic crack opening displacement under bending, given as tabulated values for various crack sizes, material R-O model exponents, and pipe geometries in Tables 6-6, 6-7, and 6-8 of Reference 3.6-2 for straight pipe, and in Tables F.1 and F.2 of Reference 3.6-5 for elbows.

$$M_0 = 4\sigma_0 R^2 t\left(\cos\frac{\theta}{2} - \frac{1}{2}\sin\theta\right) = \text{reference bending moment} \qquad \text{Eq. 3.6-19}$$

A discussion of α-correction is presented in Section 3.6.3.3.3.1.3

The total crack opening displacement δ is then calculated by

$$\delta = \delta_e + \delta_p = \frac{4\sigma_B a}{E} V_1^B\left(\frac{a}{b}, \frac{R}{t}\right) + \alpha\varepsilon_o a H_2^B\left(\frac{a}{b}, n, \frac{R}{t}\right)\left(\frac{M}{M_0}\right)^m \qquad \text{Eq. 3.6-20}$$

3.6.3.3.3.1.2 Approach to Handle Combined Axial Force and Bending Moment

To apply the influence functions from the bending condition to combined tension and bending, the axial force can be converted to an equivalent bending moment and added to the applied moment. The stress intensity factors due to axial force and bending moment can be expressed as:

$$K_T = \frac{F}{2\pi R t}\sqrt{\pi a} F_T(\theta) \qquad \text{Eq. 3.6-21}$$

$$K_B = \frac{M}{\pi R^2 t}\sqrt{\pi a} F_B(\theta) \qquad \text{Eq. 3.6-22}$$

where,

$$F_T(\theta) = 1 + 7.5\left(\frac{\theta}{\pi}\right)^{\frac{3}{2}} - 15\left(\frac{\theta}{\pi}\right)^{\frac{5}{2}} + 33\left(\frac{\theta}{\pi}\right)^{\frac{7}{2}} \qquad \text{Eq. 3.6-23}$$

$$F_B(\theta) = 1 + 6.8\left(\frac{\theta}{\pi}\right)^{\frac{3}{2}} - 13.6\left(\frac{\theta}{\pi}\right)^{\frac{5}{2}} + 20\left(\frac{\theta}{\pi}\right)^{\frac{7}{2}} \qquad \text{Eq. 3.6-24}$$

Note that the equations are derived for R/t=10. It is expected that the approximation is acceptable for R/t between 5 and 20.

The equivalent moment due to an axial force P is then calculated by:

$$M_e = \frac{FR}{2}\frac{F_T(\theta)}{F_B(\theta)} \qquad \text{Eq. 3.6-25}$$

3.6.3.3.3.1.3 α - Correction to the Crack Opening Displacement Models

In Reference 3.6-6, the improved crack opening displacement estimation scheme is proposed to better match the GE/EPRI estimation to the experimental data. For pure bending or tension, the plastic part of the crack opening displacement is given below.

For pure bending

$$\delta_p = \alpha^{\frac{1}{n}}\varepsilon_0 a H_2^B\left(\frac{a}{b}, n, \frac{R}{t}\right)\left(\frac{M}{M_0}\right)^n \qquad \text{Eq. 3.6-26}$$

For pure tension

$$\delta_p = \alpha^{\frac{1}{n}}\varepsilon_0 a H_2^T\left(\frac{a}{b}, n, \frac{R}{t}\right)\left(\frac{F}{F_0}\right)^n \qquad \text{Eq. 3.6-27}$$

Here, α is replaced by the term $\alpha^{1/n}$. Because α is normally greater than 1, the effect of this term is to reduce the crack opening displacement relative to what would be computed using Eq. 3.6-18.

A different correction is needed for the combined tension and bending case because the plastic contributions from pure tension and pure bending cannot be added linearly. For a simplified approximation, the following is used:

$$\delta_p = 0.5(\alpha + \alpha^{1/n})\varepsilon_0 a H_2^B\left(\frac{a}{b}, n, \frac{R}{t}\right)\left(\frac{M}{M_0}\right)^n \qquad \text{Eq. 3.6-28}$$

The α-correction in Eq. 3.6-27 is applied when using the bending influence function with the equivalent moment calculated by Eq. 3.6-25.

3.6.3.3.3.1.4 Crack Opening Area and Hydraulic Diameter

The crack opening profile is assumed to be elliptical. The crack opening area is calculated by:

$$A_{crack} = \pi a \delta / 2 \qquad \text{Eq. 3.6-29}$$

The perimeter of an ellipse can be approximated by

$$P_{wetted} \approx \pi[3(a + \delta/2) - \sqrt{(3a + \delta/2)(a + 3\delta/2)}] \qquad \text{Eq. 3.6-30}$$

The hydraulic diameter is then calculated by

$$D_H = \frac{4A}{P_{wetted}} \qquad \text{Eq. 3.6-31}$$

The crack opening area and the hydraulic diameter are two major crack geometric parameters that are needed for leak rate analysis, as presented in Section 3.6.3.3.3.2.

3.6.3.3.3.2 Two-phase Critical Flow Model

The Henry-Fauske thermal-hydraulic model of two-phase flow (Reference 3.6-8, Reference 3.6-9, and Reference 3.6-10) through long channels, as illustrated in

Figure 3.6-20, forms the basis for the leak rate analysis. Compared to other simplified homogenous models, this model is a slip-flow model in the sense that the vapor has a higher velocity than the liquid in the vapor-liquid mixture of a two-phase flow system. A slip ratio, defined as the ratio of gas velocity to liquid velocity, is used in the homogeneous equilibrium model equations. When the two-phase mixture experiences critical flow, the time required for the fluid to reach thermodynamic equilibrium when moving into regions of lower pressure is comparable to the time that the fluid is flowing in the crack, which leads to non-equilibrium vapor generation rates for two-phase critical flows.

To account for these non-equilibrium effects, Henry and Fauske assumed that the mixture quality relaxes in an exponential manner toward the equilibrium quality that would be obtained in a long tube. The relaxation coefficient was calculated based on their experiments with the critical flow of a two-phase water mixture in long tubes, with the ratio of flow-path length to pipe inside diameter greater than 100.

3.6.3.3.3.2.1 Thermal-hydraulic Model of Two-phase Flow

In the LBB analysis, the Henry-Fauske model of two-phase flow through long channels is applied to calculate leak rates. Mass flux equilibrium is written in the following format:

$$\Psi = G_c^{\,2} - \left[\frac{x_c v_{gc}}{\gamma_o P_c} - (\nu_{gc} - \nu_{lc}) N_1 \frac{dx_e}{dP}\right]^{-1} = 0 \qquad \text{Eq. 3.6-32}$$

Subject to the constraint in terms of pressure equilibrium

$$\Omega = P_c + \Delta P_e + \Delta P_f + \Delta P_a + \Delta P_{aa} + \Delta P_k - P_o = 0 \qquad \text{Eq. 3.6-33}$$

where,

G_c = mass flux of the fluid at the crack exit plane,

$$x_e = \frac{S_o - S_l^{\,c}}{S_g^{\,c} - S_l^{\,c}} = \text{equilibrium fluid quality} \qquad \text{Eq. 3.6-34}$$

S_o = entropy at entrance of the crack plane,

$S_l^{\,c}$ = entropy of the saturated liquid at the crack exit plane pressure,

$S_g^{\,c}$ = entropy of the saturated vapor at the crack exit plane pressure,

$$N_1 = \begin{cases} 20x_e, \text{ if } x_e < 0.05 \\ 1.0, \text{ if } x_e \geq 0.05 \end{cases}$$ Eq. 3.6-35

$$x_c = N_1 x_e \left[1 - e^{-B\left(\frac{L_a}{D_H} - 12\right)} \right]$$ Eq. 3.6-36

L_a = flow-path length,

$D_H = \dfrac{4 \cdot \text{Crack Opening Area}}{\text{Crack Opening Perimeter}}$ = the hydraulic diameter perimeter (see Eq. 3.6-31),

B=0.0523 = a constant based on experiments used in calculating exponential mixture quality relaxation,

v_{gc} = specific volume of saturated vapor at exit pressure,

v_{lc} = specific volume of saturated liquid at exit pressure,

γ_o = isentropic expansion exponent,

P = pressure,

P_c = absolute pressure of the fluid at the crack exit plane,

P_0 = absolute pressure at the entrance of the crack plane,

$\Delta P_e = \dfrac{G_o^2 v_{lo}}{2C_D^2}$ = pressure loss due to entrance effects Eq. 3.6-37

G_o = mass flux of the fluid at the crack entrance plane,

v_{lo} = specific volume of the saturated liquid at the entrance pressure,

C_D = discharge coefficient. A value of 0.95 is recommended for tight cracks,

$\Delta P_f = f\dfrac{L_a}{D_i}\dfrac{\overline{G}_c^2}{2}[(1-\bar{x})\overline{v}_l + \bar{x}\bar{v}_g]$ = Pressure loss due to friction Eq. 3.6-38

$\bar{x}$ = average fluid quality,

$\bar{v}_g$ = average specific volume of saturated vapor,

$\bar{v}_l$ = average specific volume of saturated liquid,

$$f = \left[2\log\left(\frac{D_H}{2\mu}\right) + 1.74\right]^{-2}$$ = Von Karman friction factor Eq. 3.6-39

μ = crack face roughness,

$$\Delta P_a = \bar{G}_T^2[(1-x_c)v_{lc} + x_c v_{gc} - v_{lc}]$$ = pressure loss due to acceleration of the fluid as it flows through the crack Eq. 3.6-40

$\bar{G}_T$ = average mass flux in the two-phase region of crack flow,

ΔP_{aa} = acceleration pressure loss due to area change is assumed zero,

$$\Delta P_k = e_v \frac{\bar{G}^2}{2}[\bar{v}_l + \bar{x}(\bar{v}_g - \bar{v}_l)]$$ = pressure loss due to ends and protrusions Eq. 3.6-41

$\bar{G}^2$ = average mass flux G^2 of the fluid

$$e_v = e_n L_a$$ = the total loss coefficient over the flow path Eq. 3.6-42

e_n = the number of velocity heads lost per unit flow path length, which is given in Eq. 3.6-44.

Eq. 3.6-33 and Eq. 3.6-32 are evaluated by iteration to give the leak flow rate through the crack and the exit pressure for given crack inlet stagnation conditions and crack geometry.

3.6.3.3.3.2.2 Effective Crack Morphology Parameters

In NUREG/CR-6004 (Reference 3.6-10), a modified model was developed to define the surface roughness, effective flow path length and the number of turns as a function of the ratio of the crack opening displacement (δ) to the global roughness (μ_G) of the flow path, which is considered to be more realistic. The basic idea is depicted in Figure 3.6-21.

For a very tight crack, i.e., $\delta/\mu_G < 0.1$, the effective roughness is close to the local roughness (μ_L). But for a crack with wide opening, i.e., $\delta/\mu_G > 10$, the effective roughness is close to the global roughness. A linear function is used to calculate the effective roughness in between. The effective roughness, μ, is then expressed as

$$\mu = \begin{cases} \mu_L, & 0 < \frac{\delta}{\mu_G} < 0.1 \\ \mu_L + \frac{\mu_G - \mu_L}{9.9}\left(\frac{\delta}{\mu_G} - 0.1\right), & 0.1 \le \frac{\delta}{\mu_G} \le 10 \\ \mu_G, & \frac{\delta}{\mu_G} > 10 \end{cases} \qquad \text{Eq. 3.6-43}$$

Similarly, for a very tight crack, i.e., $\delta/\mu_G < 0.1$, the effective number of turns is close to the number of local turns. But for a crack with wide opening, i.e., $\delta/\mu_G > 10$, the effective number of turns decreases to about 10 percent of the local number of turns (e_{n_L}). A linear function is used to calculate the effective number of turns in between. The effective number of turns is then expressed as

$$e_n = \begin{cases} e_{n_L}, & 0 < \frac{\delta}{\mu_G} < 0.1 \\ e_{n_L} - \frac{e_{n_L}}{11}\left(\frac{\delta}{\mu_G} - 0.1\right), & 0.1 \le \frac{\delta}{\mu_G} \le 10 \\ 0.1 e_{n_L}, & \frac{\delta}{\mu_G} > 10 \end{cases} \qquad \text{Eq. 3.6-44}$$

In a similar way, the actual crack path to thickness ratio that represents the correction factor for flow path deviation from straightness is also a function of crack opening displacement. For a very tight crack, i.e., $\delta/\mu_G < 0.1$, the effective deviation is close to the global plus local path deviation K_{G+L}. But for a crack with wide opening, i.e., $\delta/\mu_G > 10$, the effective deviation is close to the global path deviation K_G. A linear function is used to calculate the effective deviations in between. The effective deviation factor is then expressed as:

$$\frac{L_a}{t} = \begin{cases} K_{G+L}, & 0 < \frac{\delta}{\mu_G} < 0.1 \\ K_{G+L} - \frac{K_{G+L} - K_G}{9.9}\left(\frac{\delta}{\mu_G} - 0.1\right), & 0.1 \le \frac{\delta}{\mu_G} \le 10 \\ K_G, & \frac{\delta}{\mu_G} > 10 \end{cases} \qquad \text{Eq. 3.6-45}$$

These crack opening displacement-dependent effective crack morphology parameters are plotted in Figure 3.6-22.

3.6.3.3.3.3 Detectable Leak Rate

The leakage of the piping systems inside the CNV can be detected by either using the CNV pressure sensor or the CES sample vessel instrumentation. See Section 3.6.3.5 for more discussion. The minimum detectable leak rate is 0.01 lbm/min, or 0.001 gallon per minute (GPM). Per SRP 3.6.3, a safety margin of 10 is required for the detectable leak rate. However, a more conservative leak rate of 0.2 lbm/min (or 2.0 lbm/min after the margin of 10 is applied) is used as the leak rate to construct the LBB bounding curves.

3.6.3.3.4 Flaw Stability Analysis Method (Limit Load Analysis)

It is required that any subcritical cracks, including surface and through-wall cracks in circumferential and axial directions be stable so that a catastrophic break is not possible. The cracks in an elbow also need to be evaluated if not bounded by the straight piping. Crack growth evaluation is required to be performed to ensure that cracks are stable.

It is usually found that circumferential through-wall cracks are more limiting than axial or surface cracks. Because the LBB analysis is performed for austenitic stainless steel piping systems, the stability assessment is based on limit load analysis.

A modified limit load analysis based on the master curve is used to calculate the allowable stable flaw size. The master curve is constructed to be a stress index S_I as a function of the postulated total circumferential through-wall flaw size $2a_c$. The stress index S_I and the half flaw size a_c are expressed as:

$$S_I = \begin{cases} \dfrac{2\sigma_f}{\pi}(2\sin\beta - \sin\theta) + (S_m)(P_m), \text{ if } \beta + \theta \le \pi \\ \dfrac{2\sigma_f}{\pi}\sin\beta + (S_m)(P_m), \text{ if } \beta + \theta > \pi \end{cases} \quad \text{Eq. 3.6-46}$$

where,

$$\beta = \begin{cases} 0.5\left[(\pi - \theta) - \dfrac{\pi P_m}{\sigma_f}\right], \text{ if } \beta + \theta \le \pi \\ -\dfrac{\pi P_m}{\sigma_f}, \text{ if } \beta + \theta > \pi \end{cases} \quad \text{Eq. 3.6-47}$$

$$P_m = \frac{F_x}{A} = \text{primary membrane stress} \quad \text{Eq. 3.6-48}$$

Fx = total applied axial force,

A = cross-section area,

$$\theta = \frac{a_c}{R_m} = \text{postulated through-wall circumferential crack half-angle} \qquad \text{Eq. 3.6-49}$$

R_m = pipe mean radius,

S_M = 1 = safety margin on the load,

$$\sigma_f = 0.5(\sigma_y + \sigma_u) = \text{flow stress} \qquad \text{Eq. 3.6-50}$$

σ_y = yield strength, and

σ_u = ultimate strength.

The stress index is also expressed in SRP 3.6.3 as:

$$S_I = S_M(P_m + P_b) \qquad \text{Eq. 3.6-51}$$

where,

$$P_b = \frac{M \cdot R_m}{I} = \text{primary bending stress} \qquad \text{Eq. 3.6-52}$$

$$M = \frac{\left(\sigma_{max} - \frac{F_x}{A}\right)I}{R_m} = \text{applied maximum moment} \qquad \text{Eq. 3.6-53}$$

σ_{max} = applied maximum stress, and

I = area moment of inertia.

The σ_{max} can be determined by making S_I in Eq. 3.6-46 equal to that in Eq. 3.6-51.

3.6.3.3.5 Development of Smooth Bounding Analysis Curve

To develop a smooth bounding analysis curve (SBAC), the following steps are used:

1) prepare the required inputs as discussed in Geometry and Material Properties Section 3.6.3.2.1 and Section 3.6.3.2.4, and Normal Loads Section 3.6.3.3.2

2) low normal stress case - calculate the axial force for normal operating pressure and the bending moment based on a selected lower magnitude of bending stress that is lower than the expected minimum bending stress

3) calculate the leakage flaw size at 100 percent power condition for 10 times the leak detection capability using the methodology discussed in Section 3.6.3.3.3

4) perform the stability analysis using the limit load methodology for austenitic stainless steel piping discussed in Section 3.6.3.3.4. The maximum bending

moment is determined for a critical flaw size of twice the leakage flaw size. The margin of 2 on flaw size shall be satisfied.

5) calculate the low normal stress and corresponding maximum stress using the axial force and the bending moments by Eq. 3.6-11 to establish the first point on the SBAC

6) high normal stress case - calculate the axial force for normal operating pressure and the bending moment based on a selected higher magnitude of bending stress that is close to the material flow stress. Calculate the corresponding maximum stress following Steps 3 through 4

7) establish the last point on the SBAC for the High Normal Stress Case following Steps 3 through 6

8) determine intermediate points along the abscissa by equal division of abscissa points between the first and the last points

9) calculate the intermediate points following Steps 3 through 5

10) develop the SBAC by joining these points to form a smooth curve

3.6.3.3.6 Application of SBACs

The SBACs are used during the design of the piping systems to provide a design that satisfies LBB criteria. In addition, the results of the piping analysis are reconciled to the SBACs to verify that the fabricated piping systems satisfy LBB criteria. To evaluate the LBB applicability, the results of the pipe stress analysis are compared to the applicable SBAC at the critical location with highest maximum stress. At critical locations, the load combination for the normal stress and maximum stress calculation uses the methods presented in Section 3.6.3.3.2. The procedure for LBB analysis discussed in this section is illustrated by a flow chart shown in Figure 3.6-18.

3.6.3.4 Analysis of Main Steam and Feedwater Piping inside Containment

3.6.3.4.1 Analysis of Main Steam Piping

Based on piping materials (base and weld metal) and configurations (pipe and elbow) in Section 3.6.3.2.1, six sections are analyzed. For each analysis, the piping stresses are determined based on the equations in Section 3.6.3.3.2. The SBAC are developed by first performing the limit load analysis to estimate the critical crack size based on Section 3.6.3.3.4. The half critical crack size is then used in the leakage rate analysis that builds in a safety margin of 2 on the crack size. The crack opening area is assumed to be constant through the thickness. The crack opening displacement is calculated using elastic-plastic fracture mechanics following Section 3.6.3.3.3. Plastic zone correction is not applied. Finally, the piping stresses and SBAC are compared to see if the pipe qualifies for LBB.

3.6.3.4.1.1 NPS 8 Straight Pipe Base Metal

3.6.3.4.1.1.1 Normal Stress and Maximum Stress

This analysis is for straight and curved NPS 8 pipes. Various locations in both main steam lines 1 and 2 are considered in this analysis. For each location, the normal stress and maximum stress are calculated using the equations in Section 3.6.3.3.2.

By using Eq. 3.6-7 and Eq. 3.6-8, the normal axial force and moment are calculated. The maximum axial force and moment are calculated using Eq. 3.6-9 and Eq. 3.6-10. Lastly, the axial end cap force due to the internal pressure is added to the normal and maximum axial forces for calculating stress using Eq. 3.6-11.

The resultant normal and maximum stresses for the main steam lines 1 and 2 locations are plotted (legends MS1 and MS2) in Figure 3.6-23.

3.6.3.4.1.1.2 SBAC Development

The limit load analysis is performed first to estimate the critical crack size based on methodology described in Section 3.6.3.3.4. Half of the critical crack size is then used in leakage rate analysis. The crack opening displacement calculation using elastic-plastic fracture mechanics is based on the methodology discussed in Section 3.6.3.3.3.

The leakage rate is calculated for the half critical crack size, which results in a leakage rate of 2.0 lbm/min, based on the detectable leak rate discussed in Section 3.6.3.3.3.3.

Following the steps in Section 3.6.3.3.5, more points with higher normal stress are established for developing SBAC. The resultant SBAC is illustrated in Figure 3.6-23. It is observed that the stress points are below the SBAC, demonstrating the analyzed section satisfies LBB criteria.

3.6.3.4.1.2 NPS 8 Pipe-to-Pipe Weld

This analysis is for circumferential welding between NPS 8 pipe and NPS 8 pipe. All NPS 8 pipe-to-pipe weld locations in both MS lines 1 and 2 are considered in this analysis. Following the same method described in Section 3.6.3.4.1.1, the normal and maximum stresses are calculated for each location in NPS 8 pipe-to-pipe weld. The resultant stresses are plotted in Figure 3.6-24.

The SBAC is developed using the same method described in Section 3.6.3.4.1.1.2, except the weld material properties used are for ER308L. Using the methodology discussed in Section 3.6.3.3.3 for the COD calculation, the resultant SBAC is illustrated in Figure 3.6-24. It is observed that the stress points are below the SBAC, demonstrating the analyzed section satisfies LBB criteria.

3.6.3.4.1.3 NPS 8 Pipe-to-Safe-End Weld

This analysis is for circumferential welding between NPS 8 pipe and a safe end. All NPS 8 pipe-to-safe-end locations in both main steam lines 1 and 2 are considered in this analysis. The calculated normal and maximum stresses are plotted in Figure 3.6-25.

The SBAC for NPS 8 pipe-to-safe-end weld is identical to that for NPS 8 pipe-to-pipe weld because their weld material and dimensions are identical. The SBAC chart, illustrated in Figure 3.6-25, shows that the stress points are below the SBAC, demonstrating the analyzed section satisfies LBB criteria.

3.6.3.4.1.4 NPS 12 Straight Pipe Base Metal

This analysis is for straight and curved NPS 12 pipes. Various locations in both main steam lines 1 and 2 are considered in this analysis. The calculated normal and maximum stresses are plotted in Figure 3.6-26.

For developing SBAC, the methodology discussed in Section 3.6.3.3.3 is used to calculate crack opening displacement. The resultant SBAC is illustrated in Figure 3.6-26. It is observed that the stress points are below the SBAC, demonstrating the analyzed section satisfies LBB criteria.

3.6.3.4.1.5 NPS 12 Pipe-to-Safe-End Weld

This analysis is for circumferential welding between a NPS 12 pipe and a safe end. All NPS 12 pipe-to-safe-end weld locations in both MS lines 1 and 2 are considered in this analysis. The calculated normal and maximum stresses are plotted in Figure 3.6-27.

For developing SBAC, the methodology discussed in Section 3.6.3.3.3 is used to calculate crack opening displacement. The resultant SBAC is illustrated in Figure 3.6-27. It is observed that the stress points are below the SBAC, demonstrating the analyzed section satisfies LBB criteria.

3.6.3.4.1.6 NPS 8 Elbow Base Metal

This analysis is for NPS 8 elbows. Various locations in both MSS lines 1 and 2 are considered in this analysis. The calculated normal and maximum stresses are plotted in Figure 3.6-28.

The resultant SBAC is illustrated in Figure 3.6-28. Note that the SBAC is developed by only four points because the V_1 parameters become negative with higher normal stresses. This is due to the fact that the available parameters are for θ=45° and 90°, while the calculated θ beyond the fourth point is away from that range. Therefore, the calculated results beyond the fourth point are not considered. However, the trend of the four points in SBAC shows that the stress points are below the SBAC, demonstrating the analyzed section satisfies LBB criteria.

3.6.3.4.2 Analysis of Feedwater Piping

Based on piping materials (base and weld metals) and geometric parameters in Section 3.6.3.2.1, four sections are analyzed. For each analysis, the piping stresses are determined based on the equations in Section 3.6.3.3.2. The SBAC are developed by first performing the leak rate analysis based on Section 3.6.3.3.3 to estimate the leakage crack size that produces a leak rate equal to 10 times the minimum detectable leak rate. The leakage crack size is then used as the half critical crack size in the limit load analysis, based on Section 3.6.3.3.4, building in a safety margin of 2 on the crack size. The crack opening displacement is calculated using elastic-plastic fracture mechanics following Section 3.6.3.3.3. Plastic zone correction is used for the purpose of $H_2{}^B$ function calculation for the NPS 4 FWS lines, to be consistent with the method in Reference 3.6-2. Finally, the piping stresses and SBAC are compared to confirm that the pipe qualifies for LBB.

3.6.3.4.2.1 Normal and Maximum Stress Calculations

For each location considered, the normal stress and maximum stress are calculated using the equations in Section 3.6.3.3.2.

By using Eq. 3.6-7 and Eq. 3.6-8, the normal axial force and moment are calculated. The maximum axial force and moment are calculated using Eq. 3.6-9 and Eq. 3.6-10. Lastly, the axial end cap force due to the internal pressures is added to the normal and maximum axial forces for calculating stress using Eq. 3.6-11.

3.6.3.4.2.2 NPS 4 Feedwater System Line Base Metal

Various locations in both FWS lines 1 and 2 are considered in the analysis for straight and curved NPS 4 pipe base metal. For each location, the normal stress and maximum stress are calculated using the equations in Section 3.6.3.3.2, following the method described in Section 3.6.3.4.2.1. The resultant normal and maximum stresses for the locations are then plotted (legends FWS Line 1 and FWS Line 2) in Figure 3.6-29, the SBAC Chart for NPS 4 FWS line base metal.

The SBAC is developed using the method described in Section 3.6.3.3.5. The stress points are below the SBAC, demonstrating that the analyzed section satisfies the LBB criteria.

3.6.3.4.2.3 NPS 4 Feedwater System Line Welds

The analysis addressed the circumferential welds including pipe-to-tee, and pipe to safe-end welds. All NPS 4 weld locations in both FWS lines 1 and 2 were considered. Following the same method described in Section 3.6.3.4.2.1, the normal and maximum stresses were calculated for each location of the NPS 4 line welds. The resultant stresses are plotted in Figure 3.6-30, the SBAC Chart for NPS 4 FWS line welds.

The SBAC is developed using the method described in Section 3.6.3.3.5 and plotted in Figure 3.6-30. The stress points are below the SBAC, demonstrating that the analyzed section satisfies the LBB criteria.

3.6.3.4.2.4 NPS 5 Feedwater System Line Base Metal

Various locations in both FWS lines 1 and 2 are considered in the analysis for straight and curved NPS 5 pipe base metal. Following the same method described in Section 3.6.3.4.2.1, the normal and maximum stresses are calculated for each location in the NPS 5 base metal. The calculated normal and maximum stresses are plotted in Figure 3.6-31, the SBAC Chart for NPS 5 FWS line base metal.

The SBAC is developed using the method described in Section 3.6.3.3.5 and plotted in Figure 3.6-31. The stress points are below the SBAC, demonstrating that the analyzed section satisfies the LBB criteria.

3.6.3.4.2.5 NPS 5 Feedwater System Line Welds

The analysis addressed the circumferential welds including pipe to tee, and pipe to safe end welds. All NPS 5 weld locations in both FWS lines 1 and 2 were considered. Following the same method described in Section 3.6.3.4.2.1, the normal and maximum stresses were calculated for each location of the NPS 5 line welds. The resultant stresses are plotted in Figure 3.6-32, the SBAC Chart for NPS 5 FWS line welds.

The SBAC is developed using the method described in Section 3.6.3.3.5 and plotted in Figure 3.6-32.

The stress points are below the SBAC, demonstrating that the analyzed section satisfies the LBB criteria.

3.6.3.4.3 Results and Conclusions

3.6.3.4.3.1 Main Steam System Piping

The LBB allowable maximum axial and bending stress loads are compared against the actual normal operating plus SSE loadings of the MSS piping. The actual loads (the combined axial loads and the combined bending stresses as defined in SRP 3.6.3), for a given LBB location, fall within the SBAC depicted in Figure 3.6-23, Figure 3.6-24, Figure 3.6-25, Figure 3.6-26, Figure 3.6-27 and Figure 3.6-28. Therefore, it is concluded that the MSS piping meets the LBB criteria.

3.6.3.4.3.2 Feedwater System Piping

The LBB allowable maximum axial and bending stress loads are compared against the actual normal operating plus SSE loadings of the FWS piping. The actual loads (the combined axial loads and the combined bending stresses as defined in SRP 3.6.3), for a given LBB location, fall within the SBAC depicted in

Figure 3.6-29, Figure 3.6-30, Figure 3.6-31 and Figure 3.6-32. Therefore, it is concluded that the FWS piping meets the LBB criteria.

3.6.3.5 Leak Detection

Section 5.2.5 describes the leak detection system for inside the CNV. The SRP 3.6.3 states "The specifications for plant-specific leakage detection systems inside containment are equivalent to those in Regulatory Guide 1.45." As noted in Section 5.2.5, the reactor coolant pressure boundary leakage detection systems for the NPM conform to the sensitivity and response times recommended in RG 1.45, Revision 1.

This section describes the analysis methods used to support the application of LBB to high-energy piping in the NPM.

Regulatory Guide 1.45 Regulatory Position 2.1 states plant procedures should include the collection of leakage to the primary reactor containment from unidentified sources so that the total flow rate can be detected, monitored, and quantified for flow rates greater than 0.05 gpm. According to RG 1.45 Regulatory Position 2.2, the plant should use leakage detection systems with a response time of no greater than 1 hour for a leakage rate of 1 gpm.

Leakage monitoring is provided by two means, change in pressure within the CNV and collected condensate from the CES sample vessel.

The minimum detectable leak rate for the CES sample vessel is not easily quantified, because all liquid or vapor leaks within the CNV are eventually collected in the CES sample vessel. Once in the CES sample vessel, the minimum detectable volume is 0.042 gal or 0.333 lb of liquid. While there is theoretically no minimum detectable leak rate, main steam and feedwater system leak rates of 0.001 gpm or 0.01 lbm/min take less than 60 minutes to accumulate more than the minimum detectable volume.

To satisfy Regulatory Position 2.1 of RG 1.45, once the operators observe a pressure change in containment, a leak rate procedure is initiated to quantify the total leak rate. This, combined with other indications can aid in determining the leak source. In this instance, leaks can be detected using the CES sample vessel, where condensable fluids are collected after they are removed from containment via the vacuum pumps. The sample vessel level is configured to alarm the control room. Once a higher equilibrium pressure is reached during a leak scenario, leak rate measurements can be taken with the CES alone, using the CES sample tank.

3.6.4 High Energy Line Break Evaluation (Non-LBB)

3.6.4.1 Postulation of Pipe Breaks in Areas Other than Containment Penetration

Where break locations are selected without the benefit of stress calculations, breaks are postulated at the piping welds to each fitting, valve, or welded attachment. Breaks in non-ASME Class piping are addressed in Section 3.6.2.1.8. Additionally, in accordance with BTP 3-4, Part B, Item A(iii)(4), if a structure is credited with separating a high-energy line from an essential SSC, that separating structure is designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect on the structure, irrespective of the fact that the criteria described in BTP 3-4, Part B, Items A(iii)(1) through (3) might not require the postulation of a break at that location.

3.6.4.2 NuScale Power Module Piping System Parameters

Table 3.6-4 lists the NuScale NPM piping along with the respective design and operating conditions. High-energy piping systems (i.e., CVCS, MSS, FWS, and DHRS) are evaluated for HELB both inside and outside the CNV. Although the DHRS condenser is manufactured from piping products, and analyzed to ASME Code, Class 2 piping rules, it is nonetheless considered a major component and not a piping system, thus breaks are not postulated.

Moderate-energy piping systems (i.e., RCCWS, CFDS and CES) are exempt from HELB and are not addressed further herein.

3.6.4.3 NuScale Power Module Piping Material

The high-energy piping systems are manufactured using ASME SA-312, dual-certified TP304/TP304L stainless steel, with the properties shown in Table 3.6-5, which are taken from ASME Section II, Materials. Dual-certified TP304/TP304L SS maintains the low-carbon content of the TP304L SS grade and exhibits the higher strength associated with the straight grade of TP304 SS. Thus, Table 3.6-5 uses the strength properties from the straight TP304 SS grade at design temperature of 650 degrees F shown in Table 3.6-4. Note that S_A in Table 3.6-5 is calculated with a 1.0 stress range reduction factor, f.

3.6.5 References

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Table 3.6-1: High- and Moderate-Energy Fluid System Piping

System Name	Individual Line Names	Line size (NPS)	High- or Moderate-Energy
	Inside the Containment Vessel		
RCS	RCS injection	2	High
	RCS discharge	2	High
	High point vent	2	High
	Pressurizer spray	2	High
SGS	Steam	12 & 8	High
	Feedwater	5 & 4	High
DHRS	DHRS condensate return lines 1 and 2	2	High
CRDS	CRDS cooling	2	Moderate
CFD	Containment flooding and drain system	2	Moderate[1]
	Outside the CNV to the NPM Disconnect Flange		
CVCS	RCS injection (Note 4)	4 & 2	High
	RCS discharge (Note 4)	4 & 2	High
	High point vent (Note 4)	4 & 2	High[3]
	Pressurizer spray (Note 4)	4 & 2	High
CES	Containment evacuation	2 & 4	Moderate
MSS	Steam	12	High
FWS	Feedwater	6 & 5 & 4	High
DHRS	Decay heat removal system lines 1 and 2	8 & 6 & 2	High
RCCW	CRDS cooling	4 & 2	Moderate
CFD	Containment flooding and drain system	4 & 2	Moderate[1]
	In the Reactor Building (outside the NPM Disconnect Flange)		
ABS	Auxiliary boiler system	6	High
CFDS	Containment flooding and drain system	4	High
CFWS	Condensate and feedwater system	6	High
CVCS	Chemical and volume control system	3	High
MSS	Main steam system	12	High
MHS	Module heatup system	3	High
NDS	Nitrogen distribution system	2	High
PSS	Process sampling system	0.75	High(2)
BAS	Boron addition system	3	Moderate(1)
CES	Containment evacuation system	4	Moderate
CHWS	Chilled water system	6	Moderate
DWS	Demineralized water system	4	Moderate
FPS	Fire protection system	16	Moderate
IAS	Instrument and control air system	2	Moderate
LRWS	Liquid radioactive waste system	2.5	Moderate(1)
PCUS	Pool cleanup system	10	Moderate
PSCS	Pool surge control system	10	Moderate
RCCWS	Reactor component cooling water system	8	Moderate
RPCS	Reactor pool cooling system	10	Moderate
RWDS	Radioactive waste drain system	3.5	Moderate
SAS	Service air system	2	Moderate
SCW	Site cooling water	38	Moderate
SFPCS	Spent fuel pool cooling system	10	Moderate
SRW	Solid radioactive waste system	3	Moderate

Table 3.6-1: High- and Moderate-Energy Fluid System Piping (Continued)

System Name	Individual Line Names	Line size (NPS)	High- or Moderate-Energy
UWS	Utility water system	(5)	Moderate
	In the Control Building		
BPDS	Balance-of-plant drain system	8	Moderate
CHWS	Chilled water system	10	Moderate
DWS	Demineralized water system	0.5	Moderate
FPS	Fire protection system	16	Moderate
IAS	Instrument and control air system	2	Moderate
PWS	Potable water system	(5)	Moderate
	In the Radioactive Waste Building		
CHWS	Chilled water system	6	Moderate
DWS	Demineralized water system	4	Moderate
FPS	Fire protection system	12	Moderate
GRWS	Gaseous radioactive waste system	2	Moderate
IAS	Instrument and control air system	2	Moderate
LRWS	Liquid radioactive waste system	3	Moderate
NDS	Nitrogen distribution system	2	Moderate
PSCS	Pool surge control system	2	Moderate
RWDS	Radioactive waste drain system	3	Moderate
SAS	Service air system	2	Moderate
SRWS	Solid radioactive waste system	3	Moderate
	Outside the Control Building, Reactor Building, and Radioactive Waste Building		
ABS	Auxiliary boiler system	6	Moderate(1)
BPDS	Balance-of-plant drain system	14	Moderate
BPSS	Backup power supply system	(5)	Moderate
CFWS	Condensate and feedwater system	12	High
CHWS	Chilled water system	14	Moderate
CPS	Condensate polishing system	6	Moderate
CWS	Circulating water system	84	Moderate
DWS	Demineralized water system	6	Moderate
FPS	Fire protection system	16	Moderate
FWTS	Feedwater treatment system	3	High
IAS	Instrument and control air system	4	Moderate
LRWS	Liquid radioactive waste system	2	Moderate
MSS	Main steam system	16	High
NDS	Nitrogen distribution system	2	Moderate
PSCS	Reactor pool surge control system	10	Moderate
PWS	Potable water system	(5)	Moderate
PSS	Process sampling system	0.75	High(2)
RWDS	Radioactive waste drain system	2	Moderate
RWS	Raw water system	(5)	Moderate
SAS	Service air system	4	Moderate
SDS	Site drainage system	(5)	Moderate
SCWS	Site cooling water system	52	Moderate
TGS	Turbine generator system	16	High

Table 3.6-1: High- and Moderate-Energy Fluid System Piping (Continued)

System Name	Individual Line Names	Line size (NPS)	High- or Moderate-Energy
UWS	Utility water system	36	Moderate

Notes:

(1) Based on operating parameters that exceed 200 degrees F or 275 psig for less than 2 percent of the time the system is in operation, or that exceed 200 degrees F or 275 psig for less than 1 percent of the plant operation time.

(2) Based on the nominal diameter of the lines, breaks do not need to be postulated in PSS lines.

(3) The High point vent can be considered moderate-energy, but is conservatively evaluated as high-energy.

(4) The safe end-to-valve welds for the 2-inch CVCS lines outside the CNV are NPS4. NPS4 applies only to the single weld.

(5) Hydraulic calculations have not been completed to determine system piping sizes.

Table 3.6-2: Postulated Break Locations

Line	ASME Class	Postulated Break Location
Break locations inside containment		
RCS injection	1	Terminal end - RPV head
		Terminal end - containment boundary
RCS discharge	1	Terminal end - RPV head
		Terminal end - containment boundary
Pressurizer spray	1	Terminal end - RPV head
		Terminal end - containment boundary
RCS high-point vent	1	Terminal end - RPV head
		Terminal end - containment boundary
DHRS #1	2	Terminal end - containment boundary
DHRS #2	2	Terminal end - containment boundary
Break locations outside the CNV under the bioshield		
None		
Break locations in the RXB (outside the NPM bioshield) bounded, as documented in the NuScale Pipe Rupture Hazards Analysis.		

Table 3.6-3a: Not Used

Table 3.6-3b: Not Used

Table 3.6-4: NuScale Power Module Piping Systems Design and Operating Parameters

Process System (NuScale System)	ASME Code	NPS Size	Design		Operating(4)	
			Press. (psia)	Temp. (°F)	Press. (psia)	Temp. (°F)
CVCS (RCS)	Class 1	2	2100	650	1870(2)	625(2)
CVCS (CNTS, CVCS)	Class 3(1)	2(1)	2100	650	1870(2)	625(2)
MSS (steam generator system, CNTS)	Class 2	8 & 12	2100	650	500	585
FWS (steam generator system, CNTS)	Class 2	4 & 5	2100	650	550	300
DHRS	Class 2	2 & 6	2100	650	1400	635(3)
RCCWS (CRDS)	Class 2	2	165	200	80	121
RCCWS (CNTS)	Class 2	4	1050	550	80	121
CFDS (CNTS-inside CNV)	Class 2	2	165	300	85	100
CFDS (CNTS-outside CNV)	Class 2	4	1050	550	85	100
CES (CNTS)	Class 2	4	1050	550	0.037	100

Notes

(1) The weld between the CIV and the safe-end is NPS 4 SCH 160 and is designated as a Class 1 piping weld

(2) Represents the highest normal operating pressure for the injection line and highest normal operating temperature for the RPV high point degasification line.

(3) Conservatively represents the highest normal operating temperature for the steam portion (i.e., NPS 6 portion) of the DHRS.

(4) The initial conditions are selected to bound system conditions for any power level, 102 percent thermal power and hot standby operation, for which the NuScale equivalent is referred to as hot shutdown. During hot shutdown, MSS pressure and temperature are approximately 300 psia and 420 degrees F, respectively, and primary pressure and temperature are approximately 1850 psia and 420 degrees F, respectively.

Table 3.6-5: Mechanical Properties for Piping Material

<table>
<tr><th rowspan="2">System</th><th rowspan="2">ASME Class</th><th colspan="3">Room Temp</th><th colspan="6">Design Temp</th><th>Operating Temp</th></tr>
<tr><th>S_y (ksi)</th><th>S_u (ksi)</th><th>S_c (ksi)</th><th>S_y (ksi)</th><th>S_u (ksi)</th><th>S_m (ksi)</th><th>S_h (ksi)</th><th>S_A (ksi)</th><th>E (10^6 psi)</th><th>S_y (ksi)</th></tr>
<tr><td>CVCS (RCS)</td><td>1</td><td rowspan="5">30</td><td rowspan="5">75</td><td rowspan="5">20.0</td><td rowspan="5">18.0</td><td rowspan="5">63.4</td><td>16.2</td><td>NA</td><td>NA</td><td rowspan="5">25.1</td><td>18.2</td></tr>
<tr><td>CVCS (CNTS, CVCS)</td><td>3</td><td rowspan="4">NA</td><td rowspan="4">16.2</td><td rowspan="4">29.05</td><td>18.2</td></tr>
<tr><td>FWS</td><td>2</td><td>22.4</td></tr>
<tr><td>MSS</td><td>2</td><td>18.6</td></tr>
<tr><td>DHRS</td><td>2</td><td>18.1</td></tr>
</table>

Table 3.6-6: Allowable Stresses for Class 1 Piping (ksi)

Process System	$2.4S_m$	$2.25S_m$	$1.8S_y$	$1.2S_m$
CVCS (RCS)	38.88	36.45	32.40	19.44

Table 3.6-7: Allowable Stresses for Class 2 & 3 Piping (ksi)

Process System	$0.8(1.8S_h+S_A)$	$2.25S_h$	$1.8S_y$	$0.4(1.8S_h+S_A)$
CVCS (CNTS, CVCS)	46.57	36.45	32.40	23.28
FWS				
MSS				
DHRS				

Table 3.6-8: Not Used

Figure 3.6-1: Flowchart of methodology for evaluation of line breaks

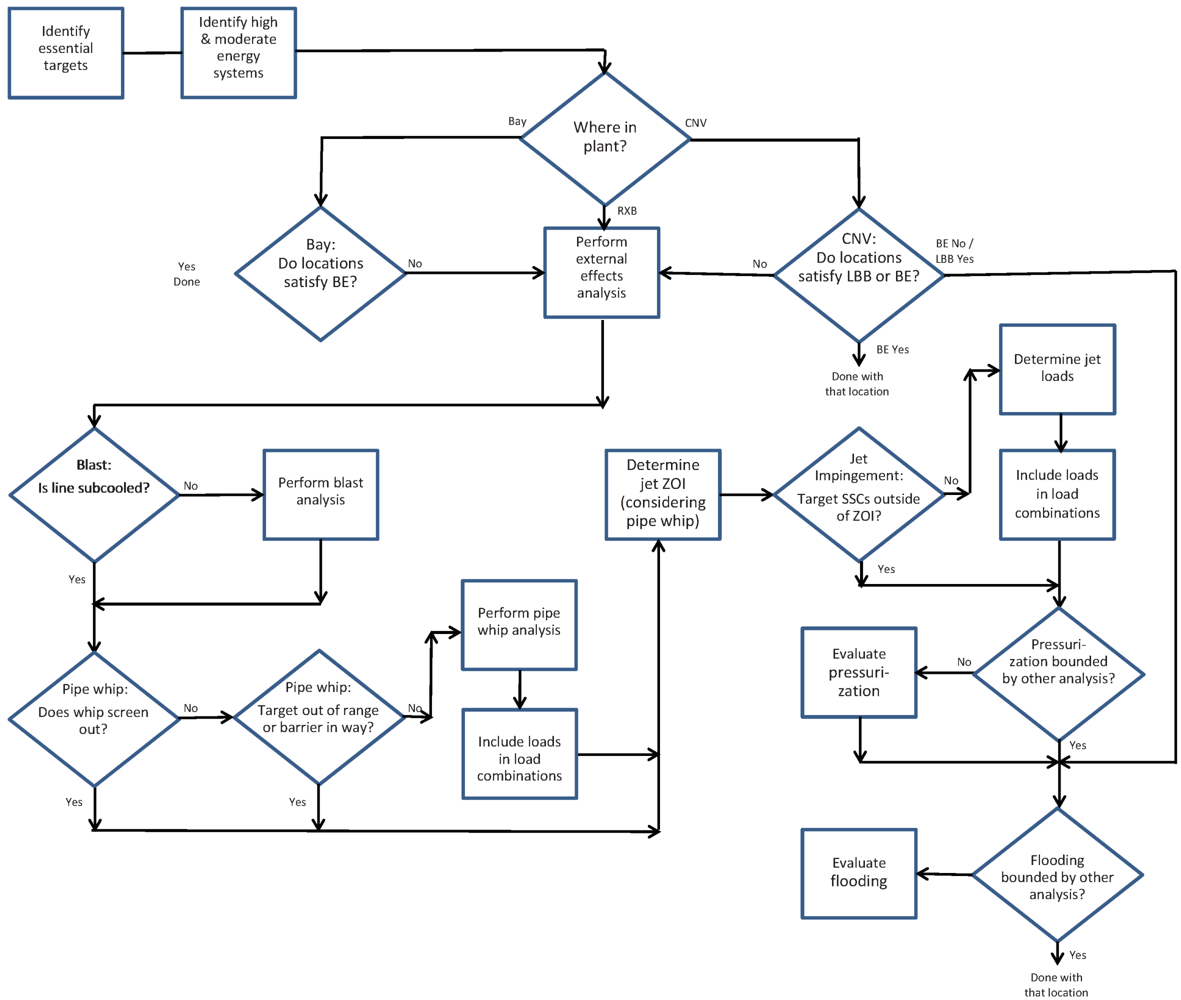

Figure 3.6-2: Not Used

Figure 3.6-3: Not Used

Figure 3.6-4: Not Used

Figure 3.6-5: Not Used

Figure 3.6-6: Not Used

Figure 3.6-7: Not Used

Figure 3.6-8: Not Used

Figure 3.6-9: Not Used

Figure 3.6-10: Not Used

Figure 3.6-11: Not Used

Figure 3.6-12: Not Used

Figure 3.6-13: Not Used

Figure 3.6-14: Not Used

Figure 3.6-15: Not Used

Figure 3.6-16: Not Used

Figure 3.6-17: Not Used

Figure 3.6-18: Flow Chart for Piping Leak-Before-Break Evaluation

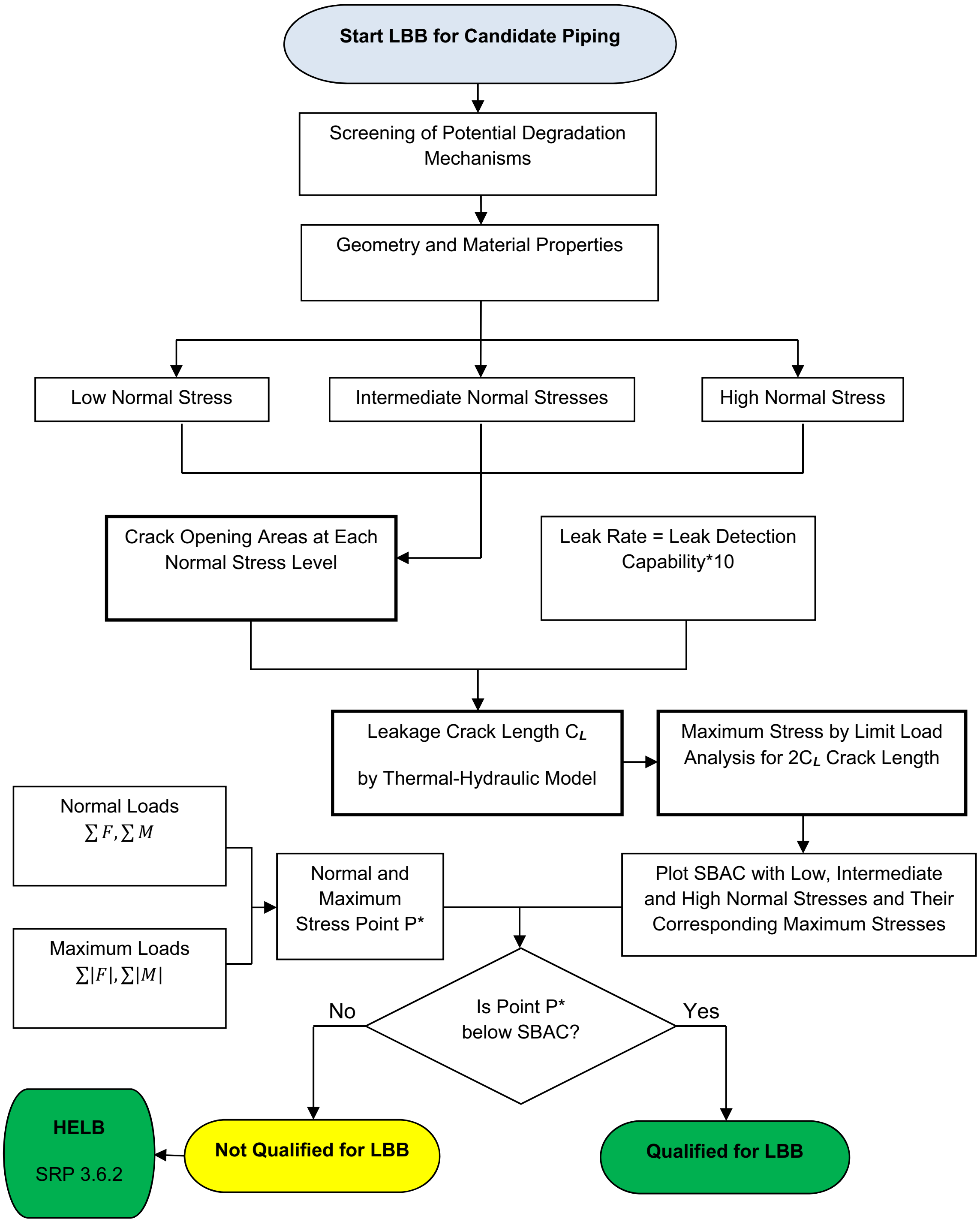

Figure 3.6-19: Illustration of Pipe with a Circumferential Through-Wall Crack

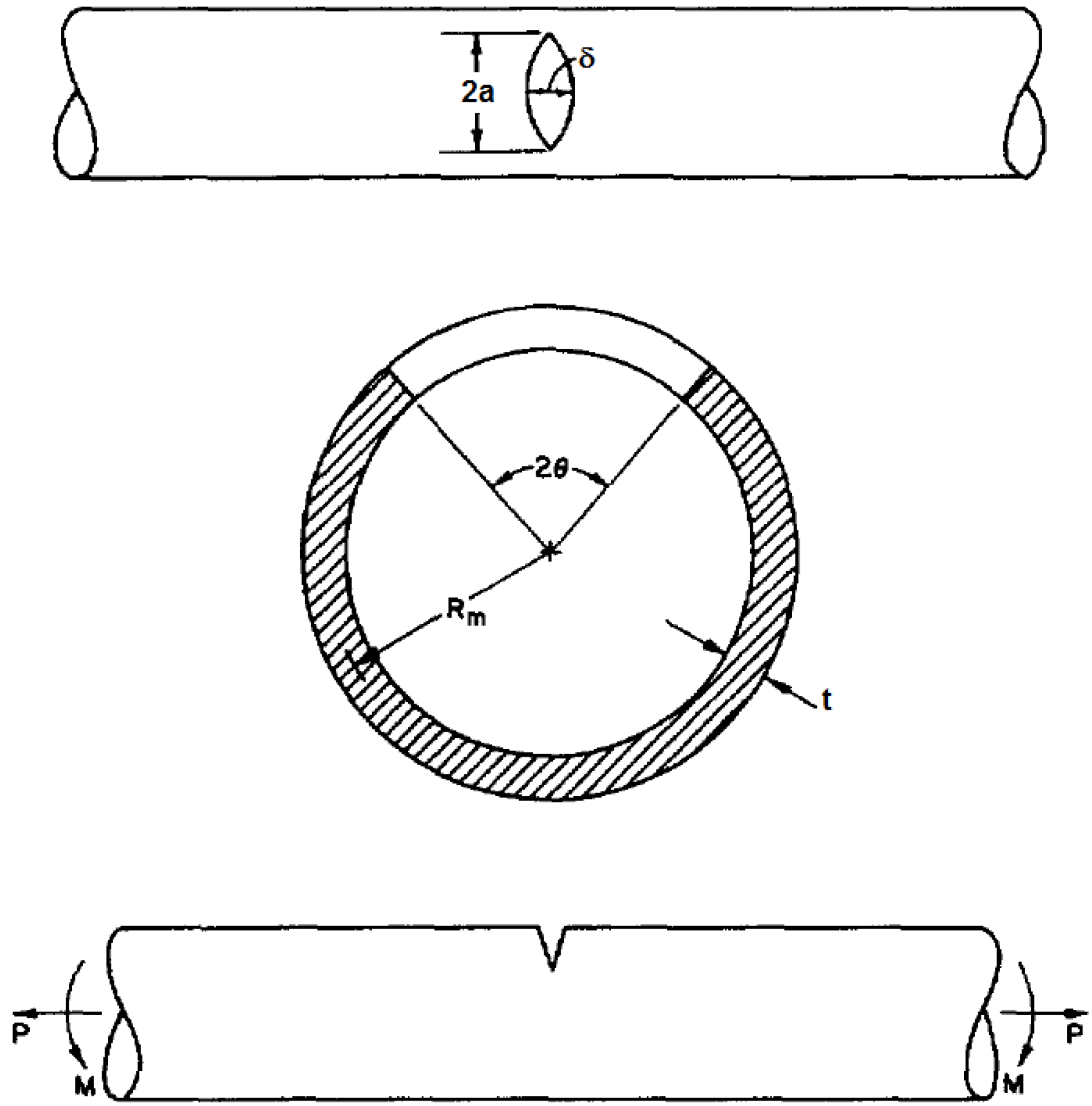

Figure 3.6-20: Henry-Fauske's Model of Two-Phase Flow

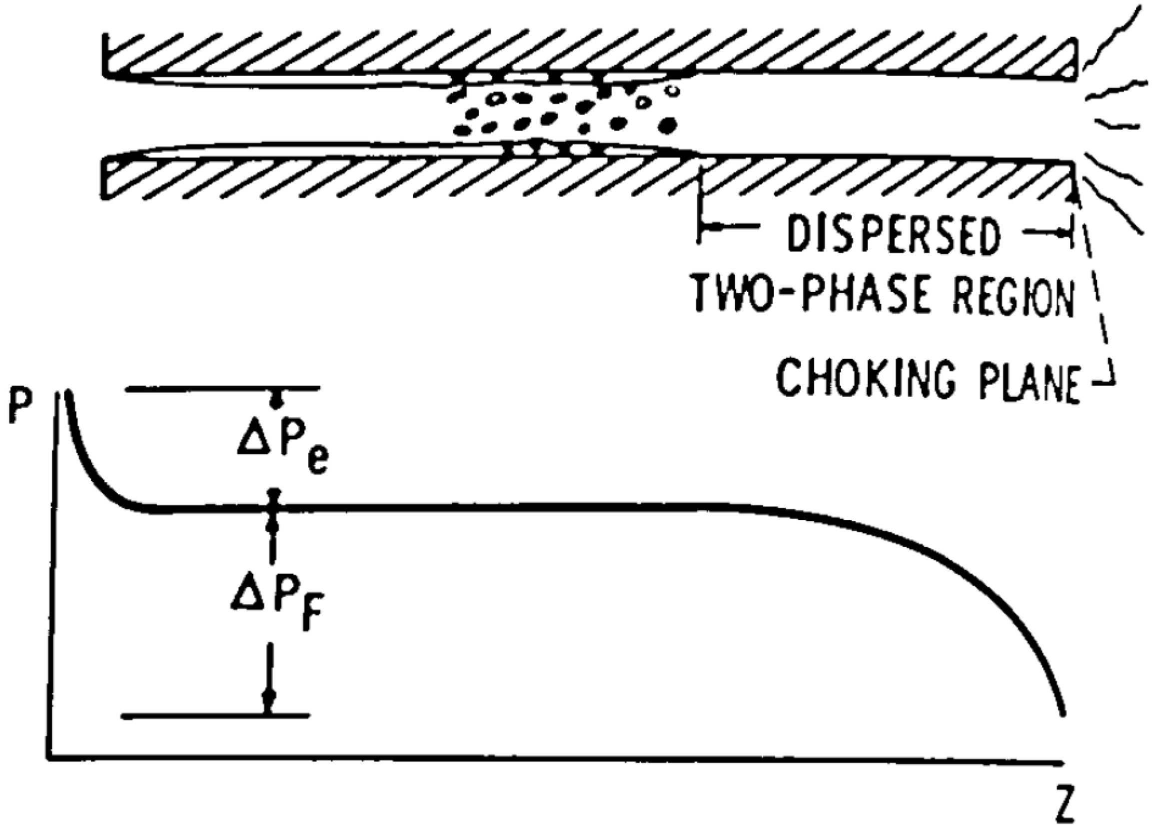

Figure 3.6-21: Local and Global Surface Roughness and Turns

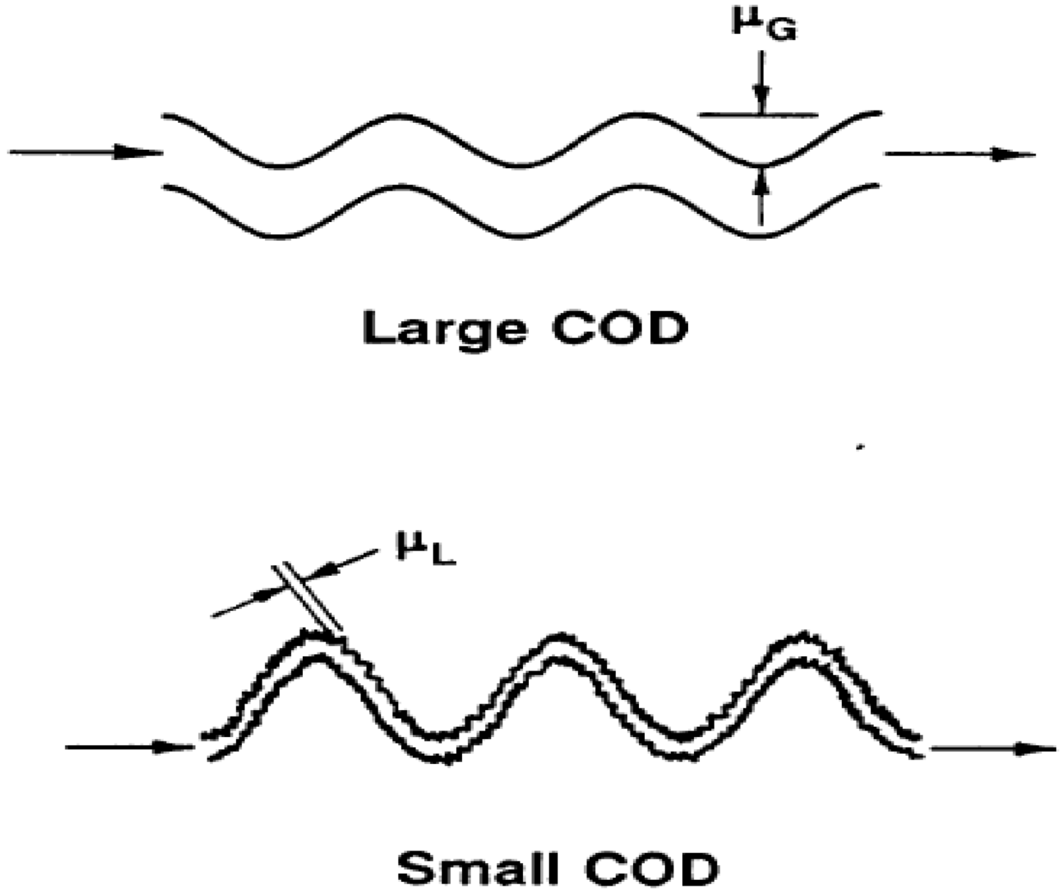

Figure 3.6-22: Crack Opening Displacement-Dependent Effective Crack Morphology

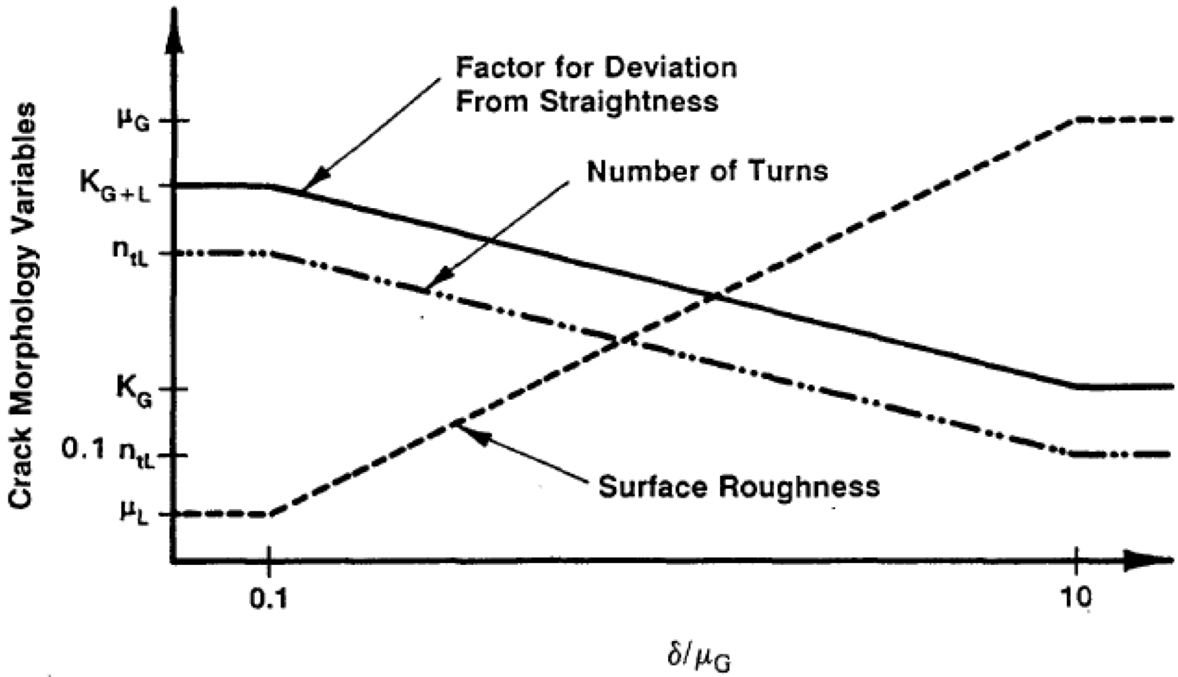

Figure 3.6-23: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 8 Straight Pipe Base Metal

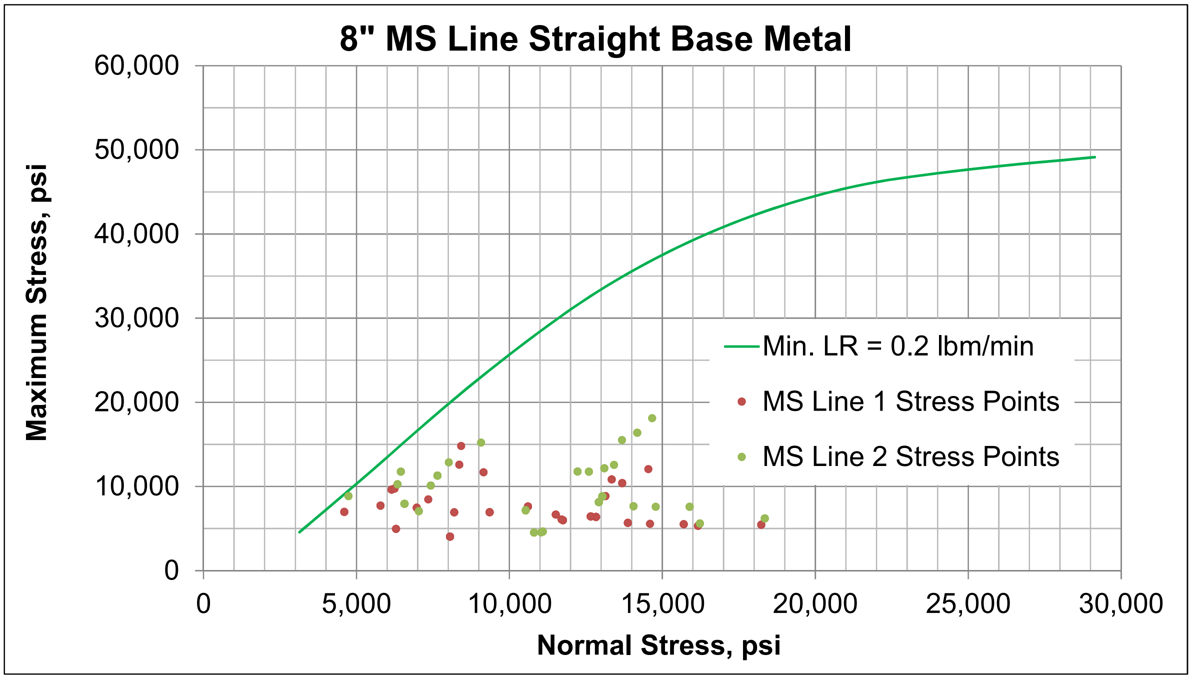

Figure 3.6-24: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 8 Pipe-to-Pipe Weld

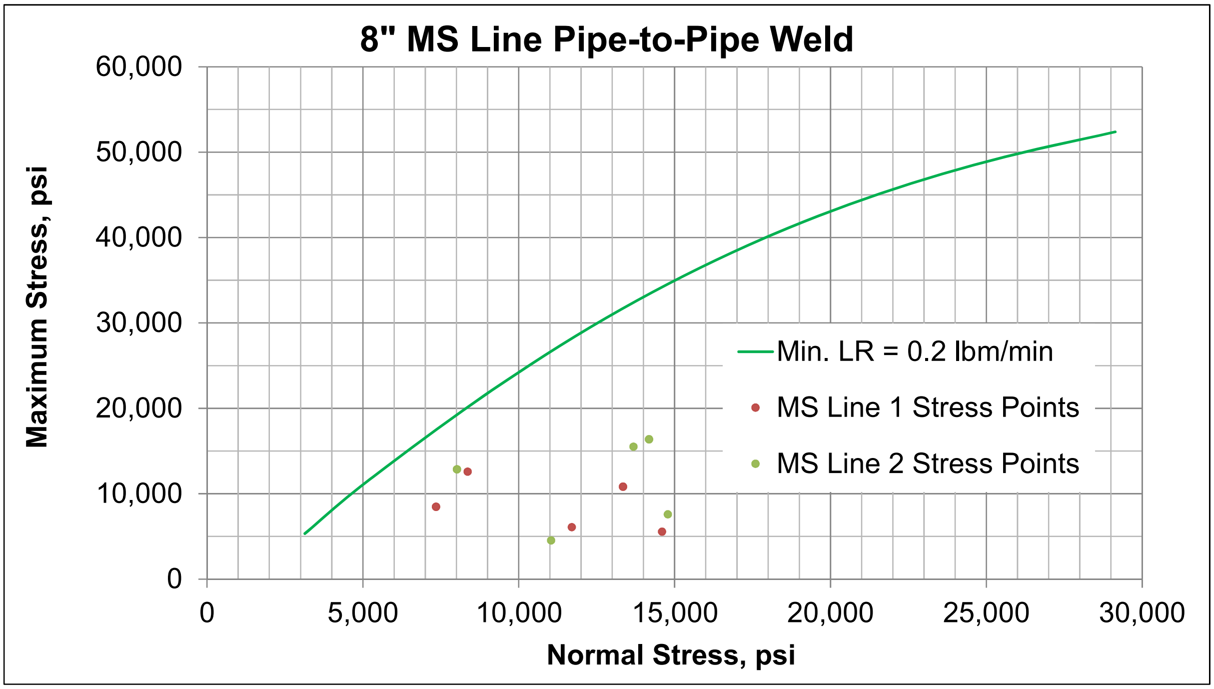

Figure 3.6-25: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 8 Pipe-to-Safe-End Weld

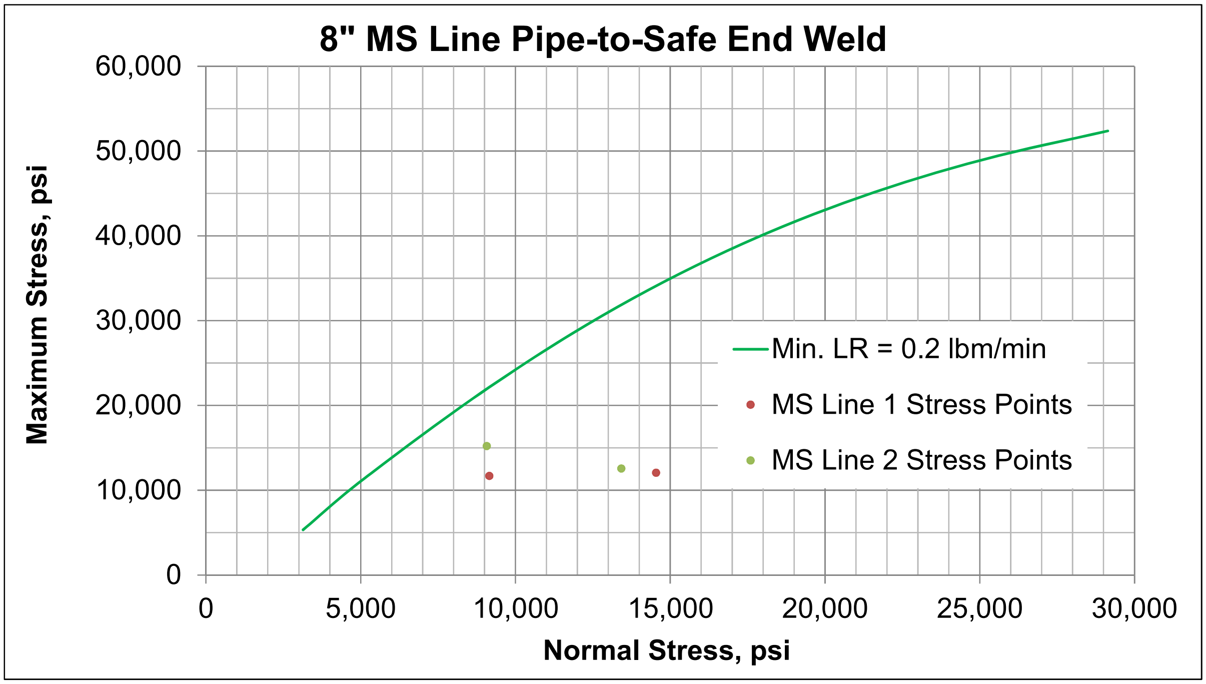

Figure 3.6-26: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 12 Straight Pipe Base Metal

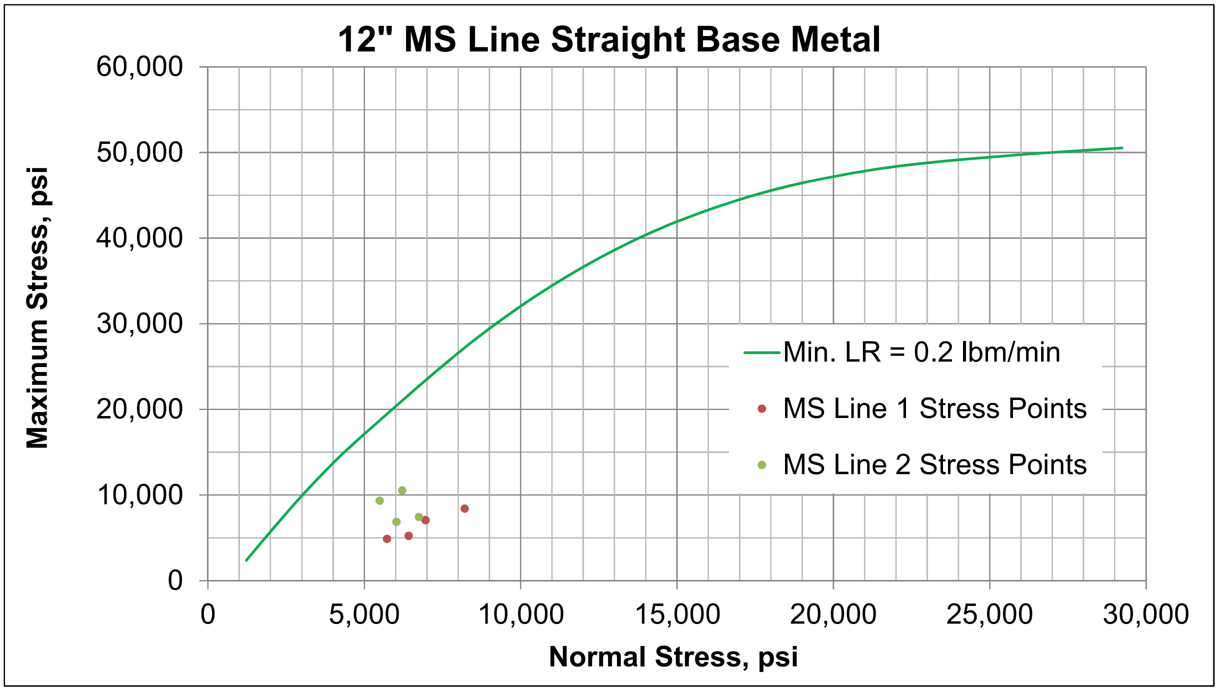

Figure 3.6-27: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 12 Pipe-to-Safe-End Weld

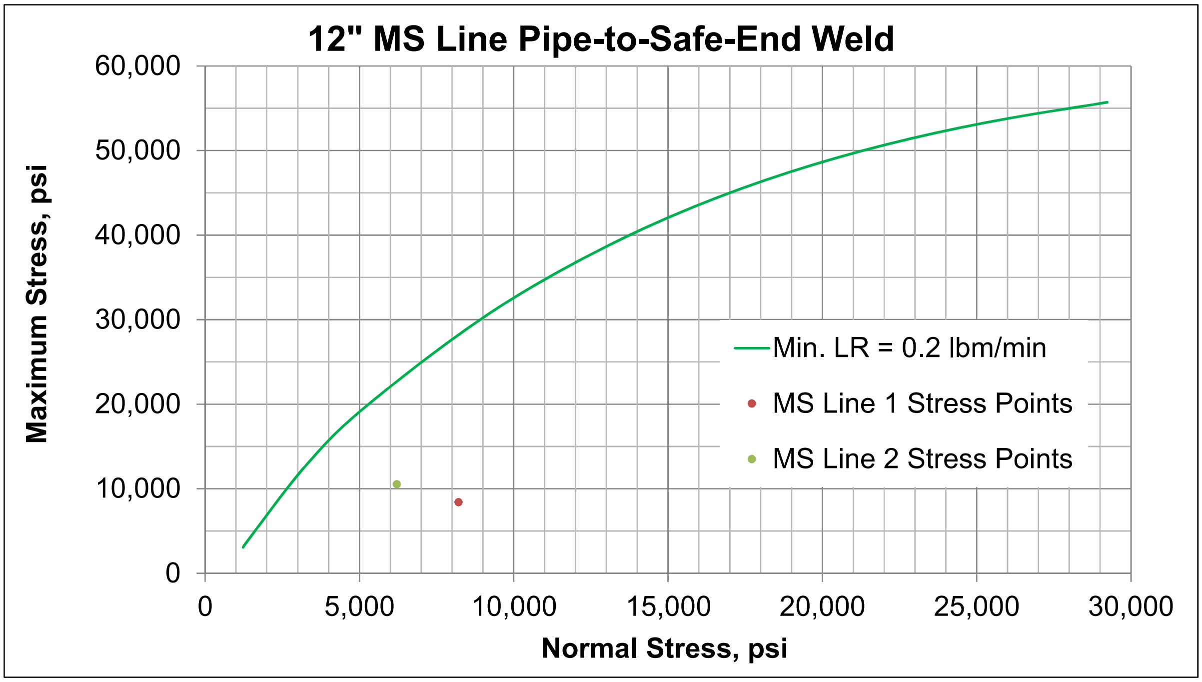

Figure 3.6-28: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 8 Elbow Base Metal

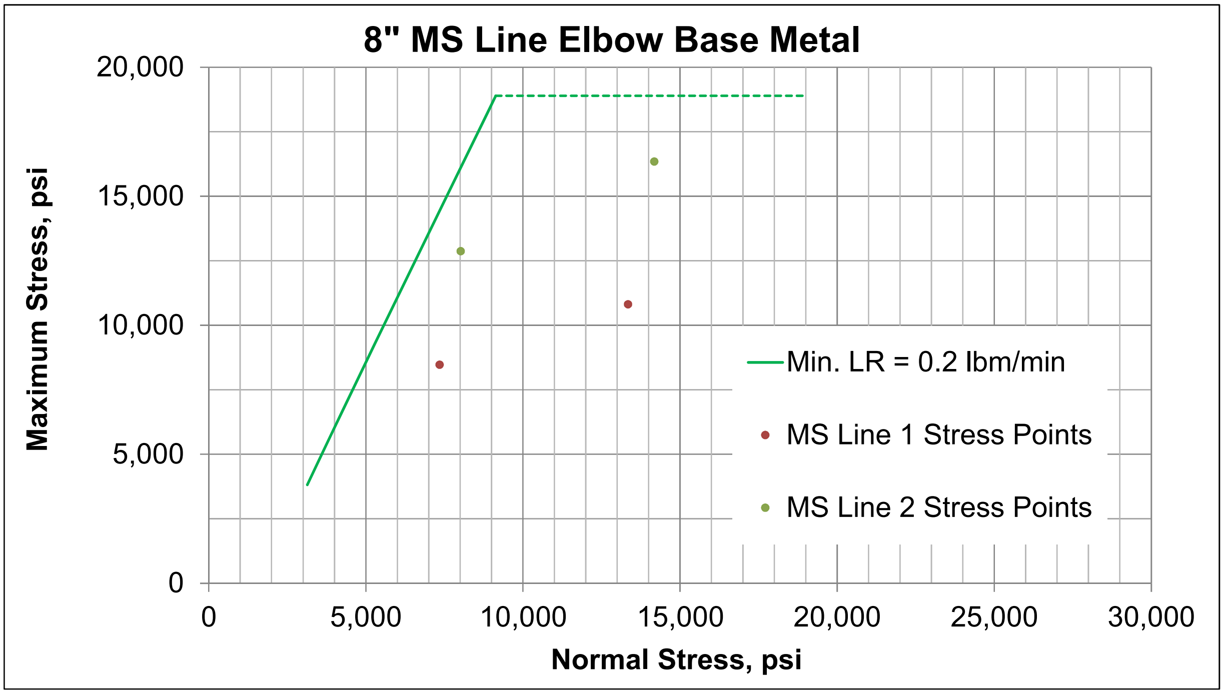

Figure 3.6-29: Smooth Bounding Analysis Curve for Nominal Pipe Size 4 Feedwater System Line Base Metal

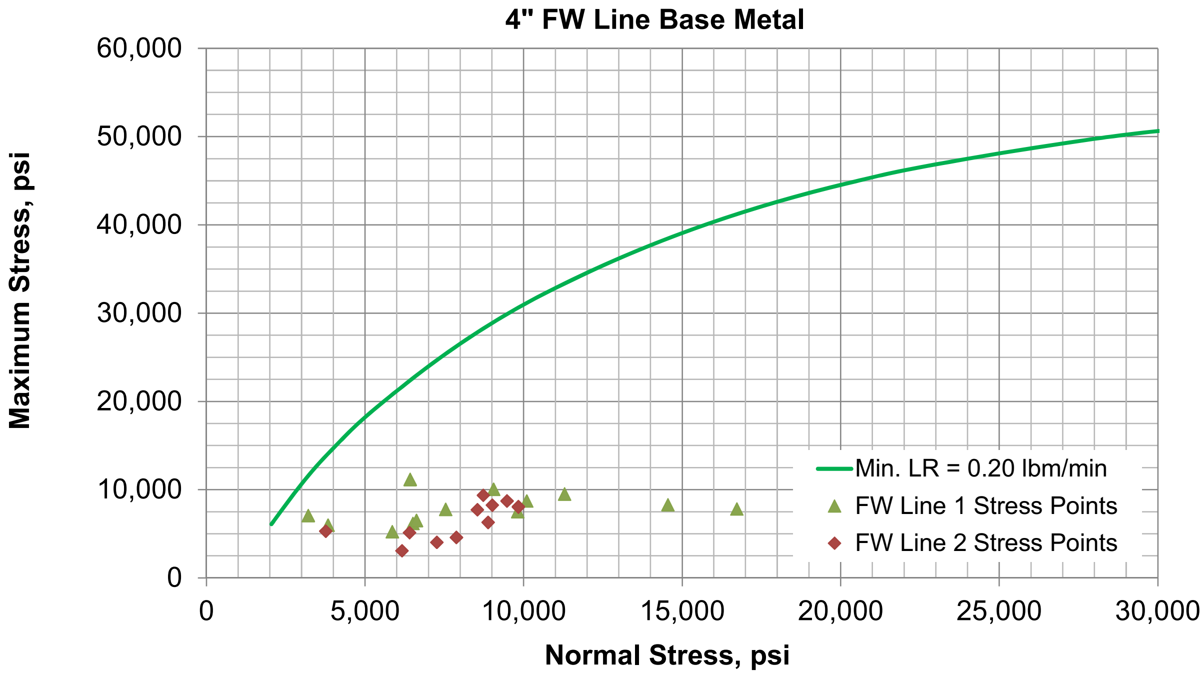

Figure 3.6-30: Smooth Bounding Analysis Curve for Nominal Pipe Size 4 Feedwater System Line Welds

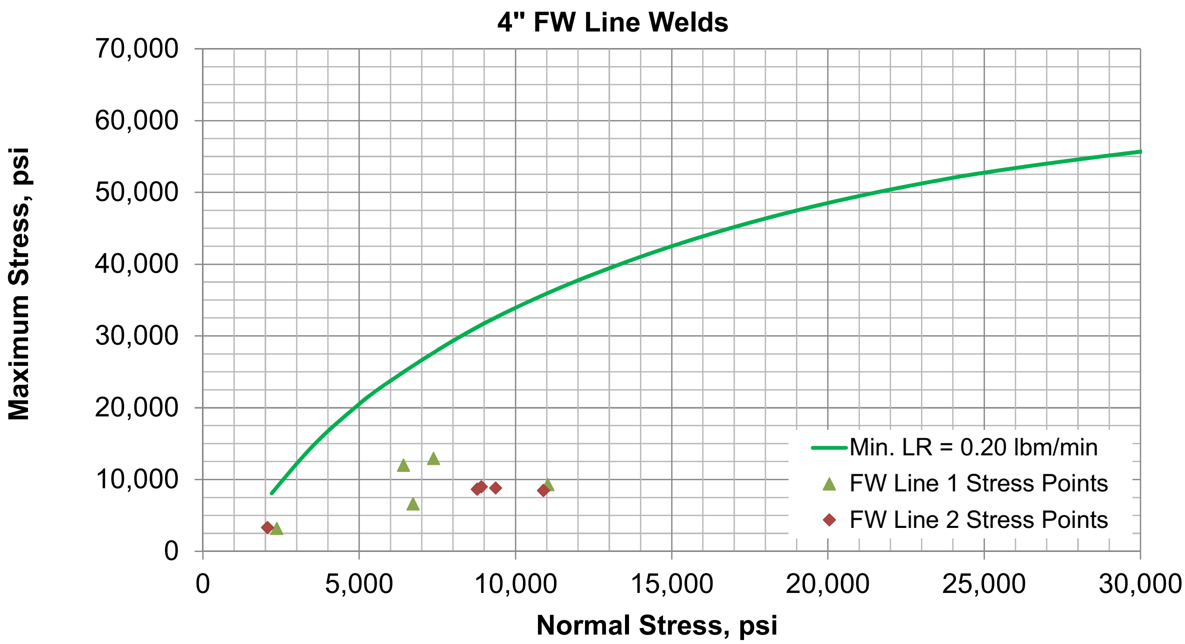

Figure 3.6-31: Smooth Bounding Analysis Curve for Nominal Pipe Size 5 Feedwater System Line Base Metal

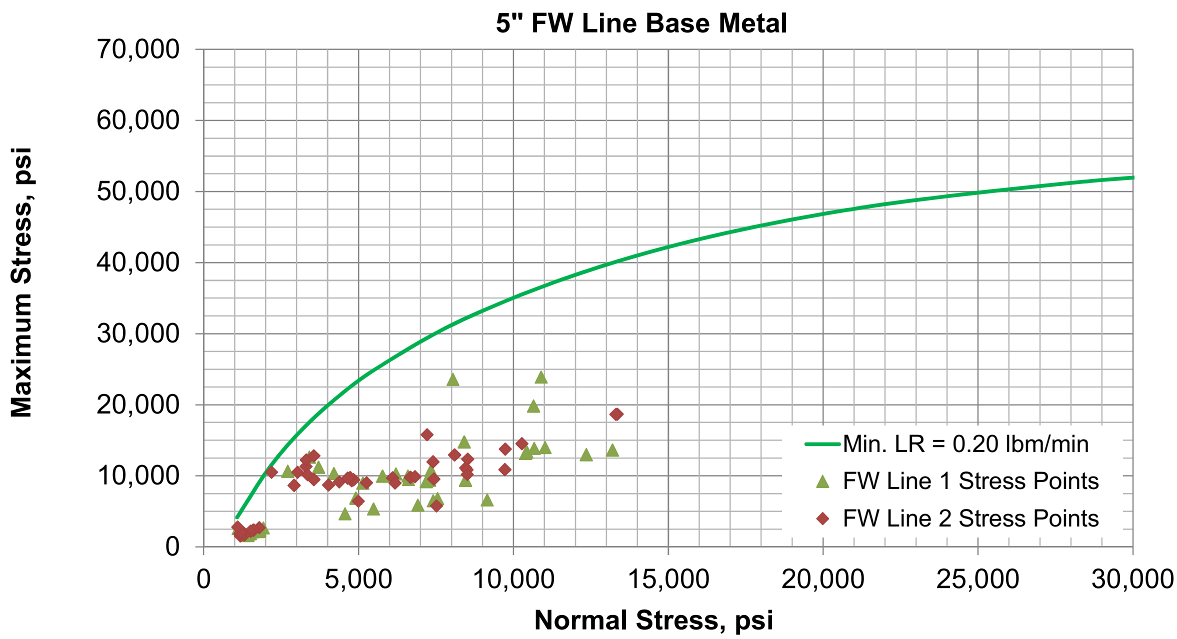

Figure 3.6-32: Smooth Bounding Analysis Curve for Nominal Pipe Size 5 Feedwater System Line Welds

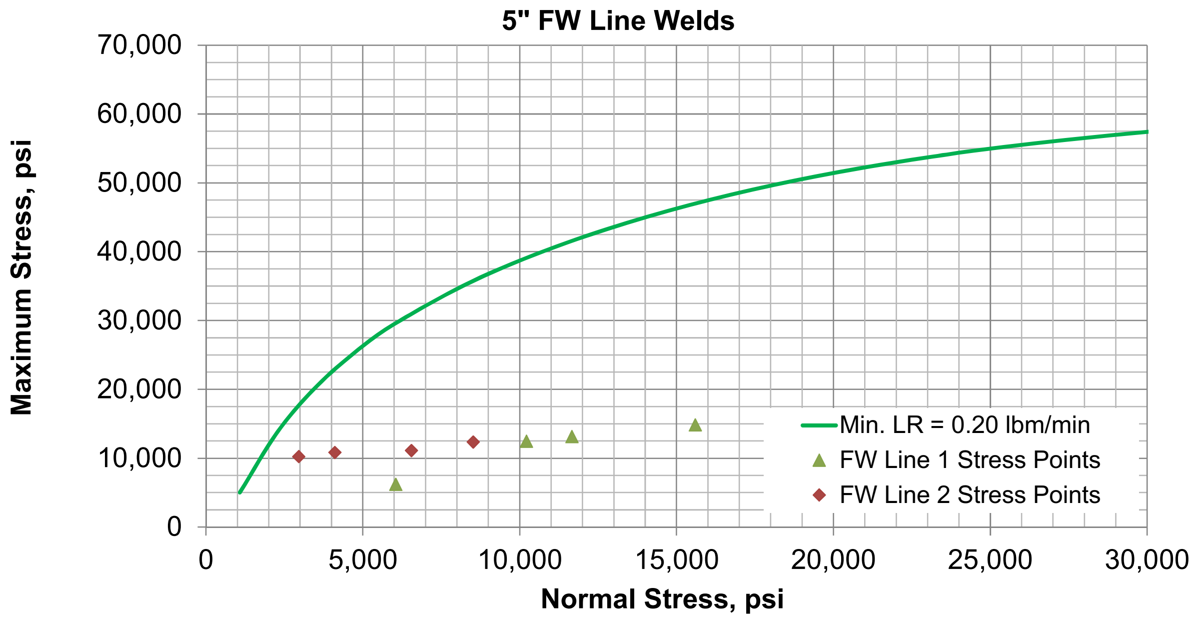

Figure 3.6-33: Application of BTP 3-4 Break Location Guidance in the NPM bay and RXB

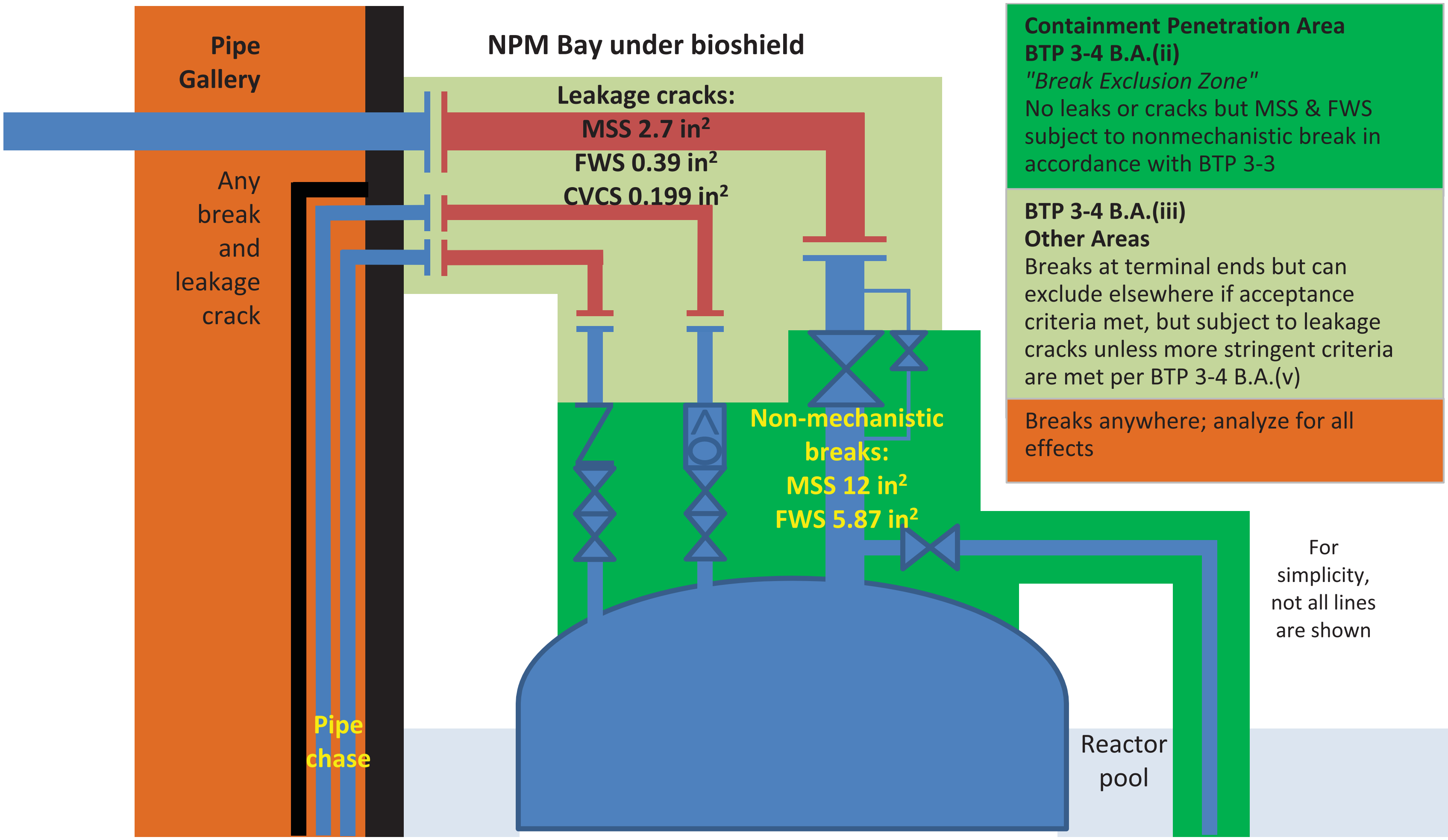

Figure 3.6-34: Not Used

Figure 3.6-35: Not Used

Figure 3.6-36: Not Used

Figure 3.6-37: Not Used

Figure 3.6-38: Not Used

Figure 3.6-39: Not Used

Figure 3.6-40: Not Used