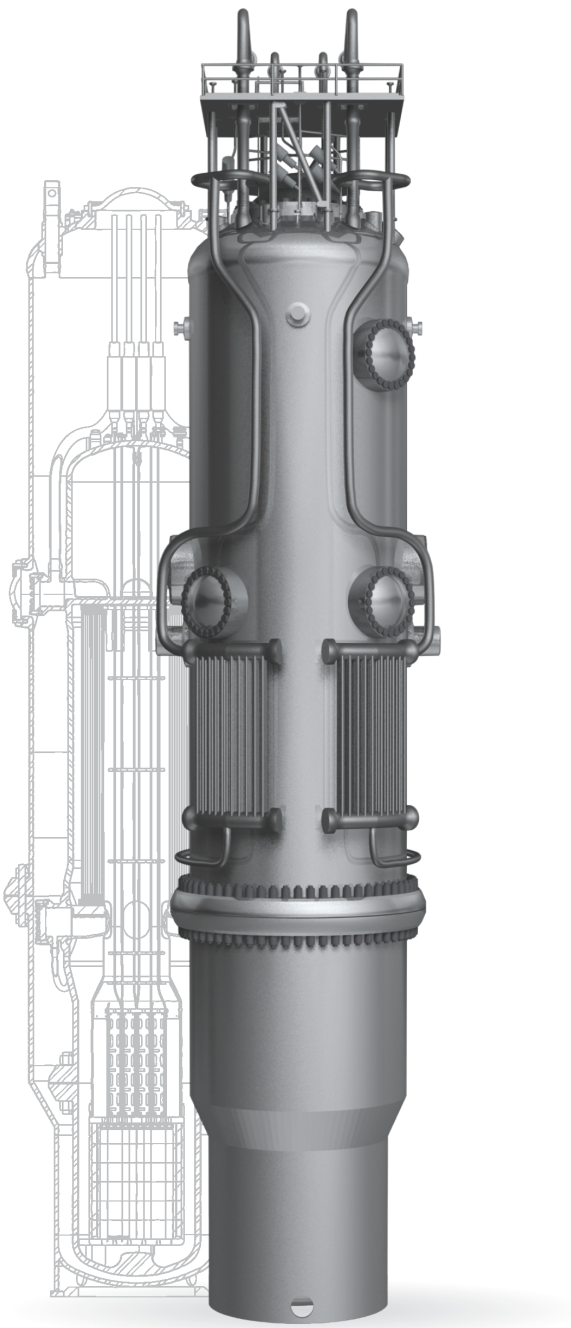


NuScale Standard Plant
Design Certification Application

Chapter Three

Design of Structures, Systems, Components and Equipment

PART 2 - TIER 2

Revision 0
December 2016

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

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

For further discussion, see the following 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

For further discussion, see the following 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

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

For further discussion, see the following 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

For further discussion, see the following 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

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

For further discussion, see the following 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 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

For further discussion, see the following chapters:

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

For further discussion, see the following section:

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

For further discussions, see the following 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

For further discussion, see the following chapters:

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 ASME B&PV Code, Section III Division 1, Subsection NB (Reference 3.1-2) and the materials selected for fabrication of the RCPB meet the ASME B&PV Code, 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

For further discussion, see the following 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 offsite 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

For further discussion, see the following 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

For further discussion, see the following 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

For further discussion, see the following chapters:

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

For further discussion, see the following chapter:

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 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 Power Plant design includes a remote shutdown station which has the necessary instrumentation and controls to maintain the NPM in a safe condition during hot shutdown and to bring the NPM to safe shutdown.

Conformance or Exception

The NuScale Power Plant design conforms to GDC 19.

Relevant FSAR Chapters and Sections

For further discussion, see the following 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

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

For further discussion, see the following chapters:

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

For further discussion, see the following chapter and section:

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

For further discussion, see the following section:

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

For further discussion, see the following chapter 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

For further discussion, see the following chapter:

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

For further discussion, see the following 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

For further discussion, see the following chapter 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, without margin for stuck rods, provided the probability for a return to power assuming a stuck rod is sufficiently small and specified acceptable fuel design limits for critical heat flux would not be exceeded by the return to power.

The CVCS, with boron addition, and CRDS are designed for a combined capability of controlling reactivity changes that assures the capability to cool the core under postulated accident conditions with margin for stuck rods as explained in Section 4.3.1.5. Conservative analysis indicates that a return to power could occur following a reactor trip under the condition that the highest worth CRA does not insert, coincident with the CVCS being unavailable. Consequently, the GDC is modified for the NuScale design to address the shutdown capability for postulated accidents.

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

For further discussion, see the following chapter 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

For further discussion, see the following 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 a ASME B&PV Code, 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

For further discussion, see the following chapter 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

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 B&PV Code, 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

For further discussion, see the following 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

For further discussion, see the following 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 B&PV Code, Section III Division 1, Subsection NB Class 1 pressure-retaining components in accordance with ASME B&PV Code, Section XI Division 1 (Reference 3.1-5) pursuant to 10 CFR 50.55(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

For further discussion, see the following 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

For further discussion, see the following chapter 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 power provisions of GDC 34 are not applicable to the NuScale design. 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

For further discussion, see the following 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

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 power provisions of GDC 35 are not applicable to the NuScale design. 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

For further discussion, see the following chapter 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

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 B&PV Code, 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

For further discussion, see the following section:

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

For further discussion, see the following 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 power provisions of GDC 38 are not applicable to the NuScale design. 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

For further discussion, see the following chapter 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 B&PV Code, 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

For further discussion, see the following section:

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

For further discussion, see the following 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 power provisions of GDC 41 are not applicable to the NuScale design. 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

For further discussion, see the following chapter 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

For further discussion, see the following section:

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

For further discussion, see the following section:

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 power provisions of GDC 44 are not applicable to the NuScale design. 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

For further discussion, see the following 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

For further discussion, see the following section:

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

For further discussion, see the following section:

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

For further discussion, see the following sections:

Section 3.8.2 Steel Containment

Section 6.2 Containment Systems

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 B&PV Code 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

For further discussion, see the following section:

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

For further discussion, see the following section:

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 B&PV Code, 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

For further discussion, see the following 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

For further discussion, see the following 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 B&PV Code, 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

For further discussion, see the following section:

Section 6.2.4 Containment Isolation System

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 B&PV Code 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

For further discussion, see the following section:

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 B&PV Code, 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 B&PV Code, 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 B&PV Code, 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

For further discussion, see the following 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

For further discussion, see the following chapter 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

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

For further discussion, see the following 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

For further discussion, see the following section:

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

For further discussion, see the following chapter 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

For further discussion, see the following 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, 2013 edition, Section III Divison 1, Subsection NB, “Class 1 Components,” New York, NY.

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

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

3.1-5 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section XI Division 1, "Rules for Inservice Inspection of Nuclear Components," 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 are classified as A1
- 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 non-safety 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/or risk-significance of the system function(s) they support. A system that primarily performs safety-related and/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 non-safety-related SSC where specific regulatory guidance applies (e.g., Regulatory Guide (RG) 1.29). The application of 10 CFR 50, Appendix B to specific non-safety-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 SSC.

3.2.1 Seismic Classification

Seismic classification of SSC is consistent with the guidance of RG 1.29 "Seismic Design Classification" and 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 Regulatory Guide (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.

The piping and instrumentation diagrams indicate the boundaries between different seismic categories. A list of piping and instrumentation diagrams is provided in Section 1.7.

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.

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.

Seismic Category I SSC are subject to the quality assurance program requirements of 10 CFR 50, Appendix B.

3.2.1.2 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, non-safety 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.

3.2.2.1 Quality Group A

Quality Group A applies to pressure-retaining components and their supports 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 B&PV Code (Reference 3.2-1).

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 B&PV Code.

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 B&PV Code.

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 and RG 1.143 are listed in Table 3.2-1.

3.2.3 References

3.2-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components, Division 1, "Metallic Components," 2001 Edition including Addenda through 2003.

3.2-2 American National Standards Institute/Instrument Society of America (ANSI/ISA)-67.02.01-1999, "Nuclear Safety-Related Instrument-Sensing Line Piping and Tubing Standard for Use in Nuclear Power Plants," November 1999.

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 1.189 or RG 1.29)
CNTS, Containment System							
All components (except as listed below)	RXB	A1	N/A	Q	None	A	I
• RXM Lifting Lugs • Top Auxiliary Mechanical Access Structure • Top Auxiliary Mechanical Access Structure Diagonal Lifting Braces	RXB	B1	None	AQ-S	• ANSI/ANS 57.1-1992 • ASME NOG-1 • NUREG-0554	N/A	I
CFDS Piping in containment	RXB	B2	None	AQ-S	None	B	II
Piping from (CES, CFDS, CVCS, FWS, MSS, and RCCWS) CIVs to disconnect flange (outside containment)	RXB	B2	None	AQ-S	None	D	I
Hydraulic Skid for valve reset	RXB	B2	None	None	None	D	III
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 • FWS, Supply to SGs and DHR HXs FWIV • 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
Containment Pressure Transducer (Wide Range)	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	II
• Containment Air Temperature (RTDs) • FW Temperature Transducers	RXB	B2	None	AQ-S	None	N/A	II
SGS, Steam Generator System							
• SG tubes • Feedwater plenums • Steam plenums • SG tube supports	RXB	A1	N/A	Q	None	A	I
• Steam piping inside containment • Feedwater piping inside containment • Feedwater supply nozzles • Main steam supply nozzles • Thermal relief valves	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

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 1.189 or RG 1.29)
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
• Wide Range RCS Pressure Element • Wide Range RCS Cold Leg Temperature Element	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
CVCS, Chemical and Volume Control System							
DWS Supply Isolation Valves	RXB	A1	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 Check Valve • Injection Spoolpiece Drain Valve • Pressurizer Spoolpiece Drain Valve • Reactor Module Removable Spoolpieces • RPV High Point Degasification Isolation Valve • RPV High Point Degasification Spoolpiece Drain Valve • Spray Check Valve	RXB	B2	None	AQ-S	None	C	I
• Hydrogen bottle and distribution assembly including excess flow valve • Pressure Indicating Transmitter for Hydrogen Injection Bottle	RXB	B2	None	AQ-S	None	D	II
• Discharge Pressure Indicating Transmitter for CVC Makeup Pump A and B • Flow Indicating Transmitters for: - CVC Makeup Line - Discharge Line - Injection - LRW Letdown - Pressurizer Spray • Pressure Indicating Transmitters for: - Discharge Line - Injection - LRW Letdown - MHS Intersystem Leakage Detection • Temperature Indicating Transmitter for: - CVC Makeup Line - Discharge Line - Injection - LRW Letdown	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 1.189 or RG 1.29)
• BAS Supply Flow Indicating Transmitter • Continuous Boron Analyzer for Downstream of Makeup Tee • Continuous Boron Analyzer on Purification Line • Discharge Line Radioactivity Transmitter • DWS Supply Flow Indicating Transmitter • Flow Indicating Transmitter for Recirculation Pump Discharge • Pressure Indicating Transmitter Downstream of Startup Heater • Pressure Indicating Transmitter for Recirculation Pump A and B Suction and Discharge • Radioactivity Transmitter • Suction Pressure Indicating Transmitter for CVC Makeup Pump A and B • Temperature Indicating Transmitter for: - RHX Shellside Outlet - RHX Shellside Inlet - RHX Tubeside Outlet	RXB	B2	None	None	None	N/A	III
All other components	RXB	B2	None	None	None	D	III
BAS, Boron Addition System							
All components	RXB	B2	None	None	None	D	III
MHS, Module Heatup System							
All components	RXB	B2	None	None	None	D	III
ECCS, Emergency Core Cooling System							
• Reactor Vent Valve (RVV) • RVV Trip Valve • Reactor Recirculation Valve (RRV) • RRV Trip Valve • Reset Valve	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	A	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
• 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

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 1.189 or RG 1.29)
• 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	B2	None	None	None	N/A	III
• RBV Supply AHUs • RBV Exhaust Fans • RBV Exhaust Filter Units • Hot Lab Exhaust Fan • RBV SFP Exhaust Fan • RBV SFP Charcoal Filter Units	RXB	B2	None	AQ	• RG 1.140 • RG 1.52	N/A	III
Ductwork and Associated Components (Dampers, grilles, etc)	RXB	B2	None	AQ	None	N/A	III
Instrumentation	RXB	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 • RG 1.52 • Table 1 of SRP 11.5	N/A	III
LRWS, Liquid Rad-Waste Management System							
All components (except as listed below)	RWB	B2	None	AQ	None	RW-IIc	III
Radioactivity Indicating Transmitter	RWB	B2	None	AQ	ANSI N42.18-2004	RW-IIc	III
LRW Grab Sample Isolation Valve	RWB	B2	None	None	None	RW-IIc	III
GRWS, Gaseous Rad-Waste Management System							
Radioactivity Indicating Transmitter	RWB	B2	None	AQ	ANSI N42.18-2004	RW-IIc	III
• Charcoal Guard Bed • Charcoal Decay Beds	RWB	B2	None	None	None	RW-IIa	RW-IIa
Charcoal Drying Heater	RWB	B2	None	None	None	N/A	N/A
All other components	Various	B2	None	None	None	RW-IIc	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 1.189 or RG 1.29)
SRWS, Solid Rad-Waste Management System							
• Spent Resin Storage Tanks • Spent Resin transfer pumps	RWB	B2	None	AQ	None	RW-IIa	RW-IIa
Grab Sample Isolation Valve	RWB	B2	None	None	None	RW-IIc	III
All other components	Various	B2	None	AQ	None	RW-IIc	III
RWDS, Radioactive Waste Drain System							
All components	Various	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 • RG 1.52	N/A	III
All other components	RWB	B2	None	None	None	N/A	III
MAE, Module Assembly Equipment							
• Reactor Module Inspection Rack • Reactor Module Upender	RXB	B2	None	AQ-S	None	N/A	II
Reactor Module Import Trolley	RXB	B2	None	None	None	N/A	III
MAEB, Module Assembly Equipment - Bolting							
RPV Support Stand	RXB	A2	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
SFPCS, Spent Fuel Pool Cooling System							
• Pumps • Strainers • Valves - Air operated (PCUS boundary isolation valves)	RXB	B2	None	AQ	ANSI/ANS 57.2-1983 with additions, clarifications, and exceptions of RG 1.13	D	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
• Sample Points • Instrumentation (pressure, temperature, flow, position)	RXB	B2	None	None	None	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 1.189 or RG 1.29)
RPCS, Reactor Pool Cooling System							
• Sample Points - SAM-0001 - SAM-0002 • Valves - Air operated - AOV-0031 - AOV-0033 - AOV-0034 - AOV-0035 • Instrumentation - Position - ZSC-0031 and ZSO-0031 - ZSC-0033 and ZSO-0033 - ZSC-0034 and ZSO-0034 - ZSC-0035 and ZSO-0035	RXB	B2	None	AQ	ANSI/ANS 57.2-1983 with additions, clarifications, and exceptions of RG 1.13	D	III
• Heat Exchangers • Reactor Pool Cooling Pumps • Strainers • Valves (not listed above) - Air operated, Check, Manual, Relief • Instrumentation (not listed above) - Flow, Position, Pressure, Temperature • Orifices	RXB	B2	None	None	None	D	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	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	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	None	D	III
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

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 1.189 or RG 1.29)
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 • RG 1.52	D	III
• Containment Service Air Pressure Valve • Sample Vessel Drain Sampler • Containment Service Air Supply Position Closed and Open Switches • Vacuum Pump Bypass Valve Position Closed and Open Switches	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 Sampling 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
PSS, Process Sampling System							
All components (except as listed below)	RXB	B2	None	None	None	N/A	III
• Reactor coolant discharge sample line isolation valve • Primary sampling system analysis panel	RXB	B2	None	AQ	ANSI N13.1	D	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
• 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

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 1.189 or RG 1.29)
• Primary sampling system sample cooler cooling water chillers • Primary sampling system grab sample panel • Primary sampling system ion chromatography analysis units	RXB	B2	None	AQ	None	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	RXB	B2	None	None	None	D	III
MSS, Main Steam System							
• Start-up Isolation Valves • Steam Taps	RXB, TGB	B2	None	AQ-S	None	D	I
• Secondary Main Steam Isolation Valves • Secondary Main Steam Isolation Bypass Valves	RXB, TGB	B2	None	AQ-S	• Technical Specification Surveillance for operability and in-service 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, TGB	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 • Steam Taps	RXB, 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	B2	None	None	None	N/A	III
• Feedwater Regulating Valve A/B • Feedwater Supply Check Valve	TGB	B2	None	AQ-S	Technical Specification Surveillance for operability and in-service testing.	D	I
Feedwater Regulating Valve Accumulators	TGB	B2	None	AQ	Technical Specification Surveillance for operability and in-service testing.	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 1.189 or RG 1.29)
Feedwater Regulating Valve A/B Limit Switch	TGB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• 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 Storage Tank • Condensate Storage Tank Makeup Level Control Valve • 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	Various	B2	None	None	None	N/A	III
ABS, Auxiliary Boiler System							
High Pressure and Low Pressure Aux Boiler skids	TGB	B2	None	None	No	D	III
Radioactivity Instruments	Various	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
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

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 1.189 or RG 1.29)
TLOSS, Turbine Lube Oil Storage System							
All components	TGB	B2	None	None	None	N/A	III
CPS, Cathodic Protection System							
Cathodic Protection System	Various	B2	None	None	None	N/A	III
CWS, Circulating Water System							
All components	TGB, Yard	B2	None	None	None	D	III
SCWS, Site Cooling Water System							
All components (except as listed below)	Yard	B2	None	None	None	D	III
Letdown line rad monitor	Yard	B2	None	AQ	ANSI N42.18-2004	N/A	III
PWS, Potable Water System							
All components	Various	B2	None	None	None	N/A	III
UWS, Utility Water System							
All components (except as listed below)	Various	B2	None	None	None	N/A	III
Letdown Line Rad Monitor	RWB	B2	None	AQ	ANSI N42.18-2004	N/A	III
DWS, Demineralized Water System							
All components (except as listed below)	Various	B2	None	None	None	D	III
• DWS potentially radioactive loop • Grab sample isolation valves • Grab sample ports • DWS headers - radiation indication instruments	Various	B2	None	AQ	None	D	III
NDS, Nitrogen Distribution System							
All components	Yard, RWB	B2	None	None	None	N/A	III
SAS, Service Air System							
All components	Various	B2	None	None	None	N/A	III
IAS, Instrument and Control Air System							
All components	Various	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	Various	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	B2	None	None	None	N/A	III
FPS, Fire Protection System							
All components	Various	B2	None	None	None	N/A	III
BPDS, BOP Drain System							
All components (except as listed below)	Various	B2	None	AQ	None	D	III
Instrumentation	Various	B2	None	None	None	N/A	III
RBSS, Reactor Building Spray System							
All components	RXB	B2	None	AQ-S	None	N/A	II
EHVS, 13.8 KV and SWYD System							
All components	Various	B2	None	None	None	N/A	III
EMVS, Medium Voltage AC Electrical Distribution System							
All components	Various	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 1.189 or RG 1.29)
ELVS, Low Voltage AC Electrical Distribution System							
B6000 series Motor Control Centers	RXB	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	B2	None	None	None	N/A	III
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	Various	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	Various	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	Various	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, Non-Safety DC Electrical and AC Distribution System							
All components	Various	B2	None	None	None	N/A	III
BPSS, Backup Power Supply System							
All components (except as listed below)	DGB, 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)	Various	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 1.189 or RG 1.29)
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
GLPS, Grounding and Lightning Protection System							
All components	Various	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 1.189 or RG 1.29)
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	B2	None	AQ	IEEE 497-2002 with CORR 1	N/A	III
• Controllers (other than above) • I/O Modules (other than above)	RXB, CRB	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	B2	None	AQ	• Backup diesel powered • Analyzed for seismic qualification	N/A	III
• Controllers for RSS indication • I/O Modules for RSS indication	CRB, RXB, TGB	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	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, TGB	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	RXB, 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 1.189 or RG 1.29)
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	CRB, RXB, TGB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
Radiation monitors that monitors Type E variables	CRB, 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, TGB	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	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	I
RBCM, Reactor Building Components							
• Pool 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	ES-0303-3677	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	D	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
[[SCB, Security Buildings (Guardhouse)]]							
• Security Building • Vehicle inspection sally port	Yard	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 1.189 or RG 1.29)
[[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	Various	B2	None	None	None	N/A	III
SMS, Seismic Monitoring System							
All components	Various	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 pertinent augmented quality assurance requirements for non-safety related SSC's are applied per the augmented requirements identified (i.e. non-safety risk significant SSC's, RG 1.143, IEEE 497, RG 1.189), in accordance with the quality assurance program.
- AQ-S = indicates that the pertinent requirements of 10 CFR 50 Appendix B are applicable to SSC classified as seismic category II in accordance with the quality assurance program.
- None = indicates no specific QA program or augmented quality requirements are applicable.

Note 3: Augmented design requirements such as application of Quality Class or Seismic Classification to non-safety systems are reflected in the associated columns.

Note 4: Quality classification per RG 1.26 is not applied to supports or instrumentation.

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 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 envelope conditions at most 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.

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 0.85. 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 or greater,

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 continental United States as identified in RG 1.76, Rev. 1.

- Maximum wind speed .... 230 mph
- Maximum translational speed .... 46 mph
- Maximum rotational speed .... 184 mph
- Radius of maximum rotational speed ...... 150 ft

- Maximum 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 for the continental United States as identified in Figures 1 - 3 of RG 1.221, Rev. 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants."

- 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 0.85. (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 or greater,

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.

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

A failure of a nearby structure could adversely affect the Seismic Category I RXB and Seismic Category I portions of the CRB. These nearby structures are assessed (or analyzed if necessary) as described below to ensure that there is no credible potential for adverse interactions. 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 can withstand the severe and extreme winds.

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.

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 RXB or Seismic Category I portion of the CRB.

3.3.4 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, 2005.

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 a truck ramp on the west side of the Radwaste Building and the CRB tunnel.

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 identify the selected mitigation strategy for each room containing structures, systems, and components subject to flood protection.

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.
- The finished grade for all building structures, except at a truck ramp on the west side of the Radwaste Building and CRB tunnel, is approximately six inches below the baseline plant elevation. The yard is graded with a minimum slope of 1.5% 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.

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 discussed in Sections 3.7.1 and 3.8.4.

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.

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 RXB or Seismic Category I portion of the CRB.

3.4.3 References

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

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

3.4-3 International Building Code, Section 1805, “Dampproofing and Waterproofing,” International Code Council, 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 codes and standards are not considered credible missiles. Bolted bonnets are prevented from becoming missiles by limiting stresses in the bonnet-to-body bolting material and by designing flanges in accordance with applicable code requirements. Pressure-seal bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in shear, and by the yoke capturing the bonnet or reducing bonnet energy.

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 reactor 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. Safety related and risk significant SSCs for the design are located principally within the RXB and CRB. The turbine generator rotor shafts are physically oriented such that the RXB and CRB are [[within]] the turbine low-trajectory hazard zone and considered to be [[unfavorably]] oriented with respect to the NPMs, as defined by RG 1.115, Revision 2. Safety-related and risk-significant SSCs within the reactor and control building are protected from the effects of turbine missiles by limiting the generation of missiles from the turbine generators to be less than 10-5 consistent with Table 1 of RG 1.115.

COL Item 3.5-1: A COL applicant that references the NuScale Power Plant certified design will provide a missile analysis for the turbine generator which demonstrates that the probability of a turbine generator producing a low trajectory turbine missile is less than 10^{-5}.

Section 10.2 describes the turbine generator requirements for turbine rotor integrity, including rotor material fracture toughness, overspeed protection, and inspection and testing. The turbine rotor inspection program along with the low probability of turbine missile generation provide assurance that safety related and risk significant SSCs are protected from the adverse effects of turbine missiles, consistent with GDC 4.

COL Item 3.5-2: A COL applicant that references the NuScale Power Plant certified design 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 walls, roofs, and openings are designed to withstand the design basis missiles discussed in Section 3.5.1.4. Section 3.8 provides additional information for the design of RXB and CRB.

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 not been credited to withstand turbine missiles.

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 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%.

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 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.

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:

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

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

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

Several empirical equations have been developed to estimate the penetration of missiles through steel barriers. The Stanford Formula (Reference 3.5-4) is used to determine the minimum steel thickness for a barrier to prevent perforation of the missile through the barrier. The Ballistic Research Laboratory Formula (Reference 3.5-5) equations may also be used if the results are comparable to the Stanford Formula results or if they are validated by penetration testing.

Stanford Formula for Penetration

The Stanford Formula calculates the energy, E, needed to perforate a steel plate of thickness, T.

$$\frac{E}{D} = \frac{S}{46{,}500}\left(16{,}000T^2 + 1{,}500\frac{W}{W_s}T\right)$$

where,

E = critical kinetic energy required for perforation, (ft-lb),

D = effective missile diameter, in.,

S = ultimate tensile strength of the target (steel), (psi),

T = Target plate thickness, in.,

W = length of square side between rigid supports, in., shall be no greater than 8D, and

W_s = length of standard window, 4 in.

The ultimate tensile strength (S) is directly reduced by the amount of bilateral tension stress already in the target. The Stanford equation is applicable within the following range of parameters:

$0.1 < T/D < 0.8$

$0.002 < T/L < 0.05$

$10 < L/D < 50$

$5 < W/D < 8$

$8 < W/T < 100$

$0.2 < W/T < 1.0$

$70 < V < 400$

where,

L = Missile length, in., and

V = impact velocity, ft/sec.

Ballistic Research Laboratory Formula for Penetration

The Ballistic Research Laboratory formula is used to calculate the thickness of steel plate that would be penetrated by a missile of known mass, velocity and equivalent diameter.

$$t_p = \frac{(E_k)^{2/3}}{672D}$$

where,

t_p = steel plate thickness for threshold of perforation, in.,

D = equivalent missile diameter, in.,

E_k = missile kinetic energy, foot-pounds

$E_k = M\,V^2/2$,

where,

M = mass of the missile, lb-sec^2/ft, and

V = impact velocity, ft/sec.

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.

The 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.

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, $\mu \leq \dfrac{0.25\varepsilon_{\mu}}{\varepsilon_y} \leq \dfrac{0.1}{\varepsilon_y}$

   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, Section III, "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).

3.5-2 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, "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).

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 (Amsterdam), V. 37, No. 2, 1976, pp. 183-203.

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

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

3.5-6 American Concrete Institute, ACI 349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," 2006.

3.5-7 American Institute of Steel Construction, AISC N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities," 2006.

3.5-8 Bechtel Topical Report, BC-TOP-9A, Rev. 2, "Design of Structures for Missile Impact", by R.B. Linderman, J.V. Rotz, G.C.K. Yeh, September 1974.

3.5-9 Stephenson, A.E., "Full Scale Tornado Missile Impact Tests", EPRI NP440, Sandia Laboratories, Tonopa, NV, prepared for Electric Power Inst., Palo Alto, CA July 1977.

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

<table>
<tr><th>Direction</th><th>Missile</th><th>N</th><th>W (lbs)</th><th>D (in.)</th><th>V (ft/s)</th><th>Concrete Strength (psi)</th><th>Penetration Distance (in.)</th><th>Perforation Distance (in.)</th><th>Thickness to Preclude Scabbing</th><th>Building</th><th>Wall/Roof Thickness (in.)</th></tr>
<tr><td rowspan="6">horizontal</td><td rowspan="3">pipe</td><td rowspan="3">0.84</td><td rowspan="3">287</td><td rowspan="3">6.625</td><td rowspan="3">251</td><td>7000</td><td>6.2</td><td>18.6</td><td>23.7</td><td>RXB</td><td>60</td></tr>
<tr><td rowspan="2">5000</td><td rowspan="2">6.7</td><td rowspan="2">19.8</td><td rowspan="2">27.8 from EC-F170-3650, Rev 1</td><td>CRB</td><td>36</td></tr>
<tr><td>RWB</td><td>24</td></tr>
<tr><td rowspan="3">sphere</td><td rowspan="3">1.00</td><td rowspan="3">0.147</td><td rowspan="3">1</td><td rowspan="3">224</td><td>7000</td><td>0.3</td><td>1.1</td><td>2.3</td><td>RXB</td><td>60</td></tr>
<tr><td rowspan="2">5000</td><td rowspan="2">0.3</td><td rowspan="2">1.1</td><td rowspan="2">2.4</td><td>CRB</td><td>36</td></tr>
<tr><td>RWB</td><td>24</td></tr>
<tr><td rowspan="6">vertical</td><td rowspan="3">pipe</td><td rowspan="3">0.84</td><td rowspan="3">287</td><td rowspan="3">6.625</td><td rowspan="3">91</td><td rowspan="3">5000</td><td rowspan="3">2.7</td><td rowspan="3">9.4</td><td rowspan="3">18.9</td><td>RXB</td><td>48</td></tr>
<tr><td>CRB</td><td>36</td></tr>
<tr><td>RWB</td><td>12</td></tr>
<tr><td rowspan="3">sphere</td><td rowspan="3">1.00</td><td rowspan="3">0.147</td><td rowspan="3">1</td><td rowspan="3">85</td><td rowspan="3">5000</td><td rowspan="3">0.1</td><td rowspan="3">0.5</td><td rowspan="3">1.2</td><td>RXB</td><td>48</td></tr>
<tr><td>CRB</td><td>36</td></tr>
<tr><td>RWB</td><td>12</td></tr>
</table>

3.6 Protection against Dynamic Effects Associated with Postulated Rupture of Piping

This section describes the design bases and measures needed to protect the reactor pressure vessel (RPV) and other essential systems and components inside or outside containment, including components of the reactor coolant pressure boundary, against the effects of pressurization, pipe rupture including jet impingement, pipe whip, and subcompartment pressurization resulting from a postulated rupture of piping located either inside or outside containment.

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 essential structures, systems, and components (SSC) are not impacted by the adverse effects of postulated piping rupture. 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.

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, and applicable Branch Technical Positions (BTPs).

Section 3.6.1 identifies the high- and moderate-energy lines that have a potential to affect essential SSC, and describes the approaches used in the NuScale Power Plant design for protection of essential SSC. Section 3.6.2 provides the analytical methodology used to determine break locations and identifies the postulated breaks. 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. Finally, Section 3.6.5 describes the mitigation approaches used for postulated break locations if the dynamic consequences of the break cannot be tolerated.

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. This includes dynamic effects (pressurization, pipe whip and jet impingement) that may result from equipment failures and from events and conditions outside the nuclear power module.

Plant designers are provided with options to address GDC 4 for high-energy line break (HELB) considerations. These options are as follows:

- On a limited basis, portions of pipe may be excluded from HELB considerations 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 leak-before-break (LBB) analysis and is discussed in SRP 3.6.3. Leak-before-break 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.
- High- and moderate-energy pipe systems that cannot be fully excluded using either the exclusion criteria of BTP 3-4, Section B.A.(ii) or LBB must be designed for HELB. The criteria for the specific locations for the postulated breaks are provided in BTP 3-4. 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 postulated rupture locations, the consequences of HELB can include pipe whip or jet impingement, both of which can potentially damage safety-related equipment required for safe shutdown. At break locations, the pipe must either be located such that there is no safety-related equipment in the area, or the pipe must be restrained from whip, and safety-related equipment protected from jet impingement, as needed.

The piping systems that must be considered include the Class 1, 2, 3, and 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.

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.

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

The shutdown of the reactor requires the following systems be protected from HELB:

- the integrity of the containment isolation valves (CIVs) and decay heat removal Class 2 piping systems and condensers (outside of the CNV), including non-safety components credited in safety analyses
- the emergency core cooling system (ECCS) valves
- the decay heat removal system (DHRS) inside the CNV
- the module protection system, including associated instruments, cables, and other components

Figure 3.6-1 shows the lines that interface with the CNV.

The main steam and feedwater lines inside containment are part of the steam generator system in the NuScale system designation scheme. For the purpose of HELB analysis, these lines are referred to in relation to their process fluid system, as main steam system (MSS) and feedwater system (FWS) regardless of whether inside or outside of the CNV. The MSS and FWS are high-energy systems. The same practice is used for other systems that penetrate containment. The chemical and volume control system (CVCS) injection, discharge, pressurizer spray and high point vent lines are part of the RCS inside the CNV. The CVCS is a high-energy system. These lines are identified as the RCS injection, RCS discharge, pressurizer spray, and high point vent lines, or collectively as CVCS. The control rod drive system (CRDS) piping inside containment is functionally part of the moderate-energy reactor component cooling water system (RCCWS).

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

The containment flooding and drain system (CFDS) is a single open pipe inside containment. This line is moderate-energy based on the amount of time in use. This line is identified as the containment system (CNTS) flooding and drain line both inside and outside the CNV.

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 SSC are consolidated into six groups. Each is discussed in a separate section.

- inside the CNV (Section 3.6.1.1.1)
- outside the CNV (to the disconnect flange) (Section 3.6.1.1.2)
- in the Reactor Building (RXB), (outside the NPM disconnect flange) (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 evaluates 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 comprehensive 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.

3.6.1.1.1 High- and Moderate-Energy Lines Inside the Containment Vessel

There are ten high-energy lines inside the CNV: two main steam, two feedwater, RCS injection, RCS discharge, high point vent, pressurizer spray, and two DHRS condensate return. There are two moderate-energy lines inside the CNV, the CRDS cooling loop and CFDS (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 High- and Moderate-Energy Lines Outside the Containment Vessel (to the NuScale Power Module Disconnect Flange)

The ten high-energy lines and two moderate-energy lines discussed in Section 3.6.1.1.1 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. Although not normally in use, this entire system is pressurized during NPM operation.

3.6.1.1.3 High- and Moderate-Energy Lines in the Reactor Building (outside the NuScale Power Module Disconnect Flange)

Within the RXB, but outside the NPM disconnect flange, 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 the nominal diameter of the PSS lines, breaks do not need to be postulated in the PSS lines.

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

Outside of the reactor pool bay, 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". A break in any of these lines would only impact the function of the CVCS equipment for that module. The pipe chase can be seen on the general arrangement drawings in Section 1.2.

The high-energy auxiliary boiler lines are routed to the module heatup heat exchangers in the CVCS rooms and to various service locations in the RXB.

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

3.6.1.1.4 High- and Moderate-Energy Lines 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 High- and Moderate-Energy Lines 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 High-Energy and Moderate-Energy Lines Outside the Reactor Building and Control Building

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

There is no 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 determine if a high-energy line break or moderate energy line break outside of the Reactor Building, Control Building, or Radioactive Waste Building could affect site-specific essential equipment (or result in a transient or other off-normal event in a second module), and install protection as necessary.

3.6.1.2 Types of Breaks

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 pipe whip, jet thrust reaction, jet impingement (dynamic effects), flooding, spray wetting, and increased temperature, pressure, and humidity (environmental effects). Through-wall leakage cracks are as defined in BTP 3-4, Revision 2, and are analyzed for localized flooding and environmental effects. For evaluation of spray wetting, flooding, and subcompartment pressurization effects, longitudinal breaks (with break flow areas equal to the piping flow area) are postulated in the main steam and feedwater piping outside the CNV. The dynamic effects of pipe whip and jet impingement are not evaluated for these breaks. 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 3.A.

Moderate-energy lines are evaluated for through-wall leakage cracks and analyzed for flooding and environmental effects. The environmental effects of postulated moderate-energy leakage cracks are less severe than the inside containment and outside containment under the bioshield environmental conditions associated with anticipated operational occurrences.

3.6.1.3 Protection Methods

Inside the CNV and reactor pool bay, including piping up to the pool wall, there is generally insufficient space to rely on distance (i.e., separation) or installation of traditional pipe whip restraints or jet shielding. Therefore, in these areas, the primary method employed by the NuScale design to mitigate the dynamic effects of pipe rupture is the integral shield restraint (ISR). This component is to be placed at all break locations identified in Figure 3.6-2 through Figure 3.6-15 and is discussed in more detail in Section 3.6.5. Currently, the ISR has been designed to be compatible with NPS 2 piping as all of the identified break locations are on NPS 2 piping. As the piping analysis is finalized other protection methods may be employed to protect against pipe whip and jet impingement, which may include equipment shields, barriers, and pipe whip restraints utilizing energy-absorbing structures. Pipe whip and jet protection methods other than the ISR are developed when postulated breaks are identified that cannot utilize an ISR.

Outside of the reactor pool bay, protection is generally provided by separation in that there are a limited number of locations where safety-related or risk-significant equipment is co-located with high-energy lines. In those locations where they are in proximity to one another, an assessment is made to determine if the safety-related or risk-significant function is required to mitigate the consequences of the rupture. In general, the equipment is associated with the function provided by the line experiencing the break thus does not need to be protected, since the functionality is lost due to the break itself. If the dynamic effects of a break outside a reactor pool bay adversely affect safety-related or risk-significant systems or components, or could cause a transient in a second NPM, an ISR is installed or other conventional methods are used for shielding/restraint on the line with the postulated break.

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, and the methodology used to define the thrust at the postulated break location and the jet impingement loading on adjacent essential safety-related SSC. Pipe breaks on the MSS and FWS inside containment are replaced by small leakage cracks when the LBB criteria are applied (See Section 3.6.3). Jet impingement and pipe whip effects are not evaluated for these small leakage cracks.

GDC 4 requires that SSC 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 by the adverse effects of postulated piping rupture, including pipe whip and jet impingement. 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 also 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 nominal diameter; longitudinal breaks in fluid system piping that is 4-inch nominal diameter and greater, and leakage cracks in fluid system piping greater than 1-inch nominal diameter. Leakage cracks are also postulated in moderate-energy lines.

The GDC 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. LBB is applied to the main steam and feedwater lines inside containment. This is discussed in Section 3.6.3.

3.6.2.1.1 Pipe Breaks Inside the Containment Vessel

Due to the tight configuration and the concentration of safety-related and risk-significant SSC inside of the CNV, dynamic effects of pipe breaks cannot be tolerated. Therefore, the following strategies are employed for HELBs 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. See Figure 3.6-2, Figure 3.6-3, Figure 3.6-4 and Figure 3.6-5 for simplified drawings of the MSS and FWS high-energy piping inside containment.

The CVCS RCS injection, RCS discharge, pressurizer spray, and high point vent lines inside containment are NPS 2, Schedule 160, ASME Class 1 stainless steel pipes. Due to their size, longitudinal breaks are not postulated. See Figure 3.6-6, Figure 3.6-7, Figure 3.6-8 and Figure 3.6-9 for simplified drawings of the CVCS high-energy RCS injection, RCS discharge, pressurizer spray, and high point vent lines, respectively, inside containment with postulated break locations indicated. 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 - The extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors, which act as rigid constraints to piping motion and thermal expansion

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 usage factor exceeds 0.4

The decay heat removal line inside containment is a NPS 2 ASME Class 2 line. Due to its 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 - The extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors, which act as rigid constraints to piping motion and thermal expansion

b) at intermediate locations selected by one of the following criteria:

   i) at each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Or, where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.

   ii) at each location 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-3653.

The DHRS condensate line inside containment runs from each feedwater line, just upstream of the feed plenum, to the containment upper cylindrical shell penetration. Breaks are postulated at the terminal ends as there are no intermediate welds in the piping run. See Figure 3.6-10 and Figure 3.6-11 for simplified drawings of the DHRS #1 and #2 high-energy lines inside and outside containment with postulated break locations indicated. The breaks are listed in Table 3.6-2.

The CRDS 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 for the CNV. As a result, leakage cracks are not postulated further for the CRDS and CFDS lines inside containment.

Final stress analysis will be performed concurrent with fabrication of the first NPM. The postulated break locations based upon the current analysis are listed in Table 3.6-2, and shown in Figure 3.6-6, Figure 3.6-7, Figure 3.6-8, Figure 3.6-9, Figure 3.6-10 and Figure 3.6-11.

ITAAC A07, Pipe Break Hazards Protective Features Verification, was established to confirm the installation of protective features to mitigate 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 in the Reactor Pool Bay (Outside Containment)

The containment isolation valves for the CVCS RCS injection, RCS discharge, pressurizer spray, high point vent line and the two feedwater lines are welded directly to Alloy 690 safe-ends that are welded to the respective nozzles on the CNV head. For the weld between the valve and the safe-end, the provisions of BTP 3-4 Section B.A.(ii) have been applied to preclude the need for breaks to be postulated.

In accordance with BTP 3-4 Section B.A.(ii), breaks are not postulated in this piping 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 reactor coolant system lines) from the CNV head to the first isolation valve 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 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 Feedwater piping from containment to the first isolation valve does not exceed the following design 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 only one circumferential, and no longitudinal welds.

4) The length of the piping is the minimum practical.

5) There are no pipe anchors or restraints.

6) Guard pipes are not used.

7) The welds are included in the ISI program as described in Section 6.6.

Outboard of the containment isolation valves, the CVCS NPS 2, Schedule 160 RCS discharge, RCS injection, pressurizer spray, and high point vent 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. Breaks in these lines are postulated in accordance with BTP 3-4 Section B.A.(iii)(2) at the following locations:

a) At terminal ends

b) At intermediate locations selected by one of the following criteria:

   i) At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Or, where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.

   ii) At each location where stresses are 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 exceed 0.8 times the sum of the stress limits given in NC/ND-3653.

Final stress analysis will be performed concurrent with fabrication of the first NPM. The postulated break locations based upon the current analysis are listed in Table 3.6-2, and shown in Figure 3.6-12 for the RCS injection line, Figure 3.6-12 for the RCS discharge line, Figure 3.6-13 for the pressurizer spray line, Figure 3.6-13 for the high point vent line, Figure 3.6-14 for the feedwater lines, and Figure 3.6-15 for the main steam lines.

Due to the unique nature of the DHRS piping and the connections to the main steam line between the CIV and the CNV, these lines are specifically discussed in Section 3.6.2.5. Additionally, for the MS and FW piping, the break exclusion zone continues past the CIVs up to the penetrations at the reactor pool wall. These portions of piping are also discussed in Section 3.6.2.5.

COL Item 3.6-2: A COL applicant that references the NuScale Power Plant design certification will finalize the stress analysis of the high-energy lines in the Reactor Pool Bay, design appropriate protection features, and update Table 3.6-2, Figure 3.6-12, Figure 3.6-13, Figure 3.6-14 and Figure 3.6-15 as appropriate.

3.6.2.1.3 Pipe Breaks in the Reactor Building (outside the Reactor Pool Bay)

ASME Section III piping terminates at the NPM disconnect flanges or at the first valve outboard of the disconnect flange. Within the NPM, there are a large number of essential SSC that require protection and relatively small amounts of piping. Therefore, postulated pipe break locations inboard of the disconnect flanges are generally addressed with installation of an ISR device, as discussed in Section 3.6.1.3.

Beyond the NPM disconnect flanges, 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 include safety-related and risk significant SSC, SSC that are credited in safety-related evaluations, and SSC that must remain operable to prevent the propagation of a more serious event.

It is appropriate, therefore, for locations beyond the NPM disconnect flanges, to identify the target SSC that must be protected against the dynamic effects of postulated breaks, and then to identify vicinity high- and moderate-energy piping systems, determine the postulated rupture locations in those systems, and determine if protection is required.

As fluid jets have the potential to impact SSC further away than pipe whip, a conservative approach is to evaluate ruptures of high- or moderate-energy piping located within 25 pipe diameters of the target SSC (Appendix A of SRP 3.6.2, Revision 3 Draft).

Pipe routing for the balance-of-plant (BOP) (beyond the NPM disconnect flanges) is finalized with fabrication of the first NPM. The postulated BOP pipe routing is shown in Figure 3.6-17 for the large-bore feedwater lines, and Figure 3.6-16 for the large bore main steam lines.

Similarly, the NuScale pipe rupture hazards analysis is completed for BOP high- and moderate-energy piping with finalization of the pipe routing outboard of the NPM disconnect flanges. The NuScale pipe rupture hazards analysis for BOP piping provides a summary of the analyses applicable to high- and moderate-energy pipe breaks, including:

- identification of the high- and moderate-energy BOP piping systems, including line sizes, and location within the plant.
- a list of target SSC that require protection based on systems identified as required to achieve safe shutdown. Additionally, SSC that provide for the continued safe operation of other NPMs are evaluated to ensure the postulated pipe rupture does not generate a more serious plant condition by initiating an operational occurrence or accident in another NPM.
- at each location where a target SSC that requires protection is located, break locations and break types are postulated in the vicinity of high- and moderate-energy piping in accordance with Section 3.6.2.1.2.
- sketches showing the locations of the postulated pipe ruptures, including identification of longitudinal and circumferential breaks and vicinity essential SSC.
- a summary of the data developed to select postulated break locations, including, as applicable, at intermediate locations selected by one of the following criteria:

- at each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Or, where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.
- at each location where stresses are calculated by the sum of equations. (9) and (10) in NC/ND-3653 of ASME Code, Section III, to exceed 0.8 times the sum of the stress limits given in NC/ND-3653, as delineated in BTP 3-4.

- identification of the protective measures taken to mitigate the effects of postulated pipe failures for each identified essential SSC. Mitigation can include installation protecting features such as pipe whip restraints, equipment shields and ISRs. Evaluation of the unconstrained pipe whip, jet impingement, spray wetting, flooding, and other adverse environmental effects within the zone of influence may also be performed to show that the effects of the rupture are acceptable without mitigation.
- for installed protective features, calculations of the imposed loads and stresses, evaluation of the unmitigated spray loads, zone of influence, installation characteristics, etc.
- an evaluation and disposition of multi-module impacts in common pipe galleries. Pipe break on a high or moderate energy system associated with one NPM should not impair the ability of another NPM to continue normal operation, safe shutdown, maintenance, outage operations, or construction.
- a conclusion that the affected NPM can be safely shut down and maintained in a safe shutdown following a pipe break.

A break in a high-energy MSS, FWS, or auxiliary boiler line in the RXB (outside of the Reactor Pool Bay) could potentially cause breaks or leakage cracks in lines of other modules, introducing an additional transient in a second module. Therefore, break locations are postulated and ISRs or other mitigating features may be used at the postulated break locations in the RXB (outside the Reactor Pool Bay) areas. Breaks are postulated in accordance with BTP Section B.A.(iii)(2) as shown above.

The CVCS lines in the RXB (outside of the Reactor Pool Bay) are not co-located with any essential SSC. Therefore, dynamic effects are not a concern 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 finalize the stress analysis and the environmental analysis of the high-energy lines outside the reactor pool bay. This includes the identification of any new detection and auto-isolation functions for mitigating an auxiliary boiler high-energy line break. The COL Applicant will update Table 3.6-2, Figure 3.6-16 and Figure 3.6-17 as appropriate.

3.6.2.1.4 Pipe Breaks in the Control Building

There are no high-energy lines in the CRB. No breaks are postulated. Leakage cracks are postulated in the moderate-energy lines for flooding and environmental evaluations 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 and essential equipment in the RWB. Therefore, no breaks or leakage cracks are postulated.

3.6.2.1.6 Pipe breaks On-site (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: A COL applicant that references the NuScale Power Plant design certification will perform stress analysis for high energy lines outside the NPM if needed to identify and mitigate the consequences of potential breaks.

3.6.2.1.7 Moderate-Energy Line breaks and High-Energy Leakage Cracks (all areas)

Moderate-energy line breaks and high-energy leakage cracks do not cause dynamic effects. They are potential sources of environmental effects, (spray, flooding, pressurization, heatup, and radioactivity.) Within the CNV the limiting environmental conditions come from design basis accidents that result in ECCS actuation. The effect due to postulated specific moderate-energy line breaks are bounded by the effects of the main steam line breaks. These conditions are used for the evaluations in Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment. Therefore specific moderate-energy breaks are not postulated here.

Outside of the Reactor Pool Bay, environmental conditions are based upon the rupture of worst (typically largest or hottest) line in the proximity of safety-related SSC. For the environmental and flooding analysis, breaks are assumed to occur anywhere in both the high-energy and moderate-energy lines.

3.6.2.2 Guard Pipe Assembly Design Criteria

Containment penetrations are fabricated as part of the CNV. Piping and components are either welded or bolted to the penetration nozzle. Guard pipes are not used within the CNV to the top of module area.

3.6.2.3 Analytical Methods to Define Forcing Functions and Response Models

Within the CNV and reactor pool bay, all postulated break locations identified in Figure 3.6-2 through Figure 3.6-15 are addressed with an ISR. With the use of an ISR at each postulated break location, the pipe whip and jet impingement forces are effectively contained. The ISRs hold the pipe in place and the dynamic effects jet force and spray is mitigated by the ISR as described in Section 3.6.5.2. Accordingly, vicinity SSC are only minimally affected (minor localized flooding from drainage). Further evaluation of jet impingement forcing functions and response models for impact assessments is therefore not necessary.

3.6.2.4 Dynamic Analysis Methods to Verify Integrity and Operability

As discussed in Section 3.6.2.3, all postulated break locations identified in Figure 3.6-2 through Figure 3.6-15 in the CNV and reactor pool bay are addressed with installation of an ISR. The ISR holds the pipe in place and the jet force and spray mitigated by the ISR as described in Section 3.6.5. Integrity and operability of vicinity SSC and building structures is therefore not challenged, and dynamic analysis to verify integrity and operability is not necessary.

3.6.2.5 Implementation of Criteria Dealing with Special Features

Main Steam and Feedwater Lines from Containment to the Penetrations at the Reactor Pool Wall (including Tees to the Decay Heat Removal System)

In accordance with BTP 3-4 Section B.A.(ii), breaks are not postulated in these segments of piping 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 main steam and feedwater lines do not exceed the following design stress and fatigue 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 $0.8(1.8\ S_h + S_A)$. The portions of the MS and FW lines beyond the CIVs also include flexible piping joints (i.e., ball joints) which significantly reduce thermal stress in the break exclusion zone.

    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. Other welded features (thermowells, branch lines) within the piping segments have been minimized and are qualified using detailed stress analysis.

3) Piping welds are minimized to the extent possible. There are two eight-inch branch connections per MSS train for the DHRS lines. This results in three circumferential welds on each NPS 12 main steam line. There is a weld between the CNV safe-end and the first DHRS tee, a weld between the first and second DHRS tee and a third weld between the second tee and the main steam isolation valve (see Figure 3.6-15). For both the MSS and FWS lines beyond the CIVs, piping welds have

been minimized to the extent possible. Pipe bends are used instead of welded fittings where space allows.

4) The length of the piping is the minimum practical.

5) There are no pipe anchors or restraints welded to the surface of the pipe.

6) Guard pipes are not used.

7) The welds are included in the ISI program as described in Section 6.6.

Even though portions of the MS and FW lines are within the break exclusion zone, environmental effects resulting from the rupture of these lines is still considered as discussed in Section 3.6.1.2.

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 only 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 main steam line between the CNV and the main steam 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 permanently attached to the outside of the CNV at approximately the 50' level. The condenser is designed 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. Figure 3.6-10 and Figure 3.6-11 provide a simplified drawing of the DHRS. 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.

In accordance with BTP 3-4 Section B.A.(ii), breaks are not postulated in this segment of piping because it meets 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 DHRS lines do not exceed the following design stress and fatigue 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. Other welded features (thermowells, branch lines) within the piping segment have been minimized and are qualified using detailed stress analysis.

3) There are no longitudinal welds in this piping. Circumferential welds have been minimized to the extent possible. Piping bends are used in place of welded fittings where space allows.

4) The length of the piping is the minimum practical, considering that bends and jogs have been added to reduce the thermal stresses in the system.

5) There are no pipe anchors or restraints welded to the surface of the pipe.

6) Guard pipes are not used.

7) The welds are included in the ISI program as described in Section 6.6.

3.6.3 Leak-Before-Break Evaluation Procedures

The GDC 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 main steam lines and feedwater lines inside containment are shown in Figure 3.6-2, Figure 3.6-3, Figure 3.6-4, and Figure 3.6-5, respectively.

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 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 present, stress corrosion-cracking (SCC) is possible. 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 yield strength as determined by the 0.2 percent offset method does not exceed 90 ksi.

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.

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 B&PV Code 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 reactor 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. The stainless steel welds are also not susceptible to thermal aging embrittlement because they are fabricated in accordance with Section III of the ASME B&PV Code and U.S. NRC Regulatory Guide 1.31.

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. (See Figure 3.6-2, Figure 3.6-3, Figure 3.6-4 and Figure 3.6-5)

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 any of 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 550 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 that is compatible with the base metals as specified by the design specification. 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 that is compatible with the base metals as specified by the design specification. The tensile material properties used in the analysis of FWS materials are at 300 degrees F.

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)	3.27(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.

$$F_{Combined} = S_M(|F_{DW} + F_{TH} + F_{PR}| + |F_{SSE}| + |F_{SAM}|) \qquad \text{Eq. 3.6-1}$$

$$M_{i,\,Combined} = S_M(|M_{i,\,DW} + M_{i,\,TH} + M_{i,\,PR}| + |M_{i,\,SSE}| + |M_{i,\,SAM}|) \qquad \text{Eq. 3.6-2}$$

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:

$$F_{Combined} = |F_{DW}| + |F_{TH}| + |F_{PR}| + |F_{SSE}| + |F_{SAM}| \qquad \text{Eq. 3.6-3}$$

$$M_{i,\,Combined} = |M_{i,\,DW}| + |M_{i,\,TH}| + |M_{i,\,PR}| + |M_{i,\,SSE}| + |M_{i,\,SAM}| \qquad \text{Eq. 3.6-4}$$

The total moment for the primary bending stress is calculated as square root of the sum of squares (SRSS):

$$M_{Combined} = \sqrt{M_{x,\,Combined}^2 + M_{y,\,Combined}^2 + M_{z,\,Combined}^2} \qquad \text{Eq. 3.6-5}$$

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:

$$\begin{aligned} F &= F_{DW} + F_{TH} + F_{PR} \\ M_X &= (M_X)_{DW} + (M_X)_{TH} \\ M_Y &= (M_Y)_{DW} + (M_Y)_{TH} \\ M_Z &= (M_Z)_{DW} + (M_Z)_{TH} \end{aligned} \qquad \text{Eq. 3.6-6}$$

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:

$$M = \sqrt{M_X^2 + M_Y^2 + M_Z^2} \qquad \text{Eq. 3.6-7}$$

For the maximum stress calculation, the maximum axial force and moments are:

$$\begin{aligned} F &= |F_{DW}| + |F_{PR}| + |F_{SSE}| \\ M_X &= |(M_X)_{DW}| + |(M_X)_{SSE}| \\ M_Y &= |(M_Y)_{DW}| + |(M_Y)_{SSE}| \\ M_Z &= |(M_Z)_{DW}| + |(M_Z)_{SSE}| \end{aligned} \qquad \text{Eq. 3.6-8}$$

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:

$$M = \sqrt{M_X^2 + M_Y^2 + M_Z^2} \qquad \text{Eq. 3.6-9}$$

In the above equations, the moment due to the internal pressure is not included although it is included in Eq. 3.6-2 and Eq. 3.6-4, because the moment due to internal pressure is negligible. For limit load analysis, the thermal expansion and SAM loads are not included in Eq. 3.6-50 because they are secondary loads.

The stresses due to axial loads and moments are then calculated by:

$$\sigma = \frac{F}{A} + \frac{M}{Z} \qquad \text{Eq. 3.6-10}$$

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 since 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. 3.6-11.

$$\frac{\varepsilon}{\varepsilon_0} = \frac{\sigma}{\sigma_0} + \alpha\left(\frac{\sigma}{\sigma_0}\right)^n \qquad \text{Eq. 3.6-11}$$

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% 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:

$$\delta_e = \frac{4\sigma_B a}{E} V_1^B\left(\frac{a}{b}, \frac{R}{t}\right) \quad \text{Eq. 3.6-12}$$

where,

$$a = R_m\theta = \text{half crack length at the mean radius,} \quad \text{Eq. 3.6-13}$$

$$b = \pi R_m = \text{half pipe circumference,} \quad \text{Eq. 3.6-14}$$

θ = half crack angle in radians,

R = mean pipe radius, $R = R_m$,

E = modulus of elasticity.

$$\sigma_B = \frac{MR}{I} = \text{remote bending stress} \quad \text{Eq. 3.6-15}$$

M = remote bending moment.

$$I = \frac{1}{4}\pi\left(R_0^4 - R_i^4\right) \cong \pi R^3 t = \text{area moment of inertia} \quad \text{Eq. 3.6-16}$$

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:

$$\delta_p = \alpha\varepsilon_0 a H_2^B\left(\frac{a}{b}, n, \frac{R}{t}\right)\left(\frac{M}{M_0}\right)^n \quad \text{Eq. 3.6-17}$$

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.

$$M_0 = 4\sigma_0 R^2 t\left(\cos\frac{\theta}{2} - \frac{1}{2}\sin\theta\right)$$ = Reference bending moment Eq. 3.6-18

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

$$\delta = \delta_e + \delta_p = \frac{4\sigma_B a}{E} V_1^B\left(\frac{a}{b}, \frac{R}{t}\right) + \alpha\varepsilon_o a H_2^B\left(\frac{a}{b}, n, \frac{R}{t}\right)\left(\frac{M}{M_0}\right)^m \quad \text{Eq. 3.6-19}$$

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:

$$K_T = \frac{F}{2\pi R t}\sqrt{\pi a}F_T(\theta) \quad \text{Eq. 3.6-20}$$

$$K_B = \frac{M}{\pi R^2 t}\sqrt{\pi a}F_B(\theta) \quad \text{Eq. 3.6-21}$$

where,

$$F_T(\theta) = 1 + 7.5\left(\frac{\theta}{\pi}\right)^{\frac{3}{2}} - 15\left(\frac{\theta}{\pi}\right)^{\frac{5}{2}} + 33\left(\frac{\theta}{\pi}\right)^{\frac{7}{2}} \quad \text{Eq. 3.6-22}$$

$$F_B(\theta) = 1 + 6.8\left(\frac{\theta}{\pi}\right)^{\frac{3}{2}} - 13.6\left(\frac{\theta}{\pi}\right)^{\frac{5}{2}} + 20\left(\frac{\theta}{\pi}\right)^{\frac{7}{2}} \quad \text{Eq. 3.6-23}$$

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:

$$M_e = \frac{FR}{2}\frac{F_T(\theta)}{F_B(\theta)} \qquad \text{Eq. 3.6-24}$$

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

$$\delta_p = \alpha^{\frac{1}{n}}\varepsilon_0 a H_2^B\left(\frac{a}{b}, n, \frac{R}{t}\right)\left(\frac{M}{M_0}\right)^n \qquad \text{Eq. 3.6-25}$$

For pure tension

$$\delta_p = \alpha^{\frac{1}{n}}\varepsilon_0 a H_2^T\left(\frac{a}{b}, n, \frac{R}{t}\right)\left(\frac{F}{F_0}\right)^n \qquad \text{Eq. 3.6-26}$$

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. 3.6-17.

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:

$$\delta_p = 0.5(\alpha + \alpha^{1/n})\varepsilon_0 a H_2^B\left(\frac{a}{b}, n, \frac{R}{t}\right)\left(\frac{M}{M_0}\right)^n \qquad \text{Eq. 3.6-27}$$

The α-correction in Eq. 3.6-27 is applied when using the bending influence function with the equivalent moment calculated by Eq. 3.6-24.

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:

$$A_{crack} = \pi a\delta/2 \qquad \text{Eq. 3.6-28}$$

The perimeter of an ellipse can be approximated by

$$P_{wetted} \approx \pi[3(a + \delta/2) - \sqrt{(3a + \delta/2)(a + 3\delta/2)}] \qquad \text{Eq. 3.6-29}$$

The hydraulic diameter is then calculated by

$$D_H = \frac{4A}{P_{wetted}} \qquad \text{Eq. 3.6-30}$$

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:

$$\Psi = G_c^{\,2} - \left[\frac{x_c v_{gc}}{\gamma_o P_c} - (v_{gc} - v_{lc}) N_1 \frac{dx_e}{dP}\right]^{-1} = 0 \qquad \text{Eq. 3.6-31}$$

Subject to the constraint in terms of pressure equilibrium

$$\Omega = P_c + \Delta P_e + \Delta P_f + \Delta P_a + \Delta P_{aa} + \Delta P_k - P_o = 0 \qquad \text{Eq. 3.6-32}$$

where,

G_c = mass flux of the fluid at the crack exit plane,

$$x_e = \frac{S_o - S_l^c}{S_g^c - S_l^c}$$ = equilibrium fluid quality Eq. 3.6-33

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. 3.6-34

$$x_c = N_1 x_e \left[1 - e^{-B\left(\frac{L_a}{D_H} - 12\right)}\right]$$ Eq. 3.6-35

L_a = flow-path length,

$D_H = \frac{4 \cdot \text{Crack Opening Area}}{\text{Crack Opening Perimeter}}$ = the hydraulic diameter perimeter (see Eq. 3.6-30),

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,

$$\Delta P_e = \frac{G_o^2 v_{lo}}{2C_D^2}$$ = pressure loss due to entrance effects Eq. 3.6-36

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,

$$\Delta P_f = f\frac{L_a}{D_i}\frac{\overline{G}_c^2}{2}[(1-\bar{x})\overline{v}_l + \bar{x}\bar{v}_g]$$ = Pressure loss due to friction Eq. 3.6-37

$\bar{x}$ = 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. 3.6-38

μ = crack face roughness,

$$\Delta P_a = \overline{G}_T^2[(1-x_c)v_{lc} + x_c v_{gc} - v_{lc}]$$ = pressure loss due to acceleration of the fluid as it flows through the crack Eq. 3.6-39

$\overline{G}_T$ = average mass flux in the two-phase region of crack flow,

ΔP_{aa} = acceleration pressure loss due to area change is assumed zero,

$$\Delta P_k = e_v\frac{\overline{G}^2}{2}[\bar{v}_l + \bar{x}(\bar{v}_g - \bar{v}_l)]$$ = pressure loss due to ends and protrusions Eq. 3.6-40

$\overline{G}^2$ = average mass flux G^2 of the fluid

$$e_v = e_n L_a$$ = the total loss coefficient over the flow path Eq. 3.6-41

e_n = the number of velocity heads lost per unit flow path length, which is given in Eq. 3.6-43.

Eq. 3.6-32 and Eq. 3.6-31 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

$$\mu = \begin{cases} \mu_L, & 0 < \dfrac{\delta}{\mu_G} < 0.1 \\ \mu_L + \dfrac{\mu_G - \mu_L}{9.9}\left(\dfrac{\delta}{\mu_G} - 0.1\right), & 0.1 \le \dfrac{\delta}{\mu_G} \le 10 \\ \mu_G, & \dfrac{\delta}{\mu_G} > 10 \end{cases} \qquad \text{Eq. 3.6-42}$$

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

$$e_n = \begin{cases} e_{n_L}, & 0 < \dfrac{\delta}{\mu_G} < 0.1 \\ e_{n_L} - \dfrac{e_{n_L}}{11}\left(\dfrac{\delta}{\mu_G} - 0.1\right), & 0.1 \le \dfrac{\delta}{\mu_G} \le 10 \\ 0.1 e_{n_L}, & \dfrac{\delta}{\mu_G} > 10 \end{cases} \qquad \text{Eq. 3.6-43}$$

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:

$$\frac{L_a}{t} = \begin{cases} K_{G+L}, & 0 < \dfrac{\delta}{\mu_G} < 0.1 \\ K_{G+L} - \dfrac{K_{G+L} - K_G}{9.9}\left(\dfrac{\delta}{\mu_G} - 0.1\right), & 0.1 \le \dfrac{\delta}{\mu_G} \le 10 \\ K_G, & \dfrac{\delta}{\mu_G} > 10 \end{cases} \quad \text{Eq. 3.6-44}$$

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 containment evacuation system (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:

$$S_I = \begin{cases} \dfrac{2\sigma_f}{\pi}(2\sin\beta - \sin\theta) + (S_m)(P_m), & \text{if } \beta + \theta \le \pi \\ \dfrac{2\sigma_f}{\pi}\sin\beta + (S_m)(P_m), & \text{if } \beta + \theta > \pi \end{cases} \quad \text{Eq. 3.6-45}$$

where,

$$\beta = \begin{cases} 0.5\left[(\pi - \theta) - \dfrac{\pi P_m}{\sigma_f}\right], \text{ if } \beta + \theta \le \pi \\ -\dfrac{\pi P_m}{\sigma_f}, \text{ if } \beta + \theta > \pi \end{cases} \qquad \text{Eq. 3.6-46}$$

$$P_m = \frac{F_x}{A} = \text{primary membrane stress} \qquad \text{Eq. 3.6-47}$$

Fx = total applied axial force,

A = cross-section area,

$$\theta = \frac{a_c}{R_m} = \text{postulated through-wall circumferential crack half-angle} \qquad \text{Eq. 3.6-48}$$

R_m = pipe mean radius,

S_M = 1 = safety margin on the load,

$$\sigma_f = 0.5(\sigma_y + \sigma_u) = \text{flow stress} \qquad \text{Eq. 3.6-49}$$

σ_y = yield strength, and

σ_u = ultimate strength.

The stress index is also expressed in SRP 3.6.3 as:

$$S_I = S_M(P_m + P_b) \qquad \text{Eq. 3.6-50}$$

where,

$$P_b = \frac{M \cdot R_m}{I} = \text{primary bending stress} \qquad \text{Eq. 3.6-51}$$

$$M = \frac{\left(\sigma_{max} - \dfrac{F_x}{A}\right)I}{R_m} = \text{applied maximum moment} \qquad \text{Eq. 3.6-52}$$

σ_{max} = applied maximum stress, and

I = area moment of inertia.

The σ_{max} can be determined by making S_I in Eq. 3.6-45 equal to that in Eq. 3.6-50.

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. 3.6-10 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. 3.6-6 and Eq. 3.6-7, the normal axial force and moment are calculated. The maximum axial force and moment are calculated using Eq. 3.6-8 and Eq. 3.6-9. 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. 3.6-10.

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 since 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. 3.6-6 and Eq. 3.6-7, the normal axial force and moment are calculated. The maximum axial force and moment are calculated using Eq. 3.6-8 and Eq. 3.6-9. 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. 3.6-10.

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 data for SBAC are summarized in Table 3.6-3a. 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 data for SBAC are summarized in Table 3.6-3b. 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, since 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)

The GDC 4 requires that components be appropriately protected against the dynamic effects that may result from pipe ruptures. High-energy and moderate-energy piping systems that cannot be fully excluded using either the BTP 3-4, Section B.A.(ii) criteria, or LBB, must be designed for HELB. The specific locations for the postulated break locations are determined using the criteria in BTP 3-4. In general, welds 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 postulated rupture locations, the consequences of HELB can include pipe whip or jet impingement, both of which can potentially damage safety related equipment required for safe shutdown. At break locations, the pipe must either be located such that there is no essential equipment in the area, or the pipe must be restrained from whip, and equipment protected from jet impingement, as needed.

The piping systems that must be considered include the Class 1, Class 2, Class 3 and B31.1, high-energy and moderate-energy systems, located inside and outside of the CNV up to the reactor pool wall penetrations. Piping outside of the NPM is the responsibility of the COL applicant.

3.6.4.1 Postulation of Pipe Breaks in Areas Other than Containment Penetration

With the exceptions of those portions of piping identified in Section 3.6.2.1.1, breaks in Class 1 piping (ASME Code, Section III) are postulated at the following locations in each piping and branch run:

a) at terminal ends

b) at intermediate locations where the maximum stress range as calculated by Eq. (10) and either Eq. (12) or Eq. (13) exceeds 2.4 S_m

c) at intermediate locations where the cumulative usage factor exceeds 0.1

The RCS/CVCS discharge piping is representative of the NPM ASME Class 1 piping with respect to deadweight, seismic, thermal transient and fatigue loading. The discharge line is longer than other Class 1 lines, with more seismic supports and longer spans between thermal restraints. Therefore, this analysis presents the more challenging analysis case.

As a result of piping reanalysis, the highest stress locations may be shifted; however, the initially determined intermediate break locations need not be changed unless one of the following conditions exists:

i) the dynamic effects from the new (as-built) intermediate break locations are not mitigated by installation of ISRs.

ii) a change is necessary in pipe parameters such as major differences in pipe size, wall thickness, and routing.

With the exceptions of those portions of piping identified in Section 3.6.2.1.1, breaks in Class 2 and 3 piping (ASME Code, Section III) are postulated at the following locations in those portions of each piping and branch run:

d) at terminal ends

e) at intermediate locations selected by one of the following criteria:

i) At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Or, where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.

ii) At each location where stresses are calculated by the sum of Eqs. (9) and (10) in NC/ND-3653 of ASME Code, Section III, to exceed 0.8 times the sum of the stress limits given in NC/ND-3653.

As a result of piping reanalysis, due to differences between the design configuration and the as-built configuration, the highest stress locations may be shifted; however, the initially determined intermediate break locations may be used unless redesign of the piping resulting in a change in pipe parameters (diameter, wall thickness, routing) is necessary, or the dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe-whip restraints and jet shields.

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 seismically analyzed non-ASME Class piping are addressed in Section 3.6.2.1.3.

3.6.4.2 Reactor 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 Reactor 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.

The bases for break exclusion zones in areas away from containment penetrations and areas within containment penetrations are described in Section 3.6.2.1.2. The guidance relates to both Class 1 and Class 2 piping systems, where the allowable stresses identified in Section 3.6.2.1.2, which are itemized in Table 3.6-6 and Table 3.6-7, are based upon limits given in Table 3.6-5. The guidance is derived from BTP 3-4 Section B.A.(ii).

3.6.4.4 Jet Loads and Piping Moments

Jet loads have been calculated for NPS 2 through NPS 12 for the CVCS, FWS, MSS, and DHRS process piping using guidance in SRP 3.6.2 and BTP 3-4. All piping runs generally employ 5D (i.e., five diameters) radius bends, with several larger radius bends (greater than 24 inch). Nonetheless, CVCS, FWS, MSS, and DHRS jet loads from 5D bends, assuming a pipe support near one end of the bend, result in creating a fully plastic hinge (i.e., plastic cross-section) as demonstrated in Table 3.6-8. Creation of a plastic hinge with jetting fluid has the potential for causing pipe whipping, as well as potential jet impingement on nearby essential equipment.

HELB jet loads and associated maximum bending moments that occur on the supported-end of a 5D bend pipe when subjected to operating temperature and pressure conditions are shown in Table 3.6-8. In accordance with SRP 3.6.2, the jet thrust load is based on operating pressure and temperature (see Table 3.6-4). A 5D length moment-arm is utilized to determine if the lower-bound values result in creating plastic hinges. The second column from the right shows the value of the bending moment that would cause a fully-plastic cross-section.

As evident from the right-most column of Table 3.6-8 for $R = M / M_p$, which is the ratio of maximum bending moment to fully-plastic bending moment, values are greater than unity. This implies that the high-energy lines postulated for HELB are subject to pipe whip. Further, piping runs with larger bend radii than 5D automatically are subject to pipe whip. Lastly, at locations with ISRs employed, a full- circumference pipe rupture at a weld might still occur, but the joint is restrained from moving further apart than the tolerance between the welded pipe collars and the ISR grooves (See Section 3.6.5). As such, typical ISRs allow a 0.125" gap at weld failure, such that a disk-type jet is developed (see Figure 3.6-35). Standard ANS 58.2 contains jet impingement force models for full-circumference break with limited separation. (Reference 3.6-15)

3.6.4.5 Break Locations inside the Containment Vessel

As discussed in Section 3.6.2.1.1, only the CVCS and DHRS process piping are subject to HELB inside the CNV. The MSS and FWS are qualified for LBB (Section 3.6.3), thus the welds in MSS and FWS are excluded from break dynamic effects.

3.6.4.5.1 Chemical and Volume Control System and Decay Heat Removal System Piping inside Containment

Table 3.6-2 identifies the resultant postulated break locations within the high-energy CVCS reactor coolant system discharge, RCS injection, pressurizer spray, RCS high-point degasification, and DHRS lines inside containment. To preclude the need to further evaluate the consequences of pipe whip and jet impingement at these locations, NuScale ISRs are installed (see Section 3.6.5).

3.6.4.6 High-Energy Piping outside Containment

Table 3.6-2 identifies the postulated break locations within the high-energy CVCS RCS discharge, RCS injection, PZR spray, and RCS high-point degasification lines outside containment. To preclude the need to further evaluate the consequences of pipe whip and jet impingement at these locations, NuScale ISRs are installed (see Section 3.6.5). Piping outside of the NPM is the responsibility of the COL applicant.

3.6.5 Integral Jet Impingement Shield and Pipe Whip Restraint

One method used in the NuScale design to mitigate the dynamic effects of a pipe rupture is installation of an ISR. The basic design of the ISR is a cylindrical sleeve which encases the postulated rupture location. The sleeve contains circumferential grooves on the inside surface to accommodate collars that are welded to the pipe and to hold the ISR in place after its installation and during plant operations. In the event of a pipe rupture, the collars are captured by the ISR grooves, thus preventing the pipe from whipping. The ISR, which encloses the rupture, also restricts the escaping fluid and shields the surroundings from fluid jets. A typical ISR is shown in Figure 3.6-33 and Figure 3.6-34.

The ISRs are designed to be compatible with the plant design so that when an ISR is required at a specific location, the impacts to the design of the piping system and to plant operations and maintenance are minimized. Because ISRs fit closely around the pipe, the physical envelope of the ISR is small and unlikely to interfere with neighboring components. Additionally, because an ISR is fully supported by the encased piping, additional supporting structures are not necessary. The only requirements for the implementation of an ISR at a particular location are that there is sufficient space on either side of the postulated break that is free of interferences such that the ISR can be designed and installed, and collars can be placed. The ISR is designed to be removable for inspection of the piping welds if required, with sufficient clearance between the collars and welds to provide for ultrasonic inspection of each weld. To achieve this, the ISR sleeve is fabricated in two halves and is bolted in place over the postulated break location.

The methodology used to size and qualify the ISRs is described below.

In order for the ISR to perform its function without failing, the design considers the jet thrust load, which pushes the broken pipe ends apart, the internal pressure increase acting on the sleeve, as well as the preexisting loads that were carried by the piping prior to rupture.

To compute the static jet thrust load due to a jet from a broken pipe end, the guidance in SRP 3.6.2 Revision 2, Section III.2.C. (iv) is employed. This is conservative because no actual jet is permitted to form, as the ISR is held in place by collars which constrain the break separation to less than 0.125 inch. The jet thrust force is computed for operating conditions using the simplified approach described in SRP 3.6.2, as

$$T = (K)(P)(A) \qquad \text{Eq. 3.6-53}$$

T = jet thrust force, pounds

K = thrust coefficient equal to 1.26 for steam, saturated water or steam-water mixtures, 2.0 for subcooled, non-flashing water

P = system pressure prior to the pipe break, (psi)

A = pipe break area, (in^2)

The thrust due to pipe break is computed with the full area of the pipe, including the metal area, because of the confinement of the sleeve, even though the sleeve includes holes for pressure relief. The system pressures used to compute the pipe break thrust load are assumed to be the normal operating pressure of the piping. System pressure for the CVCS is 1850 psia.

The design of the ISRs is such that thrust loads are taken in pure shear by the collars and sleeve with bearing on the collars and sleeve where contact occurs. This design method, therefore, limits the primary stresses on individual critical parts of the ISR. The thickness of each collar is less than or equal to the wall thickness of the pipe to which it is attached, for welding considerations, and a full penetration weld with stress-relief treatment is used. The thickness of the collars are specified to meet the allowable shear stress criteria. To preclude pullout of the collar, the collar long axis is designed to be perpendicular to the pipe longitudinal axis. The collars are completely captured by the sleeve grooves. The sleeve grooves are designed to receive the collars, ensuring the shear forces are applied near the base of the collars to minimize bending moment on the collars.

To accommodate the pressure increase internal to the ISR following a pipe rupture, the design uses a pressure relief chamber. The pressure relief chamber is shown in Figure 3.6-34. The chamber includes holes through which fluid from the postulated pipe break is released, thereby relieving pressure inside the ISR. However, for qualification, the pressure acting on the onside surface of the chamber is conservatively assumed to be the initial operating pressure in the pipe prior to the break. The essential features of the pressure relief chamber are:

- the same outside diameter as that of the sleeve.
- an increased inside diameter to permit the chamber to be formed.

- pressure relief holes to permit fluid to be released in a controlled way.
- raised longitudinal ridges on the inside surface of the pressure relief chamber maintain a consistent gap with the outside diameter of the pipe along the entire length of the ISR
- The holes are arranged symmetrically on each side of the postulated pipe break weld.

Analysis of an ISR designed for a NPS 2 straight pipe to straight pipe configuration has been performed. Many of the locations where pipe breaks are postulated are equivalent to a straight pipe to straight pipe configuration because of the following considerations.

- Breaks at straight pipe connections to a long neck flange are postulated; custom flanges may be used which are similar to a straight pipe to straight pipe configuration.
- Safe end to straight pipe welds may be made the same as straight pipe to straight pipe welds by use of long safe ends.

Structural evaluation of the ISR design is performed using finite element analysis. The finite element code ANSYS is used to perform both a linear and a nonlinear analysis. A linear analysis is used to qualify the ISR to ASME Level D requirements, while a separate plastic analysis is performed to verify that the piping segments will collapse before if the ISR experiences excessive plasticity. Both analyses utilize a 3D finite element model which includes the sleeve, eight bolts, washers and nuts, and two piping sections with collars, which represent the broken pipe which is being restrained.

Linear Analysis

The ISR is analyzed to Level D limits specified within Appendix F of the ASME Boiler and Pressure Vessel Code. The ISR and the pipe collar are designed using the limits given in F-1331. The ISR bolts are designed using the limits given in F-1335.

Loads included in the analysis are: an overturning moment equal to the maximum moment carried by the pipe applied to a pipe segment, thrust forces applied to each pipe segment, pressure applied the cavity between the ISR and the pipe, and a preload added to each bolt.

Cut lines in the model are used to analyze the stresses in both the ISR and the collar. It should be noted that this is a conservative approach to calculate the membrane stress. Membrane stress is defined as normal stress that is uniformly distributed and equal to the average stress across the thickness of the section under consideration. The stresses extracted by the cut lines are stresses averaged over a single line at a highly stressed location, not a whole section, and therefore produce conservative results.

The results of the linear finite element analysis of the ISR are provided in the tables below.

ISR Stress Results

Stress Classification	Cut Line Maximum Stress (psi)	Allowable (psi)	Ratio
Membrane	38332	38880	0.99
Membrane Plus Bending	55508	58320	0.95

Bolt Stress Results

Check	Stress (psi)	Allowable (psi)	Ratio
Tension	56661	96180	0.60
Tension + Bending	91274	134900	0.68

Collar Stress Results

Check	Stress (psi)	Allowable (psi)	Ratio
Average Shear Stress	8787	26628	0.33

Plastic Analysis

A plastic analysis is performed to verify that the piping will collapse before the ISR. This analysis is not a code requirement and is only done to determine if the ISR itself can withstand a greater moment than the piping system.

The model used in the linear analysis is also used for this analysis, however is modified to increase the length of the piping segments. All other geometry is kept the same. The boundary conditions are identical to the linear analysis with the exception of the applied moment. The maximum moment was increased to ensure a moment high enough to cause the piping system to collapse was applied.

For the material properties, the bolts use bi-linear kinematic hardening curve while the ISR and piping segments use a multilinear kinematic hardening curve. The last four inches of the piping segments use elastic material properties in order to help the model converge.

The analysis shows that the pipe and collar have developed through wall plastic strains prior to the ISR developing through wall plastic strains. Therefore the piping segments will collapse prior to the ISR.

The criteria for postulating ruptures, and thus locating ISRs, are discussed in Section 3.6.2 and Section 3.6.4. If stress criteria are used (as opposed to assuming a rupture at every weld and fitting), then these criteria are evaluated during the code stress analysis of the piping systems. The design of the ISRs is such that the impacts to this analysis for piping systems that use ISRs are minimized. If it is determined during the analysis that an ISR is required at a location, the weight (lumped mass) of the ISR is added to the piping model and the analysis is performed again. This process is iterated until the piping passes its code analysis criteria while accounting for the added mass of the required ISRs. The ISRs are designed and located with sufficient clearance between the pipe and the ISR such that they do not normally interact and cause additional piping stresses. A design hot position gap is provided, which allows maximum predicted displacements (e.g., thermal and seismic) to occur without ISR interaction.

A total of 27 ISRs are employed inside the containment vessel. This includes 17 on the RCS injection, discharge, high point vent, and pressurizer spray lines, and two on the DHRS condensate return lines.

In the reactor pool bay, there are an additional 12 ISRs: each CVCS line (RCS injection, discharge, high point vent, and pressurizer spray line) has three.

In addition to the ISRs directly associated with the NPM, additional ISRs may be used within the RXB, depending on the routing of COL applicant scope high- and moderate-energy piping beyond the NPMs and through the BOP.

These ISRs correspond to break locations discussed in Section 3.6.2. Postulated break locations that do not have ISRs installed are also discussed in Section 3.6.2.

The ISR shown in Figure 3.6-33 and Figure 3.6-34, and whose sizing methodology is discussed above, is intended to mitigate the adverse effects of a circumferential pipe break. Although longitudinal pipe breaks may also be required to be postulated per the criteria given in BTP 3-4, there are currently none being postulated. An objective of the piping design is to design the piping such that stress and fatigue limits as specified in Section 3.6.2 are satisfied, thus precluding the need to postulate pipe breaks in the run piping (i.e. locations other than terminal ends). Therefore, the majority of postulated breaks are located at terminal ends. Per the criteria in BTP 3-4, longitudinal pipe breaks need not be postulated at terminal ends. Current detailed stress analyses of piping systems have identified only two locations where piping ruptures must be postulated in locations other than terminal ends, both of which are located on NPS 2 piping. Additional criteria in BTP 3-4 excludes the postulation of longitudinal breaks in piping of sizes less than NPS 4. The ISR designs will continue to be developed to be compatible with the various configurations of piping where breaks are postulated and with the types of break which must be postulated, including longitudinal breaks, as the detailed piping analyses indicate they are required.

3.6.5.1 Integral Jet Impingement Shield and Pipe Whip Restraint Computational Fluid Dynamics Analysis

The dynamic jet effect for a postulated pipe break was evaluated as required by GDC 4 and SRP 3.6.2. Computational fluid dynamics (CFD) models were developed of the flow exiting the ISR following the pipe break. The CFD models were used to determine the loads on nearby components caused by jet impingement.

ANSYS CFX was used for the detailed CFD evaluation, utilizing an axisymmetric 2D model to capture the flow out of a single ISR hole.

The RCS pipe break case uses a high temperature of 550 degrees F that bounds the cold leg temperatures. At 550 degrees F the saturation pressure is 1,045 psia. The ISR relief hole exit flow is taken at a pressure of 1,044 psia, a temperature of 550 degrees F, and a constant inlet speed of 1,602 ft/s (the speed of sound at these conditions). The DHRS has both a high temperature condition that flashes to steam and a low temperature condition that does not flash. The DHRS high temperature pipe break uses a temperature of 310 degrees F, an inlet pressure of 77 psia, and a constant inlet speed of 1,624 ft/s. The DHRS low temperature pipe break uses a temperature of 40 degrees F and an inlet pressure of 800 psia. 800 psia correlates to an inlet speed of 341 ft/s.

The CFD models are 5 degree wedges with side planes set to symmetry boundary conditions. The ISR walls are set to be free-slip walls. The inlet is set to supersonic for the RCS break and DHRS high temperature break cases and subsonic for the DHRS low temperature case.

For the RCS ISR, Figure 3.6-36 shows a graph of total pressure along the axis. Note that X=0 is the ISR relief hole exit with +X normal to the hole. The total pressure drops below 50 psi within the first three inches. Figure 3.6-37 shows a plot of total pressure above the axis at distances of 5, 10, 15, and 20 inches away from the relief hole exit, with Y=0 at the centerline. The plots above the axis show how large the jet is at these distances (force can be calculated by multiplying total pressure and area, where Figure 3.6-37 gives the radius of the circular jet area). Total pressure is reported as relative pressure. For absolute total pressure, add 1 atm.

Figure 3.6-38 shows a graph of total pressure along the axis for the DHRS high temperature ISR case. Note that X=0 is at the ISR relief hole exit with +X going normal to the hole. The total pressure drops below 20 psi within the first five inches.

For the DHRS low temperature ISR case, Figure 3.6-39 shows a graph of total pressure along the discharge access. Note that X=0 is at the ISR relief hole exit with +X normal to the hole. Figure 3.6-40 shows a plot of total pressure above the axis at distances of 5, 10, 15, and 20 inches away from the relief hole exit, with Y=0 at the centerline. The plots above the axis demonstrate how large the jet is at these distances (force is calculated by multiplying total pressure and area, where Figure 3.6-40 provides the radius of the circular jet area). Total pressure is relative pressure, for absolute total pressure, add 1 atm.

As demonstrated by these CFD analysis results, for the RC,S ISR, and the DHRS high temperature ISR applications, total pipe break discharge pressure within five inches of the ISR drops below 50 psia. The lower temperature DHRS ISR application has higher total pressures that extend farther radially from the pipe break, due to the absence of discharge flashing.

Application of the NuScale ISR device in areas with common pipe routing and essential SSC spacing therefore assures the mitigation of detrimental jet impingement effects to nearby safety-related, risk significant SSC. Jet impingement loads at reasonable radial distances from the ISR are low, allowing for proper design and placement of vicinity SSC.

3.6.5.2 Integral Jet Impingement Shield and Pipe Whip Restraint Confirmatory Test Program

As the NuScale ISR is a first-of-a-kind jet impingement shield and pipe whip restraint, proof of concept testing is being performed to validate the analytical model and demonstrate that the ISR performs its intended function to mitigate the dynamic effects of postulated high energy pipe breaks.

The ISR test objectives are to measure the total pressure in the jet exiting the ISR as a function of distance from the ISR, to measure the pressure inside the ISR chamber to validate the analytical models, to measure the acceleration on the pipe ends during a simulated pipe break to validate the structural design, and to confirm the ability to fabricate and install the ISR on a prototypic section of pipe.

The test facility replicates the transient nature of a pipe break by having two ends of pipe initially held together. After the simulated break is initiated, the two ends of pipe accelerate in opposite directions and a gap is established between the two separated ends of pipe. The gap opening is controlled by the clearance between the ISR grooves and the welded collars on the pipe. Fluid escapes the pipe through the gap into the ISR pressure relief chamber and then exits the chamber into the surrounding environment through the ISR holes.

Of the high-energy piping systems that use ISRs, the CVCS lines contain the highest pressure. The CVCS injection and discharge lines are approximately at RCS pressure and contain subcooled liquid water. The temperature of the discharge line is approximately equal to the RCS downcomer temperature.

The as-installed clearance between the pipe collars and the ISR grooves is measured before each test. The gap length between the ends of the pipe is measured after each test so that the break area can be calculated. The ISR is disassembled and inspected after each test to ensure that the dimensions meet the fabrication drawings and components performed as intended.

The facility design and closure force minimize leakage between the pipe ends before the simulated break in order that leakage not accumulate in the ISR chamber during startup.

The opening time for the ISR gap is as small as achievable to replicate a realistic pipe break.

Pressure instrumentation is provided inside the chamber of the ISR. Additional pressure instruments are located outside the ISR to measure the pressure distribution as a function of distance from the ISR.

Recorded test parameters include pipe internal pressure and temperature at the break, pipe flow rate through the break, ISR chamber pressure, pressure external to the ISR, and pipe acceleration.

3.6.6 References

3.6-1 Electric Power Research Institute, "An Engineering Approach for Elastic-Plastic Fracture Analysis," NP-1931, July 1981.

3.6-2 Electric Power Research Institute, "Advances in Elastic-Plastic Fracture Analysis," NP-3607, August 1984.

3.6-3 Electric Power Research Institute, "Elastic-Plastic Fracture Analysis of Through-Wall and Surface Flaws in Cylinders," NP-5596, January 1988.

3.6-4 U.S. Nuclear Regulatory Commission, "Analysis of Experiments on Stainless Steel Flux Welds," NUREG/CR-4878, April 1987.

3.6-5 U.S. Nuclear Regulatory Commission, "The Development of a J-estimation Scheme for Circumferential and Axial Through-wall Cracked Elbows," NUREG/CR-6837, Vol. 2, Appendix F, June 2005.

3.6-6 Electric Power Research Institute, "Crack-Opening Area Calculations for Circumferential Through-Wall Pipe Cracks," NP-5959-SR, August 1988.

3.6-7 U.S. Nuclear Regulatory Commission, "Evaluation and Refinement of Leak-Rate Estimation Models," NUREG/CR-5128, June 1994.

3.6-8 Henry, R.E., "The Two-Phase Critical Discharge of Initially Saturated or Subcooled Liquid," Nuclear Science and Engineering, Vol. 41, pp. 336-342, 1970.

3.6-9 Henry, R. E. and Fauske, H. K., "Two-Phase Critical Flow at Low Qualities, Part I: Experimental," Nuclear Science and Engineering, Vol. 41, pp. 79-91, 1970.

3.6-10 U.S. Nuclear Regulatory Commission, "Probabilistic Pipe Fracture Evaluations for Leak-Rate-Detection Applications," NUREG/CR-6004, April 1995.

3.6-11 NuScale Power, LLC, Reactor Module Test and Inspection Elements, ER-A010-2186, Revision 0.

3.6-12 ASME Operation and Maintenance Code Assessment, ER-A010-3875, Revision 0.

3.6-13 U.S. Nuclear Regulatory Commission, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," Regulatory Guide 1.45, Revision 1, May 2008.

3.6-14 "Improved J and COD Estimation by GE/EPRI Method in Elastic to Fully Plastic Transition Zone," Engineering Fracture Mechanics, Volume 73, Issue 14, pages 1959-1979 September 2006.

3.6-15 American National Standards Institute/American Nuclear Society, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," ANSI/ANS-58.2-1988.

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
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
UWS	Utility water system	(5)	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
	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	High
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 nozzle-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 (see Figure 3.6-6 thru Figure 3.6-11)
Break locations inside containment		
RCS injection (Figure 3.6-6)	1	Terminal end - RPV head
		Tee/pipe weld
		Pipe-to-pipe weld
		Valve/pipe weld
		Terminal end - containment boundary
RCS discharge (Figure 3.6-7)	1	Terminal end - RPV head
		Valve/pipe weld
		Pipe-to-pipe weld
		Terminal end - containment boundary
Pressurizer spray (Figure 3.6-8)	1	Terminal end - RPV head
		Valve/pipe weld
		Terminal end - containment boundary
		Tee/pipe weld
RCS high-point vent (Figure 3.6-9)	1	Terminal end - RPV head
		Valve/pipe weld
		Pipe-to-pipe weld
		Terminal end - containment boundary
DHRS #1 (Figure 3.6-10)	2	Terminal end - containment boundary
DHRS #2 (Figure 3.6-11)	2	Terminal end - containment boundary
Break locations outside the CNV to the NPM disconnect flange		
RCS injection (Figure 3.6-12)	3	Valve/pipe weld
		Tee/pipe weld
		Tee/flange weld
RCS discharge (Figure 3.6-12)	3	Valve/pipe weld
		Tee/pipe weld
		Tee/flange weld
Pressurizer spray (Figure 3.6-13)	3	Valve/pipe weld
		Tee/pipe weld
		Tee/flange weld
RCS high-point vent (Figure 3.6-13)	3	Valve/pipe weld
		Tee/pipe weld
		Tee/flange weld
Break locations in the RXB (outside the NPM disconnect flange), documented in the NuScale Pipe Rupture Hazards Analysis.		

Table 3.6-3a: Summary of Main Steam Line Bounding Analysis Curves

NPS 8 Base Metal		NPS 8 Welds (Pipe- to-Pipe and Pipe-to-Safe-End)		NPS 12 Base Metal		NPS 12 Welds		NPS 8 Elbow	
Normal Stress (psi)	Max Stress (psi)	Normal Stress (psi)	Max Stress (psi)	Normal Stress (psi)	Max Stress (psi)	Normal Stress (psi)	Max Stress (psi)	Normal Stress (psi)	Max Stress (psi)
3,136	4,571	3,136	5,334	1,229	2,383	1,229	3,089	3,136	3,819
5,136	10,741	5,136	11,447	1,309	2,753	1,309	3,517	5,136	8,910
9,136	23,204	9,136	22,114	3,229	10,826	3,229	12,613	9,136	18,896
13,136	33,733	13,136	31,290	5,229	17,876	5,229	19,807	18896	18,896
17,136	41,056	17,136	38,738	9,229	30,077	9,229	30,789		
21,136	45,548	21,136	44,580	13,229	39,046	13,229	39,078		
25,136	47,720	25,136	49,026	17,229	44,763	17,229	45,324		
29,136	49,131	29,136	52,380	21,229	47,966	21,229	49,915		
				25,229	49,526	25,229	53,274		
				29,229	50,534	29,229	55,715		

Table 3.6-3b: Summary of Feedwater System Line Bounding Analysis Curves

NPS 4 Base Metal		NPS 4 Welds		NPS 5 Base Metal		NPS 5 Welds	
Normal Stress psi	Max Stress psi	Normal Stress psi	Max Stress psi	Normal Stress psi	Max Stress psi	Normal Stress psi	Max Stress psi
2,052	6,698	969	1,100	1,079	4,611	1,079	5,465
3,135	11,722	2,200	8,648	2,160	11,692	2,308	14,294
4,219	15,872	3,430	14,718	3,241	17,021	3,536	20,677
5,302	19,419	4,661	19,592	4,321	21,218	4,764	25,563
7,469	25,358	5,892	23,666	5,402	24,691	5,992	29,516
9,635	30,227	8,354	30,318	7,563	30,233	8,448	35,615
11,802	34,232	10,815	35,553	9,724	34,548	10,904	40,302
13,968	37,641	13,277	39,878	11,885	38,104	13,360	44,110
16,135	40,516	15,738	43,529	14,046	41,058	15,816	47,253
18,301	42,925	18,200	46,600	16,207	43,510	18,272	49,860
20,468	44,936	20,661	49,175	18,368	45,543	20,728	52,025
22,634	46,616	23,123	51,330	20,529	47,229	23,184	53,827
26,968	49,204	25,584	53,136	22,690	48,631	25,640	55,331
31,301	51,052	30,508	55,931	27,012	50,784	30,552	57,653
44,300	54,180	35,431	57,938	31,334	52,317	35,464	59,319
		50,200	61,361	44,300	54,908	50,200	62,153

Table 3.6-4: NuScale Power Module Piping Systems Design and Operating Parameters

Process System (NuScale System)	ASME Code	NPS Size	Design		Operating	
			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	1000	550	80	121
CFDS (CNTS-inside CNV)	Class 2	2	165	300	85	100
CFDS (CNTS-outside CNV)	Class 2	4	1000	550	85	100
CES (CNTS)	Class 2	4	1000	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.

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: Jet loads and Maximum Bending Moments

Process System	Pipe Size (NPS)	Jet Load (kip)	5D Bend (in.)	Bending Moment, *M* (in-kip)	Plastic Moment, M_p (in-kip)	R = *M* / M_p
CVCS	2	8.36	10	83.6	26.07	3.21
FWS	4	11.35	20	226.9	162.51	1.40
	5	17.99	25	449.7	288.03	1.56
DHRS	2	6.26	10	62.6	25.93	2.41
	6	59.17	30	1775.0	456.18	3.89
MSS	8	25.56	40	1022.3	838.21	1.22
	12	57.18	60	3430.8	2574.16	1.33

Figure 3.6-1: Piping Systems Associated with the NuScale Power Module

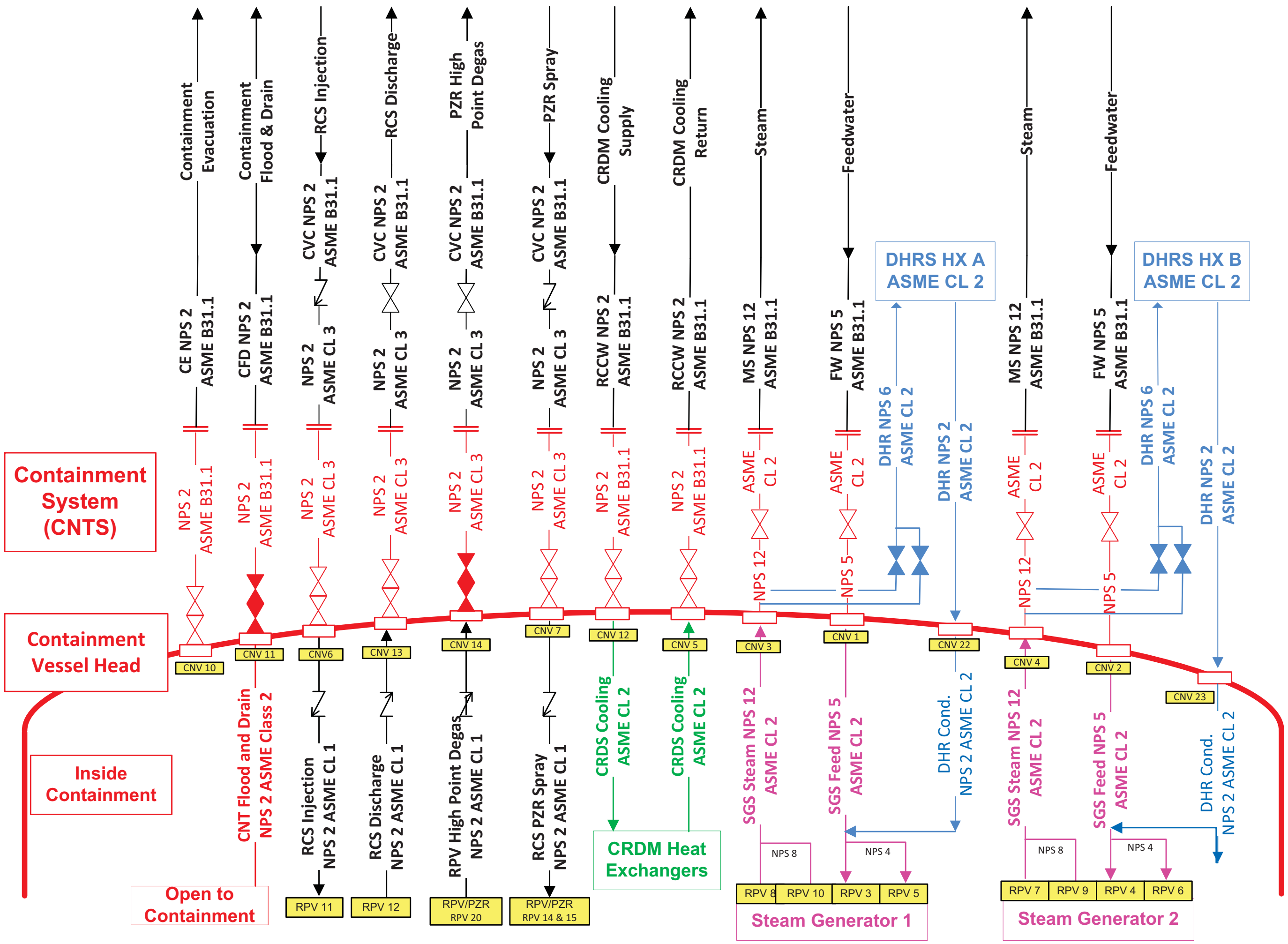


Figure 3.6-2: Main Steam Line 1

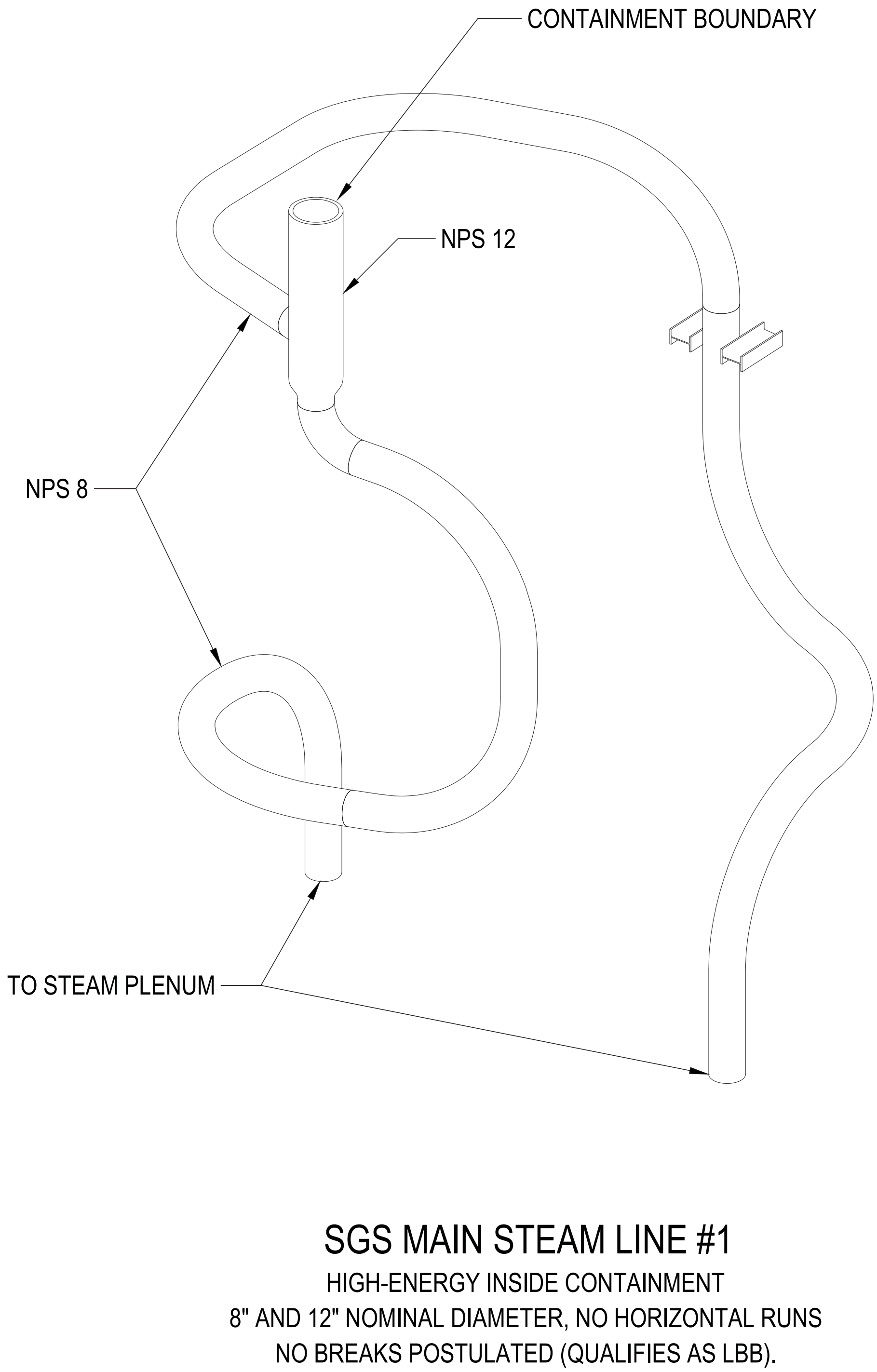


Figure 3.6-3: Main Steam Line 2

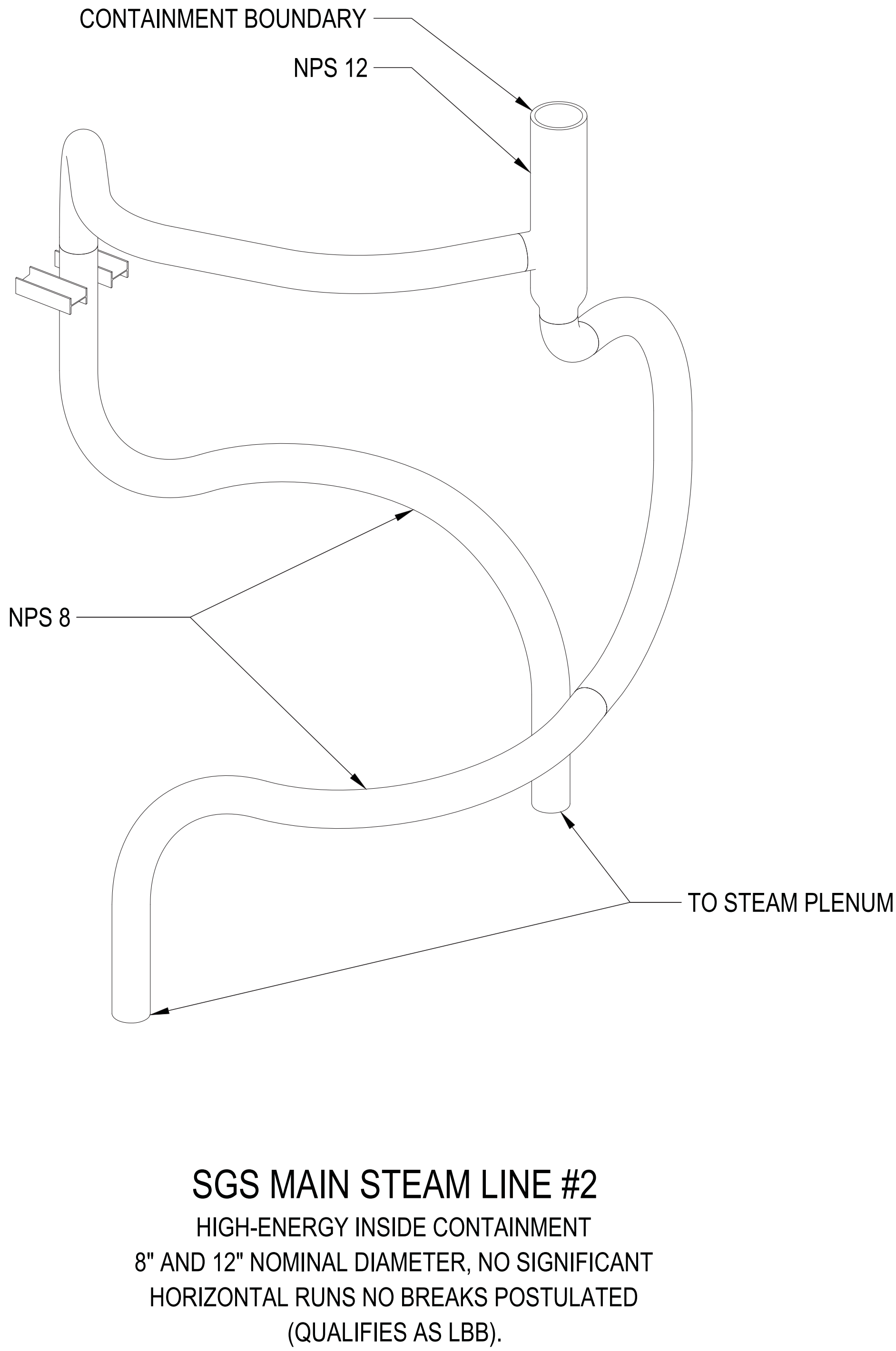


Figure 3.6-4: Feedwater Line 1

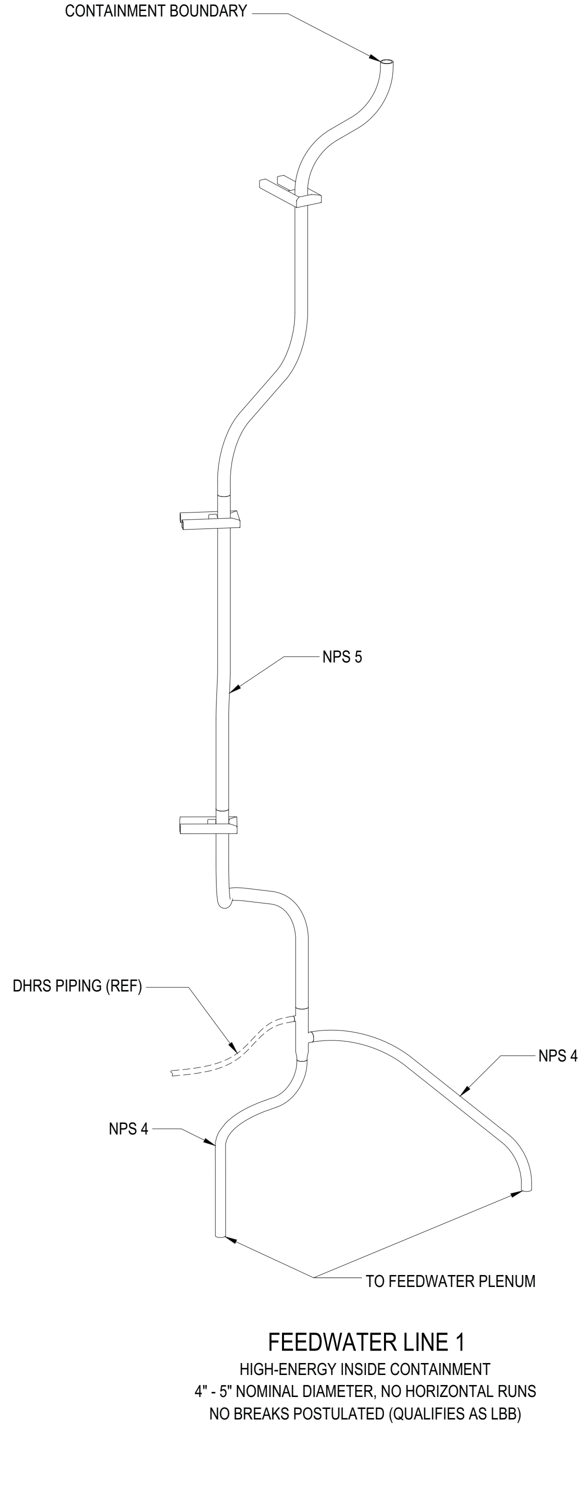
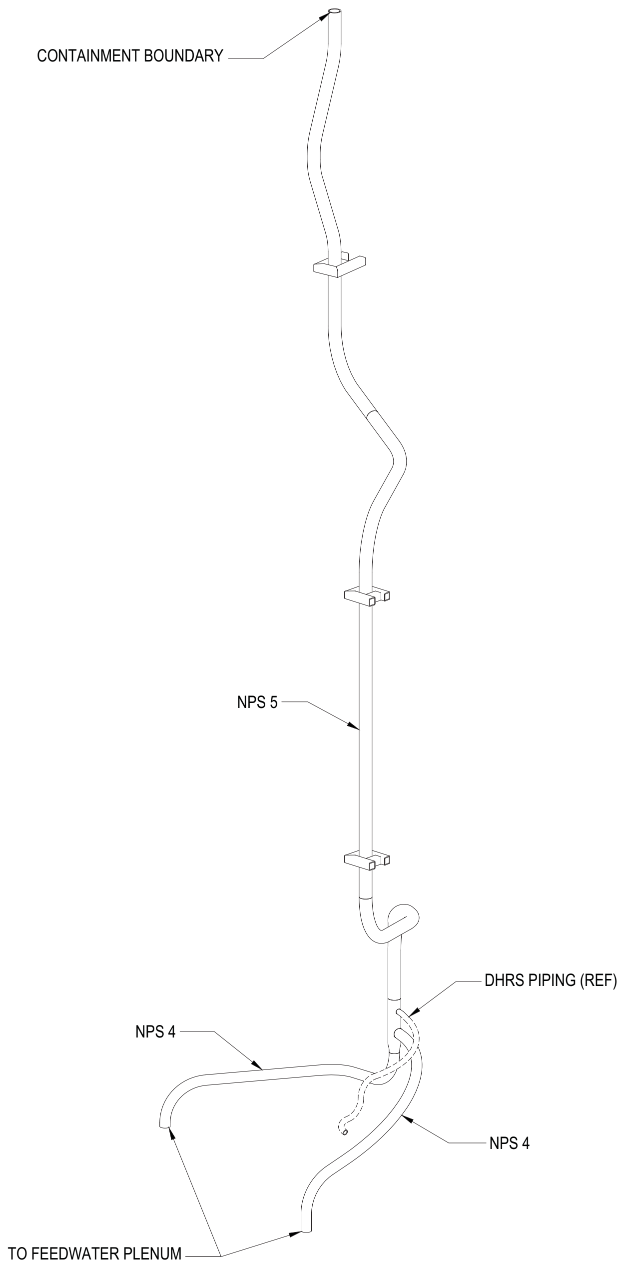


Figure 3.6-5: Feedwater Line 2



FEEDWATER LINE 2
HIGH-ENERGY INSIDE CONTAINMENT
4" - 5" NOMINAL DIAMETER, NO HORIZONTAL RUNS
NO BREAKS POSTULATED (QUALIFIES AS LBB)

Figure 3.6-6: Chemical and Volume Control System - Reactor Coolant System Injection Line Postulated Break Locations

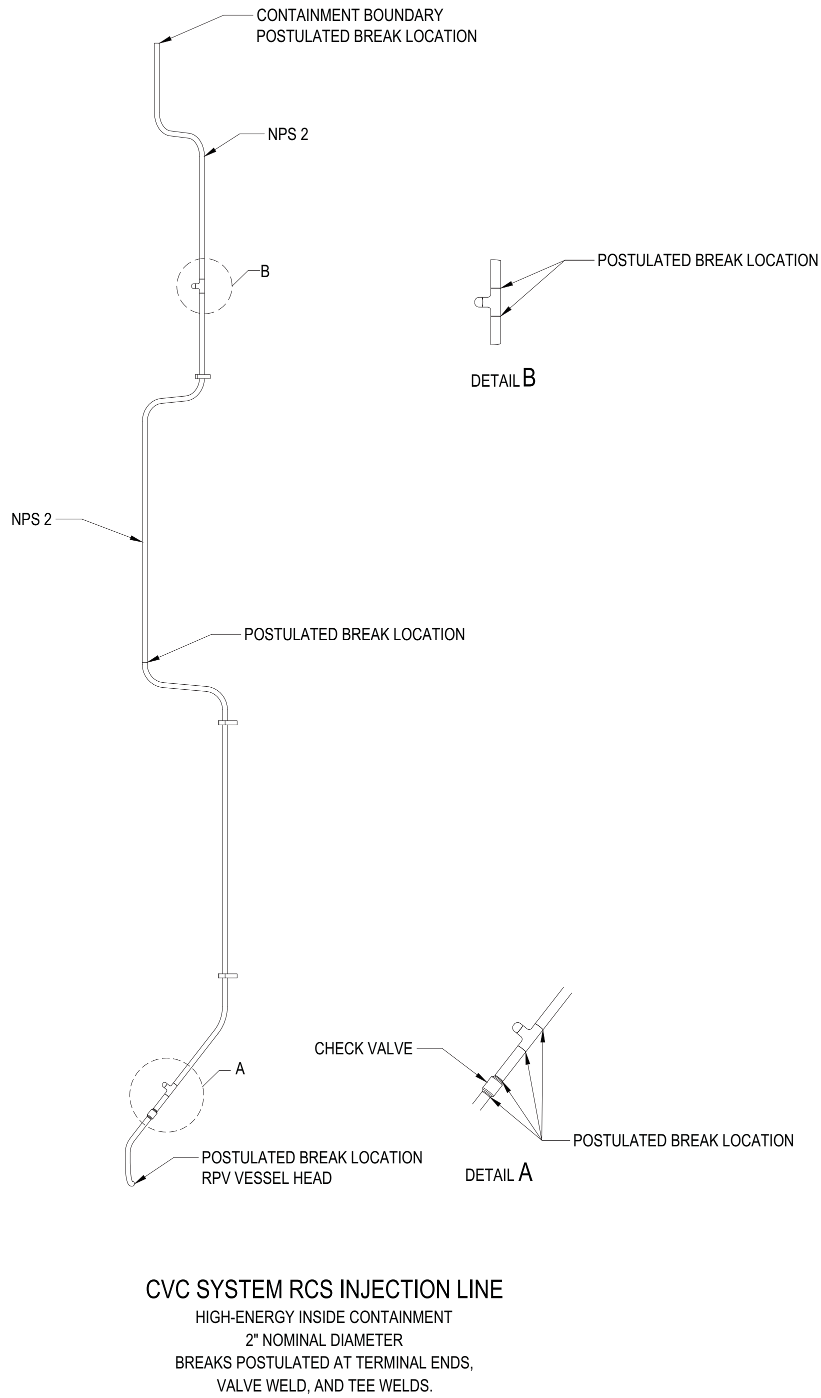


Figure 3.6-7: Chemical and Volume Control System - Reactor Coolant System Discharge Line Postulated Break Locations

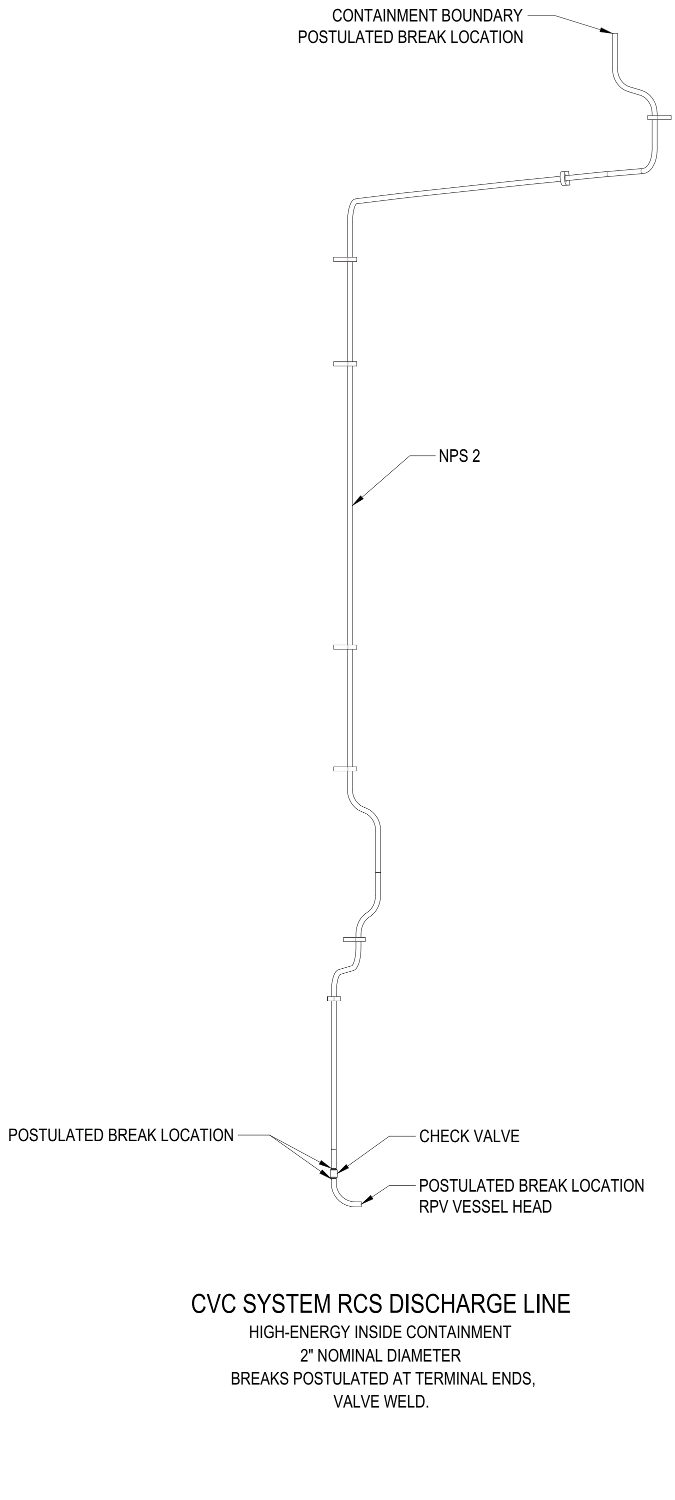


Figure 3.6-8: Chemical and Volume Control System - Pressurizer Spray Line Postulated Break Locations

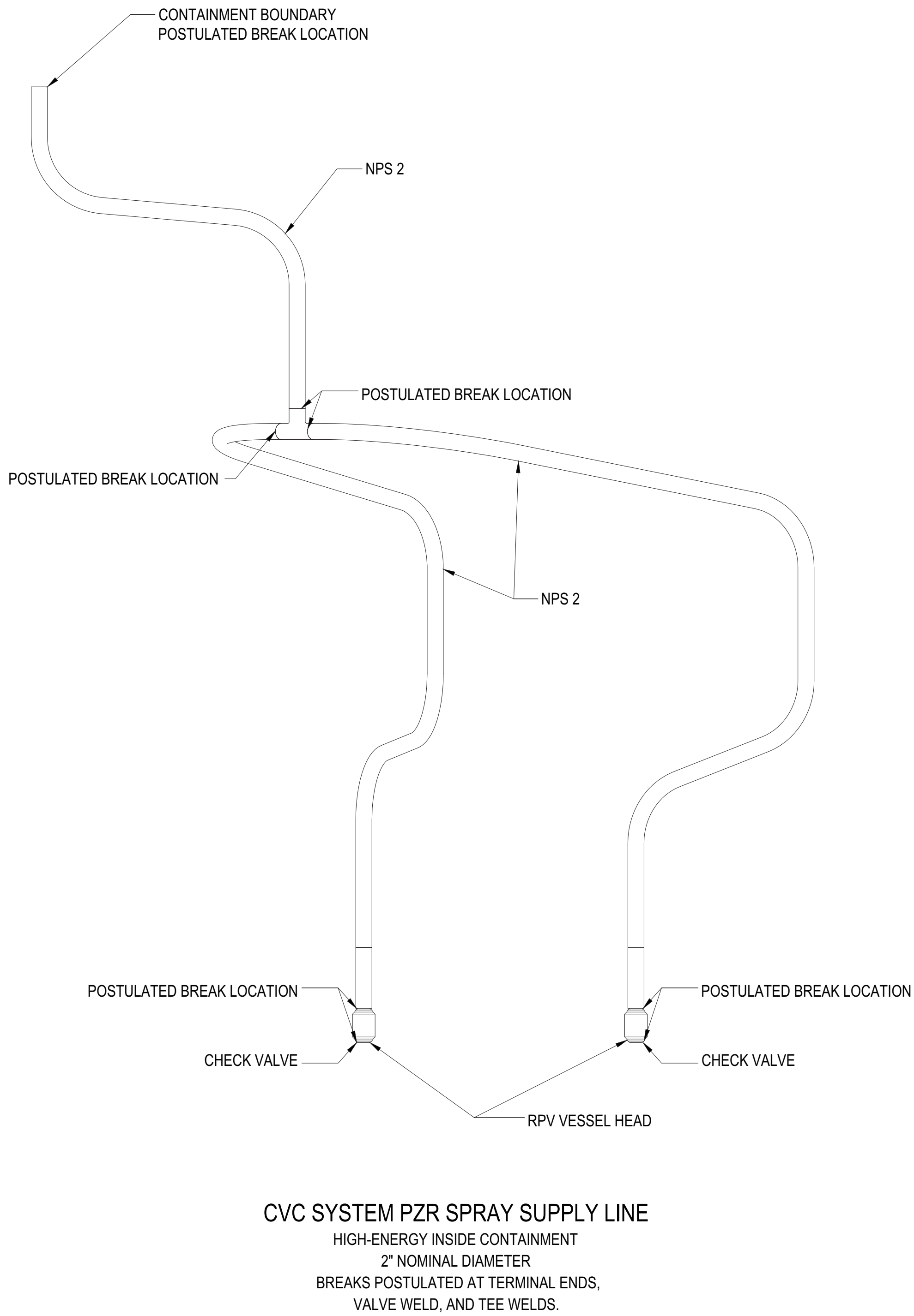


Figure 3.6-9: Chemical and Volume Control System - High Point Vent Postulated Break Locations

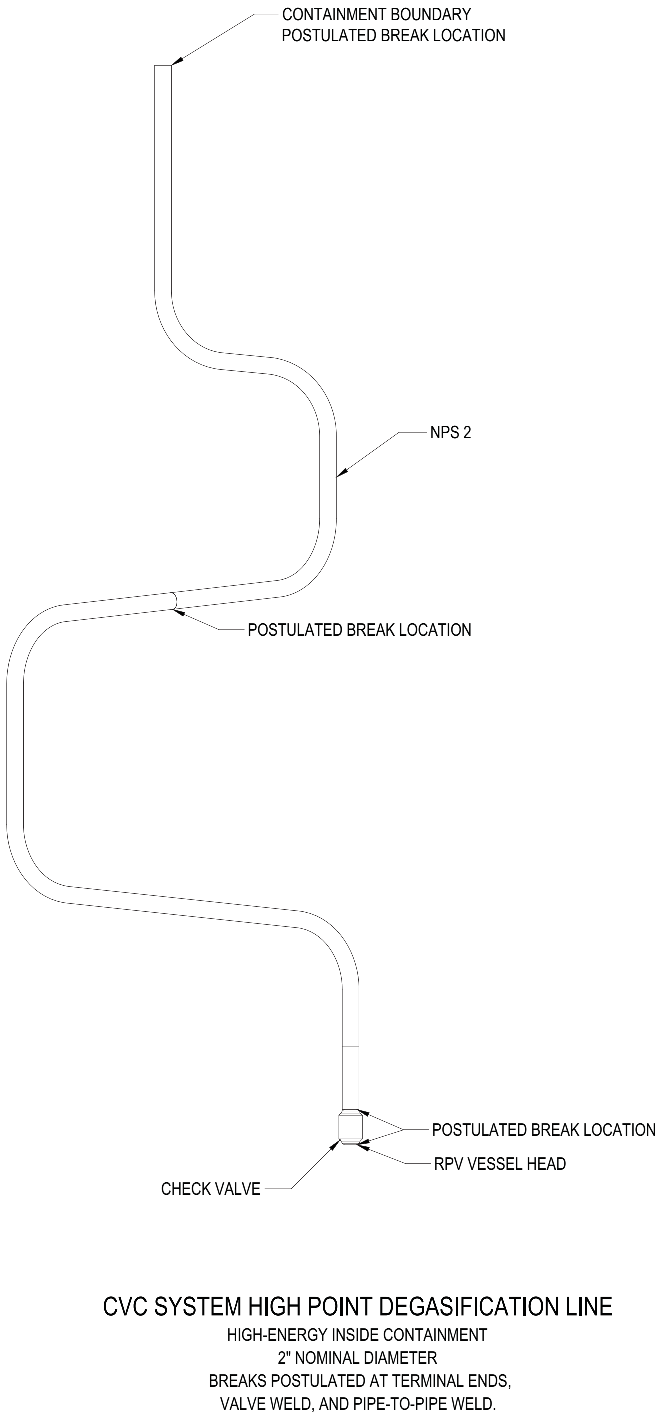
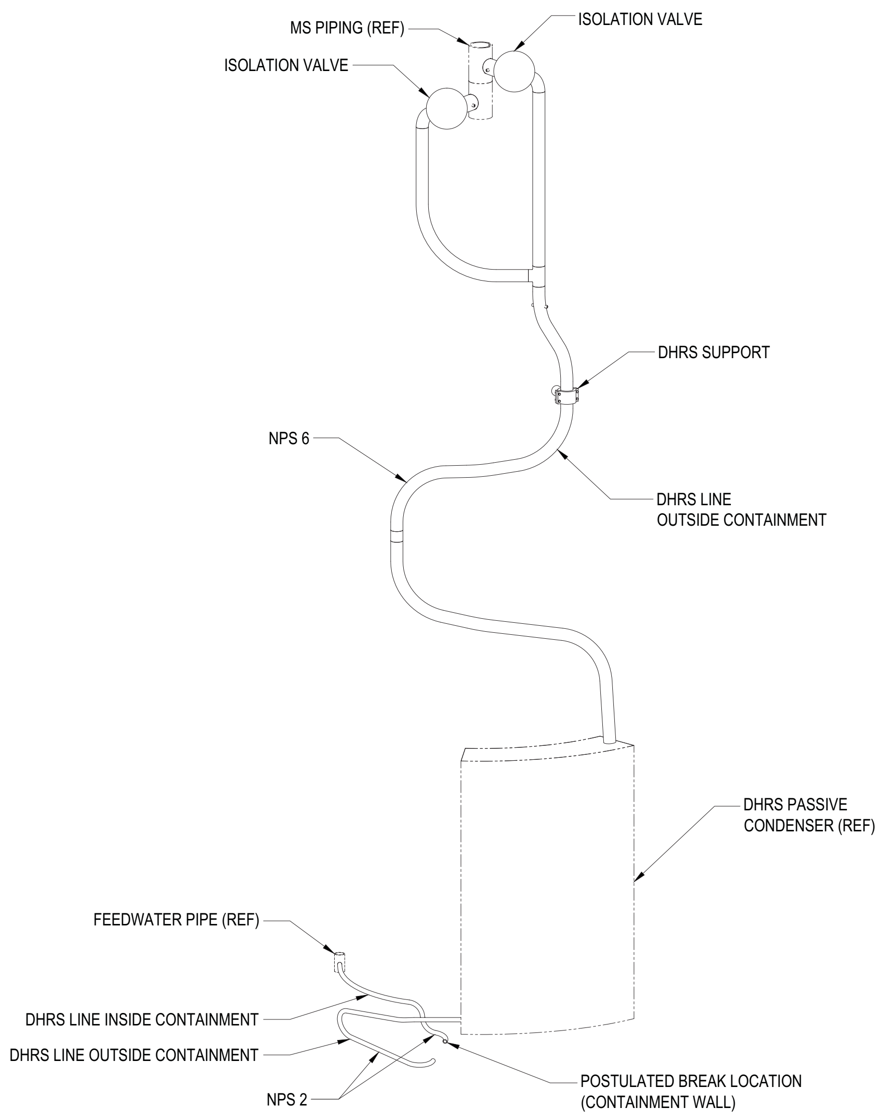
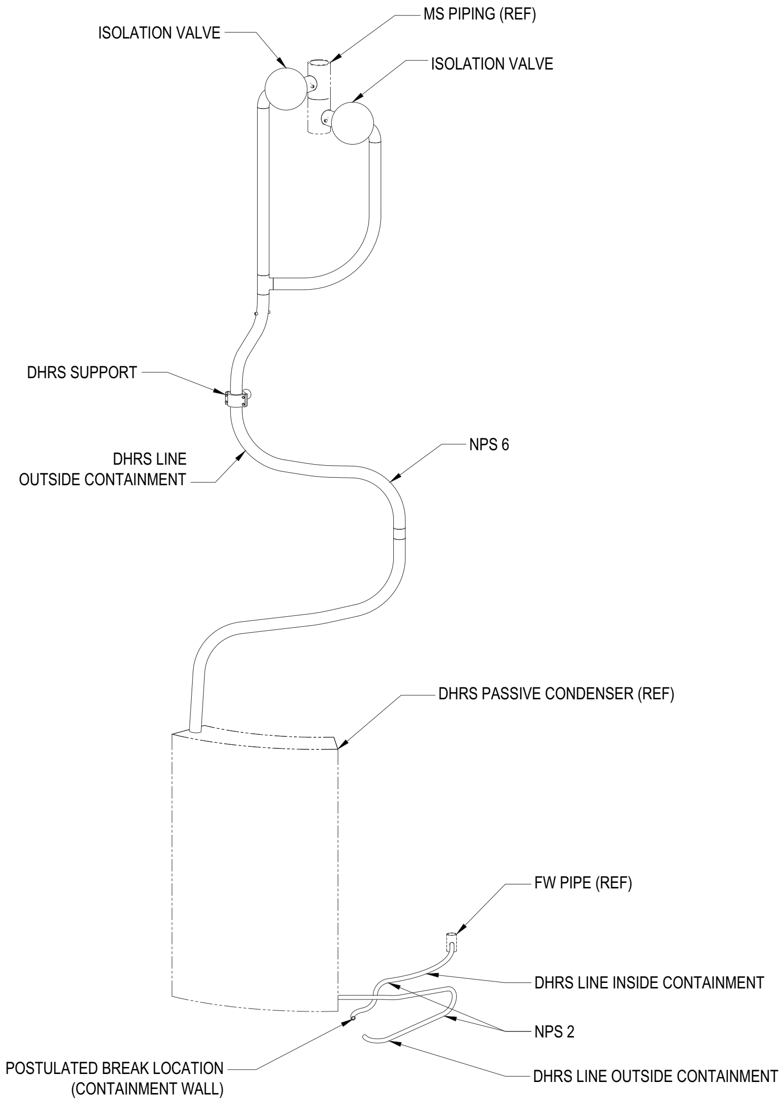


Figure 3.6-10: Decay Heat Removal System Line 1 Postulated Break Locations



DECAY HEAT REMOVAL SYSTEM LINE #1
HIGH-ENERGY INSIDE AND OUTSIDE CONTAINMENT
2" NOMINAL DIAMETER INSIDE CONTAINMENT
BREAKS (INSIDE CONTAINMENT) POSTULATED AT LOCATIONS
INDICATED. 2" AND 6" NOMINAL DIAMETER OUTSIDE
CONTAINMENT (NO BREAKS POSTULATED OUTSIDE CONTAINMENT,
PIPING QUALIFIES TO BTP 3-4 B.A. (ii)).

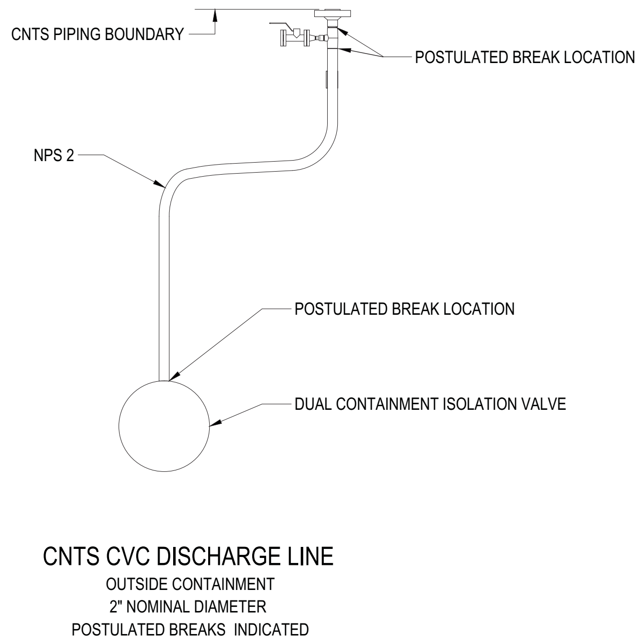
Figure 3.6-11: Decay Heat Removal System Line 2 Postulated Break Locations



DECAY HEAT REMOVAL SYSTEM LINE #2

HIGH-ENERGY INSIDE AND OUTSIDE CONTAINMENT
2" NOMINAL DIAMETER INSIDE CONTAINMENT
BREAKS (INSIDE CONTAINMENT) POSTULATED AT LOCATIONS INDICATED. 2" AND 6" NOMINAL DIAMETER OUTSIDE CONTAINMENT (NO BREAKS POSTULATED OUTSIDE CONTAINMENT, PIPING QUALIFIES TO BTP 3-4 B.A. (ii)).

Figure 3.6-12: Containment System Chemical and Volume Control Discharge and Injection Line Postulated Break Locations



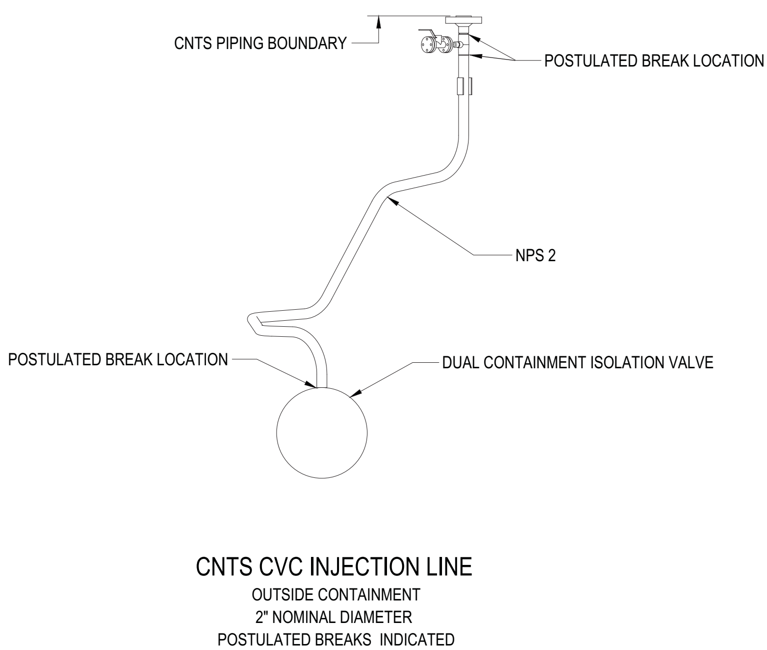
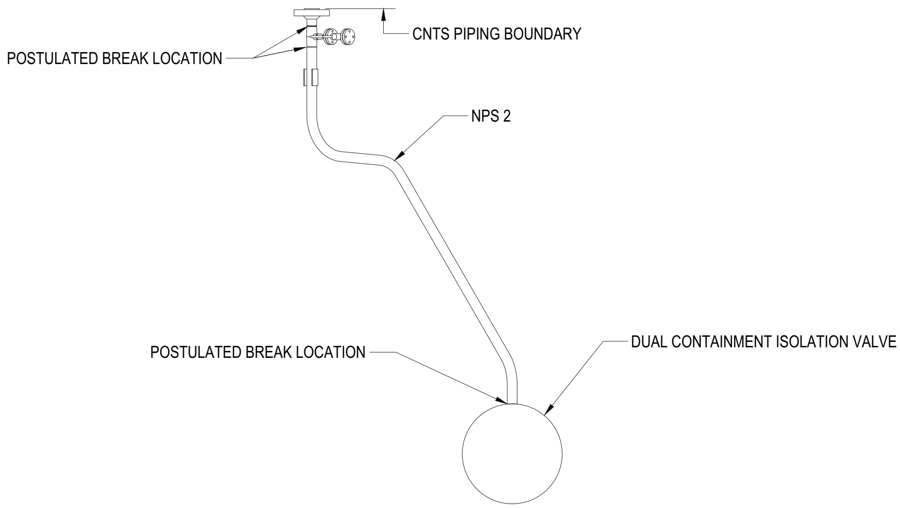
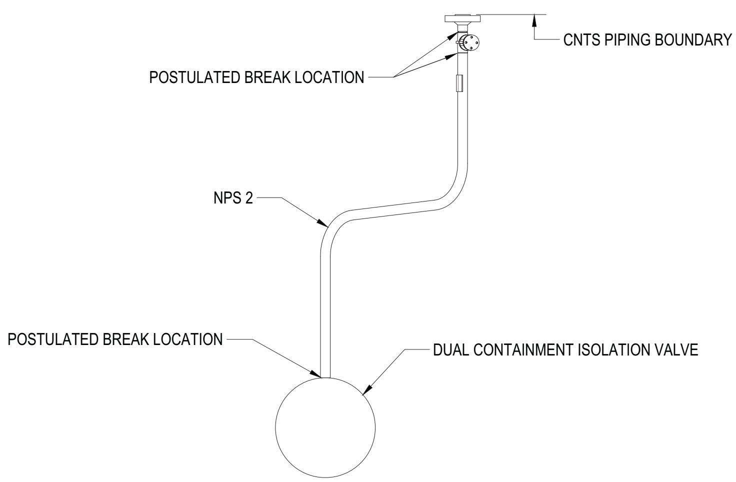


Figure 3.6-13: Chemical and Volume Control System Postulated Break Locations

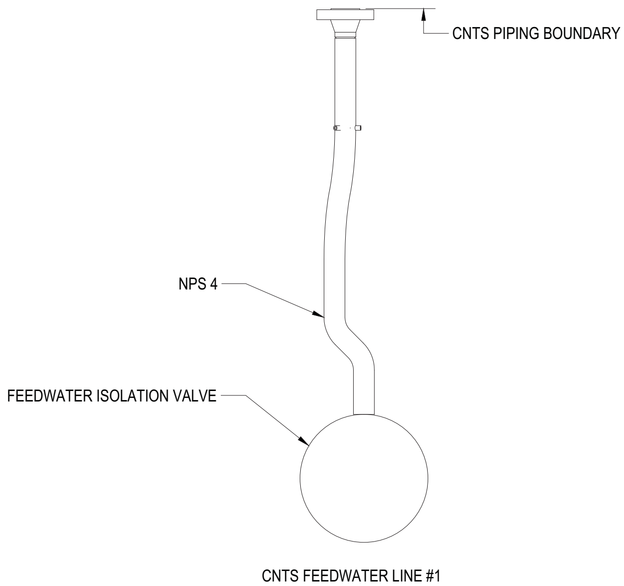


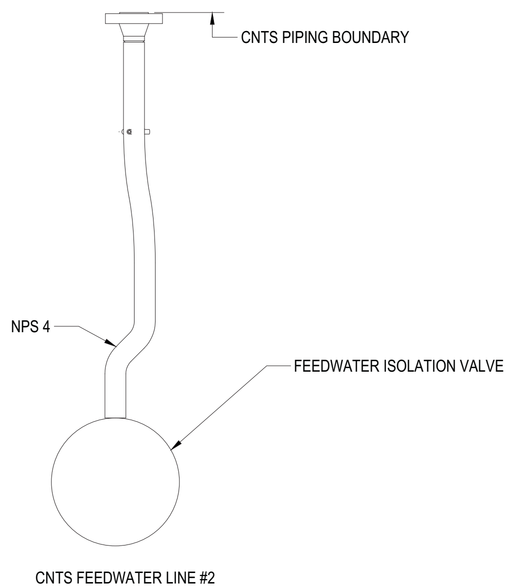
CNTS CVC PRESSURIZER SPRAY LINE
OUTSIDE CONTAINMENT
2" NOMINAL DIAMETER
POSTULATED BREAKS INDICATED



CNTS RPV HIGH POINT DEGASIFICATION LINE
OUTSIDE CONTAINMENT
2" NOMINAL DIAMETER
POSTULATED BREAKS INDICATED

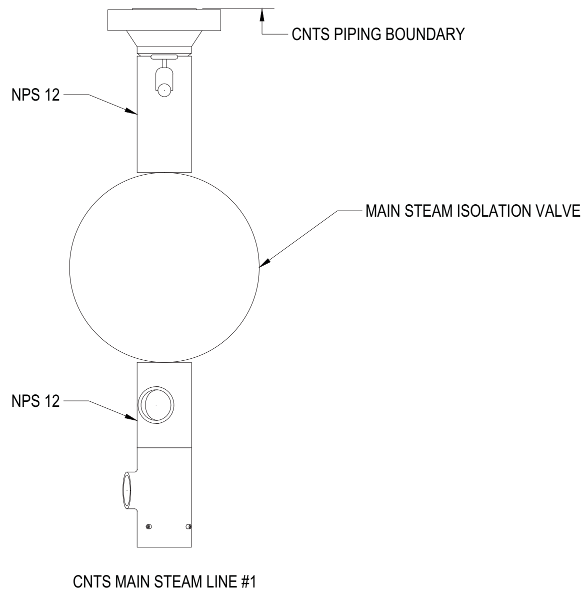
Figure 3.6-14: Feedwater Line Postulated Break Locations

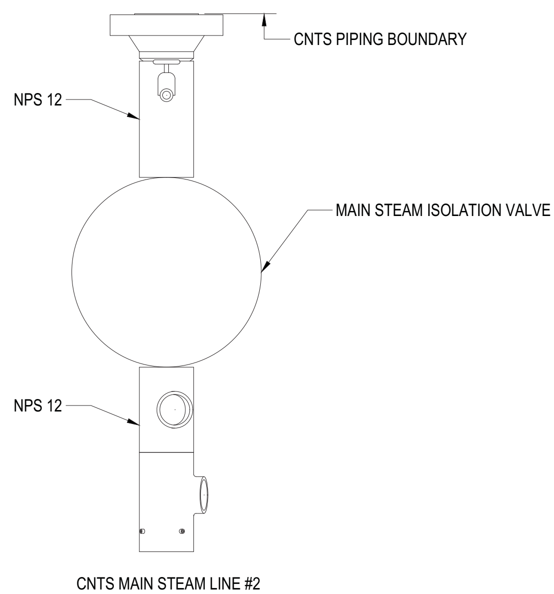




CNTS FEEDWATER LINE #1 AND LINE #2
OUTSIDE CONTAINMENT
4" NOMINAL DIAMETER
(NO BREAKS POSTULATED OUTSIDE CONTAINMENT,
PIPING QUALIFIES TO BTP 3-4 B.A. (ii))

Figure 3.6-15: Main Steam Line Postulated Break Locations





HIGH ENERGY OUTSIDE CONTAINMENT
12" NOMINAL DIAMETER
(NO BREAKS POSTULATED OUTSIDE CONTAINMENT,
PIPING QUALIFIES TO BTP 3-4 B.A. (ii))

Figure 3.6-16: Postulated High-Energy Main Steam System Pipe Routing Beyond the NuScale Power Module (COL Applicant Scope)

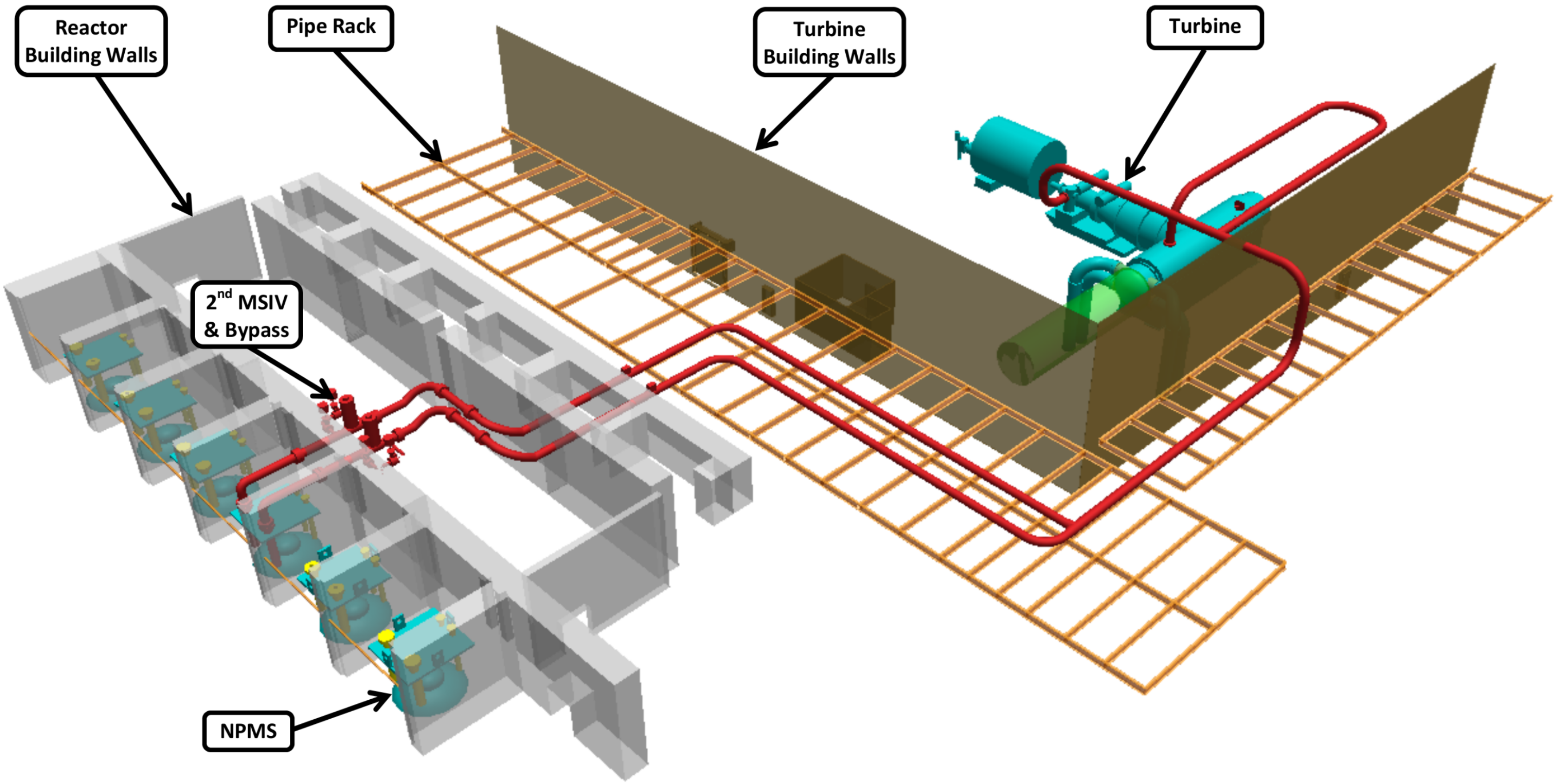


Figure 3.6-17: Postulated High-Energy Feedwater System Pipe Routing Beyond the NuScale Power Module (COL Applicant Scope)

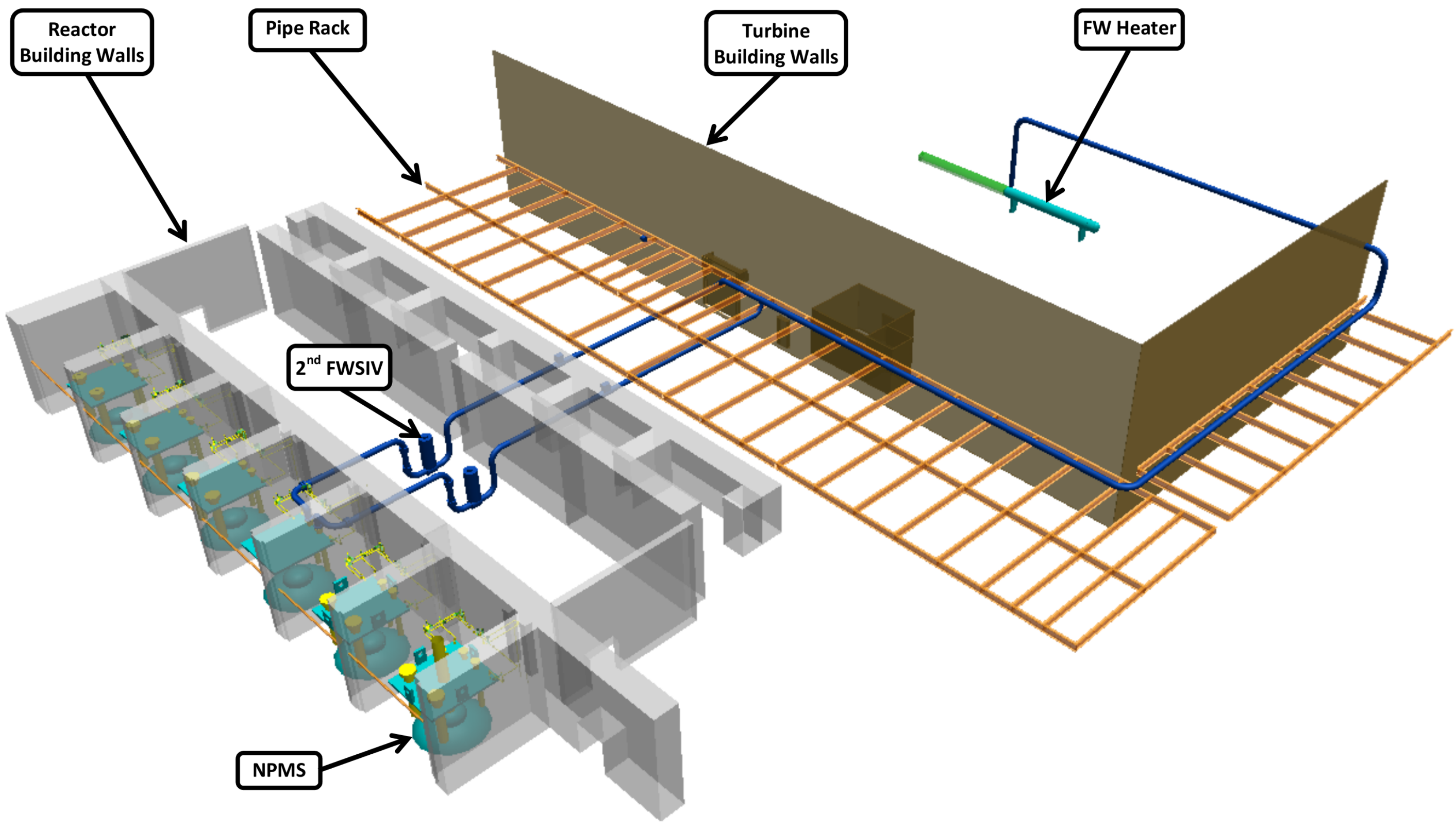


Figure 3.6-18: Flow Chart for Piping Leak-Before-Break Evaluation

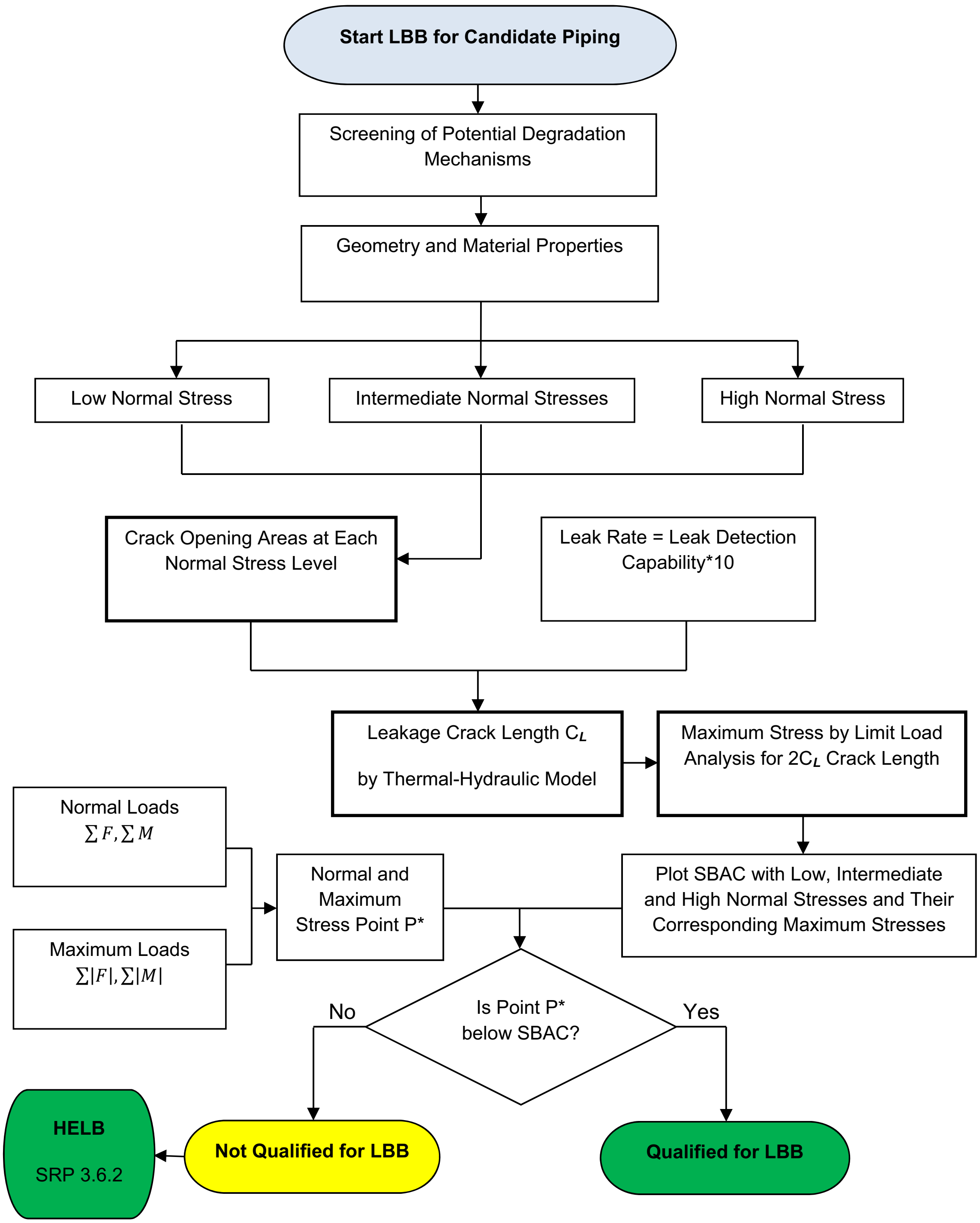


Figure 3.6-19: Illustration of Pipe with a Circumferential Through-Wall Crack

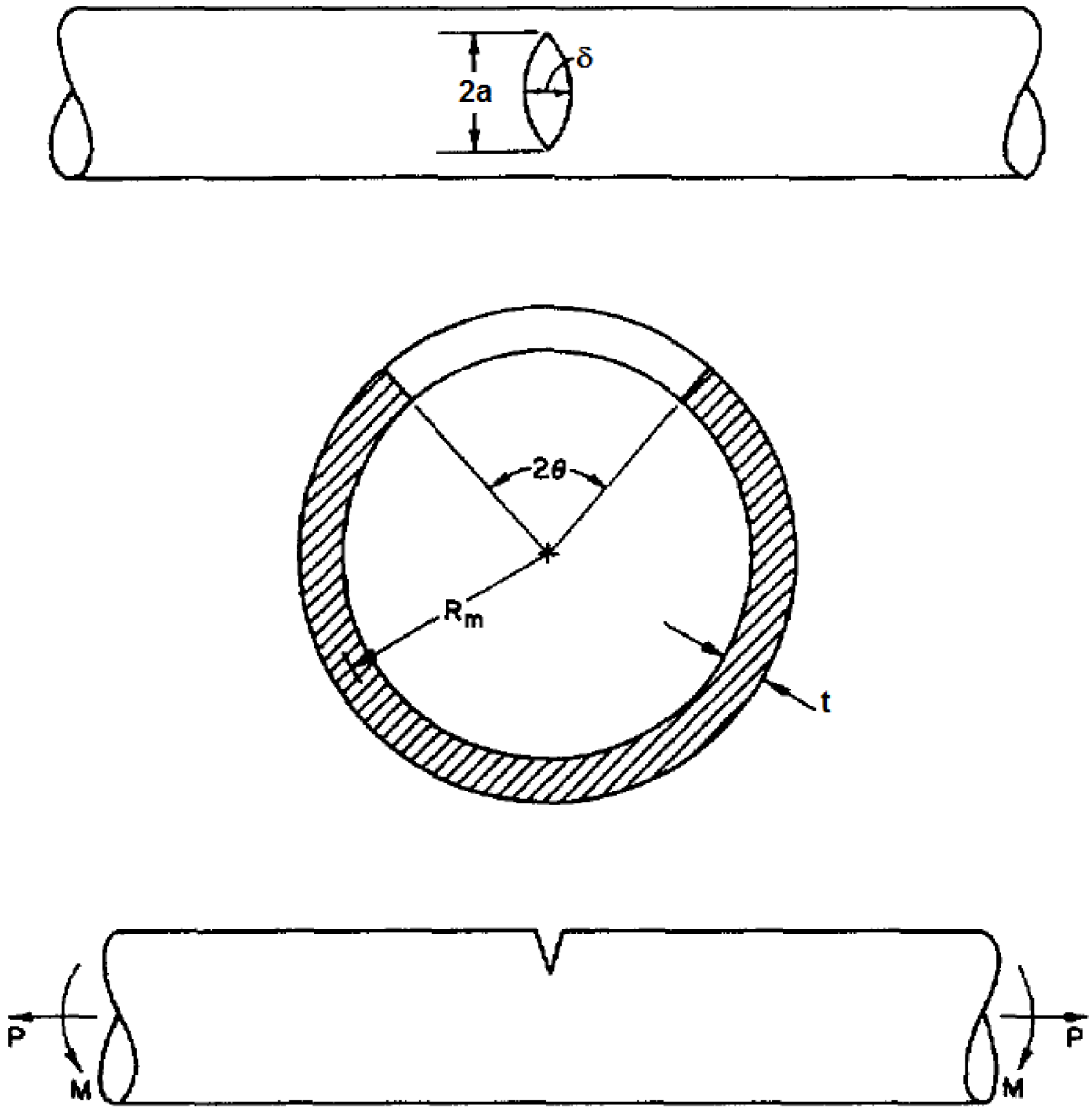


Figure 3.6-20: Henry-Fauske's Model of Two-Phase Flow

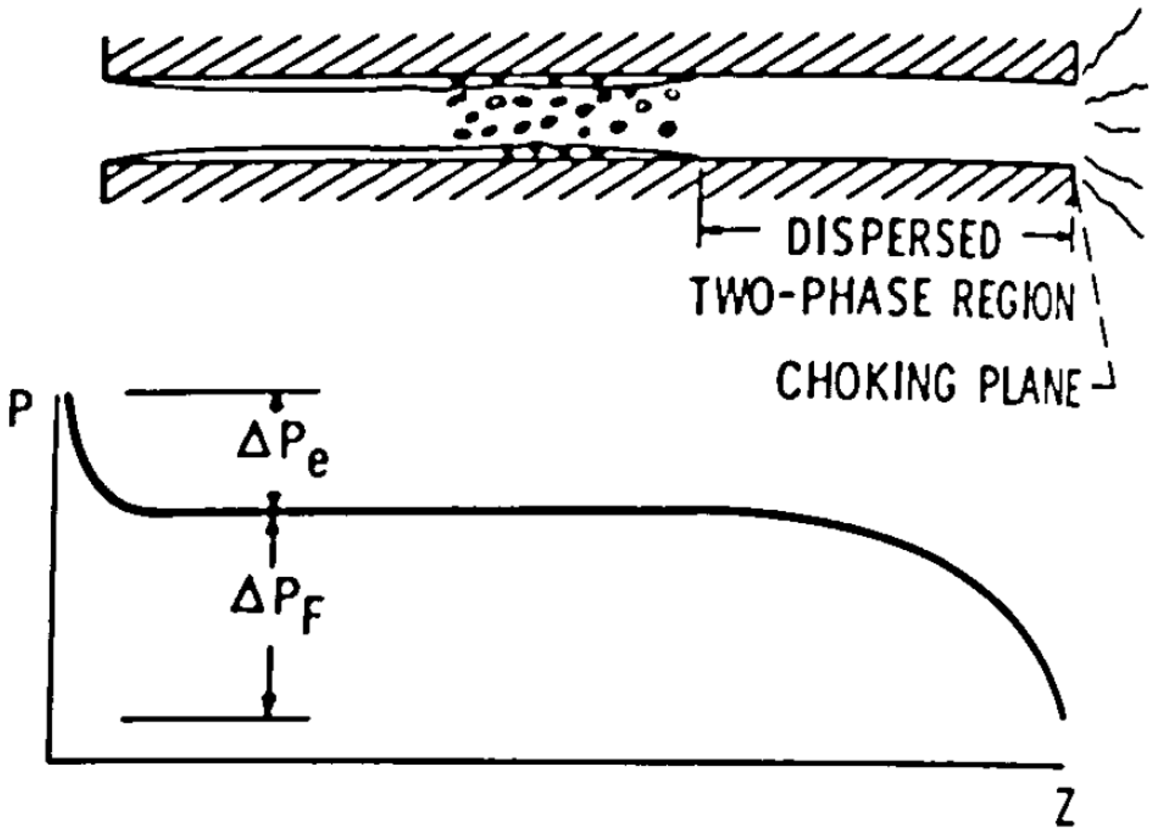


Figure 3.6-21: Local and Global Surface Roughness and Turns

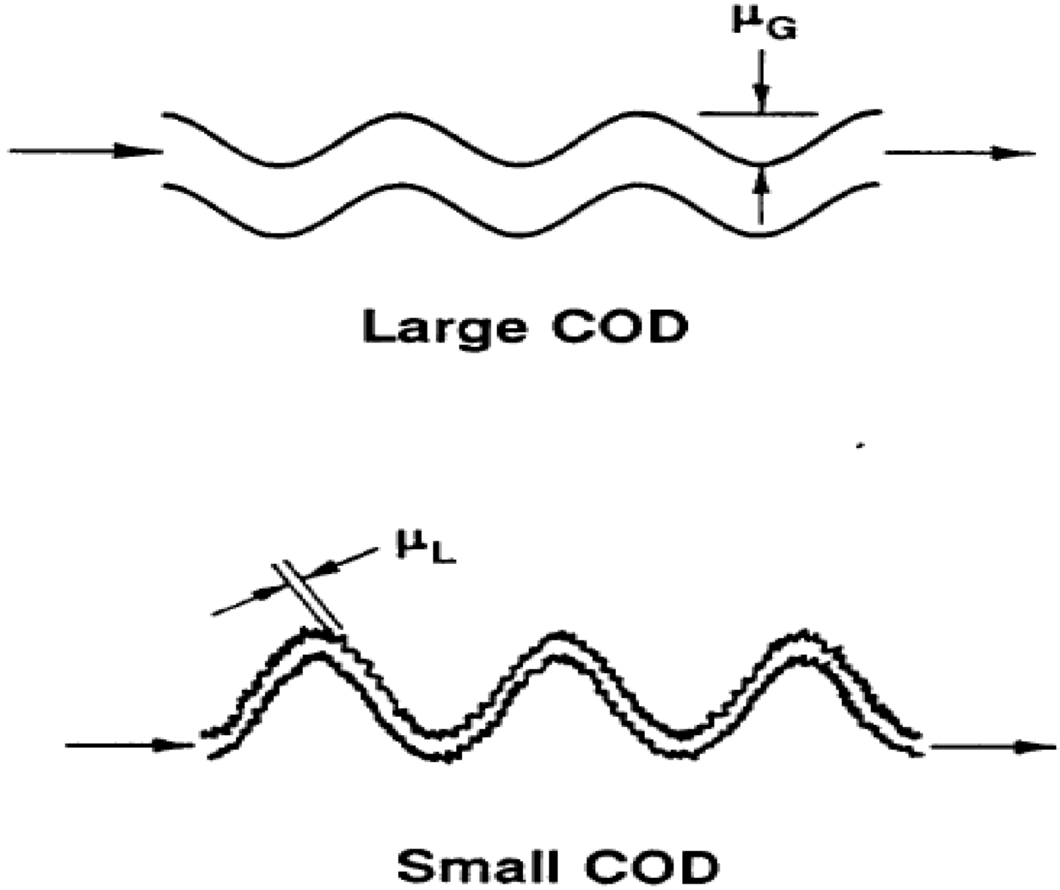


Figure 3.6-22: Crack Opening Displacement-Dependent Effective Crack Morphology

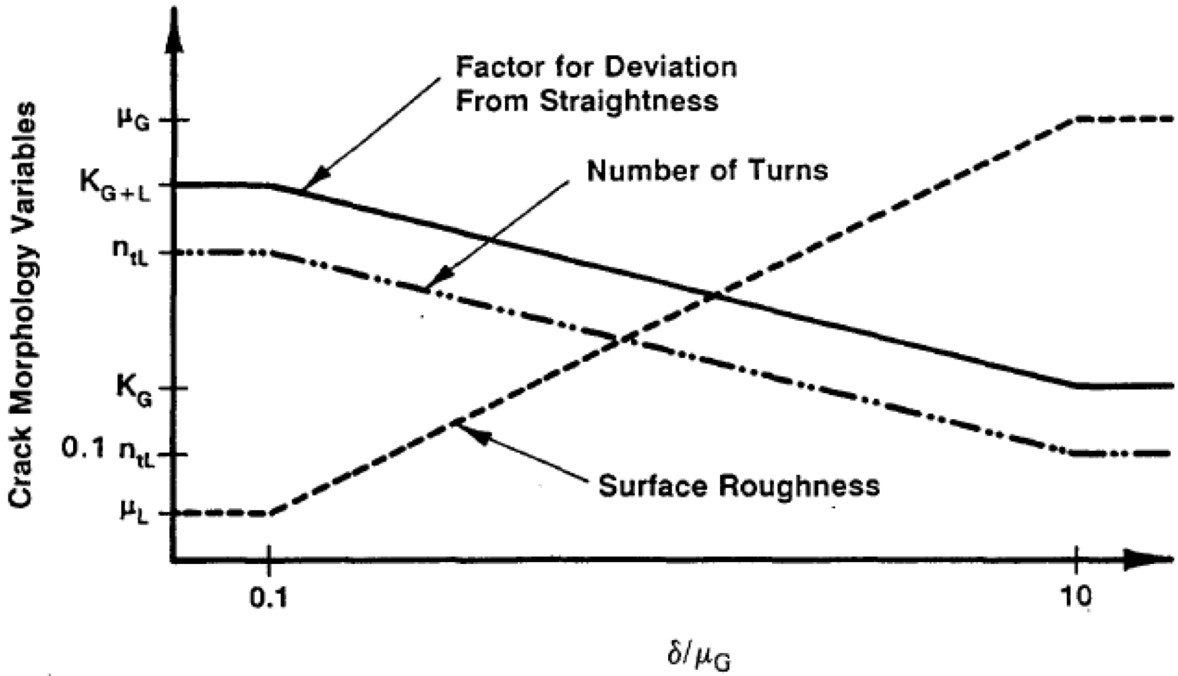


Figure 3.6-23: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 8 Straight Pipe Base Metal

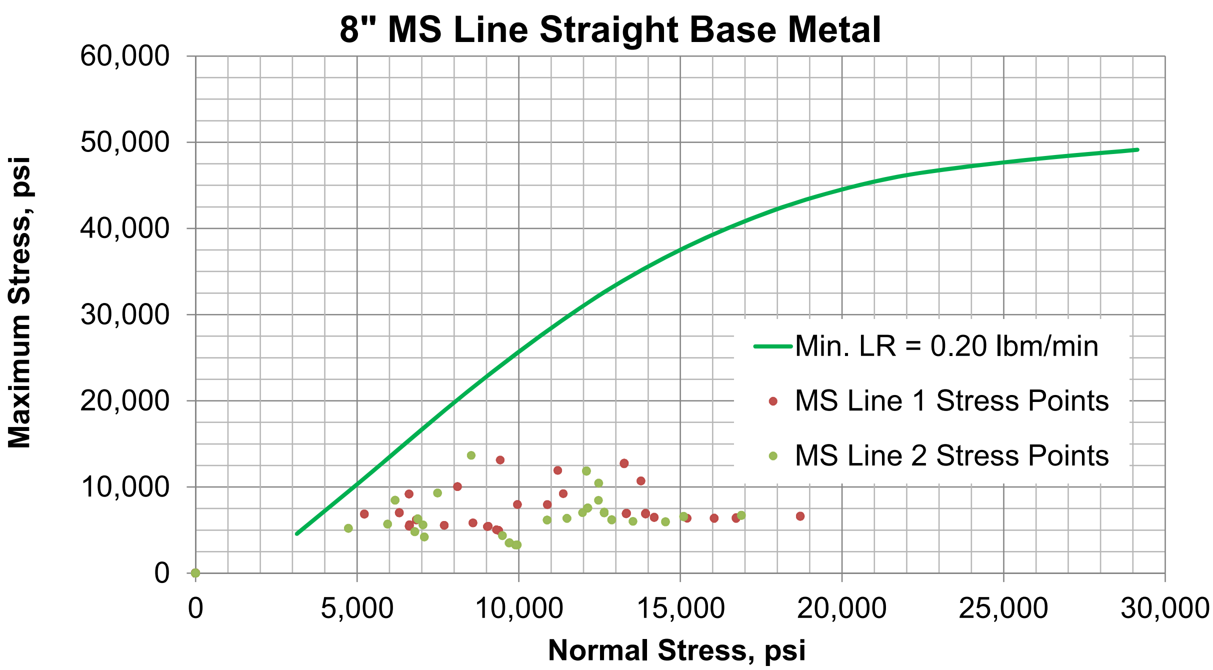


Figure 3.6-24: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 8 Pipe-to-Pipe Weld

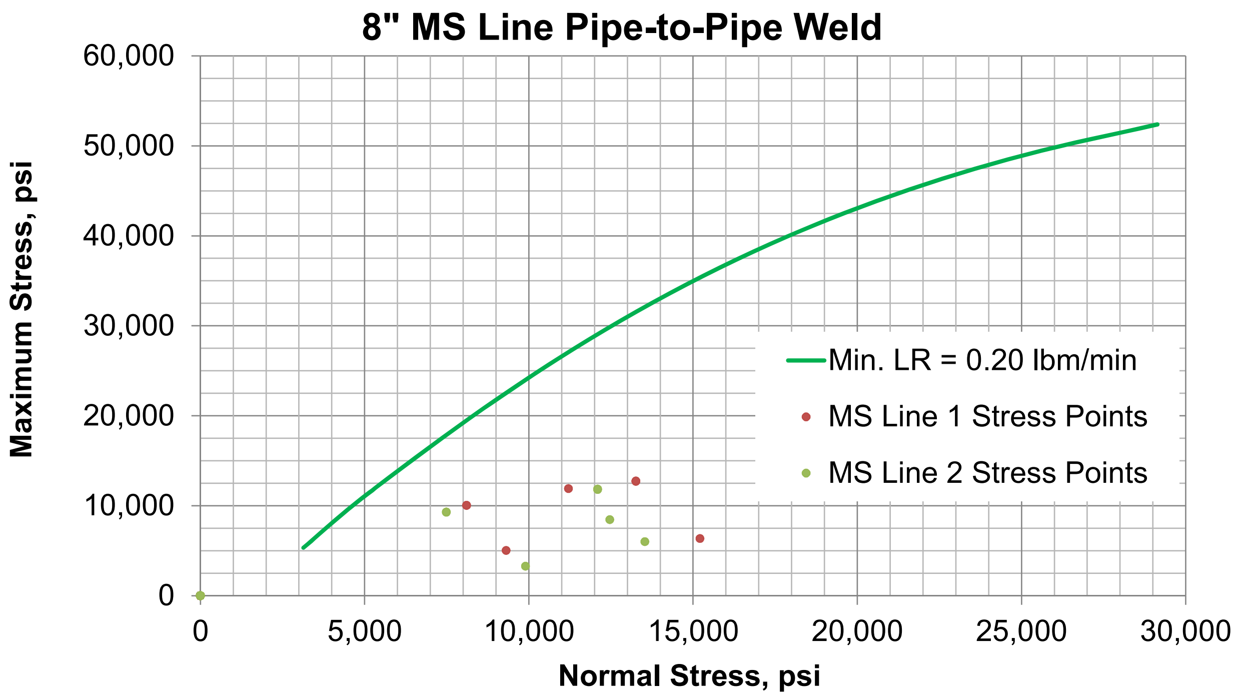


Figure 3.6-25: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 8 Pipe-to-Safe-End Weld

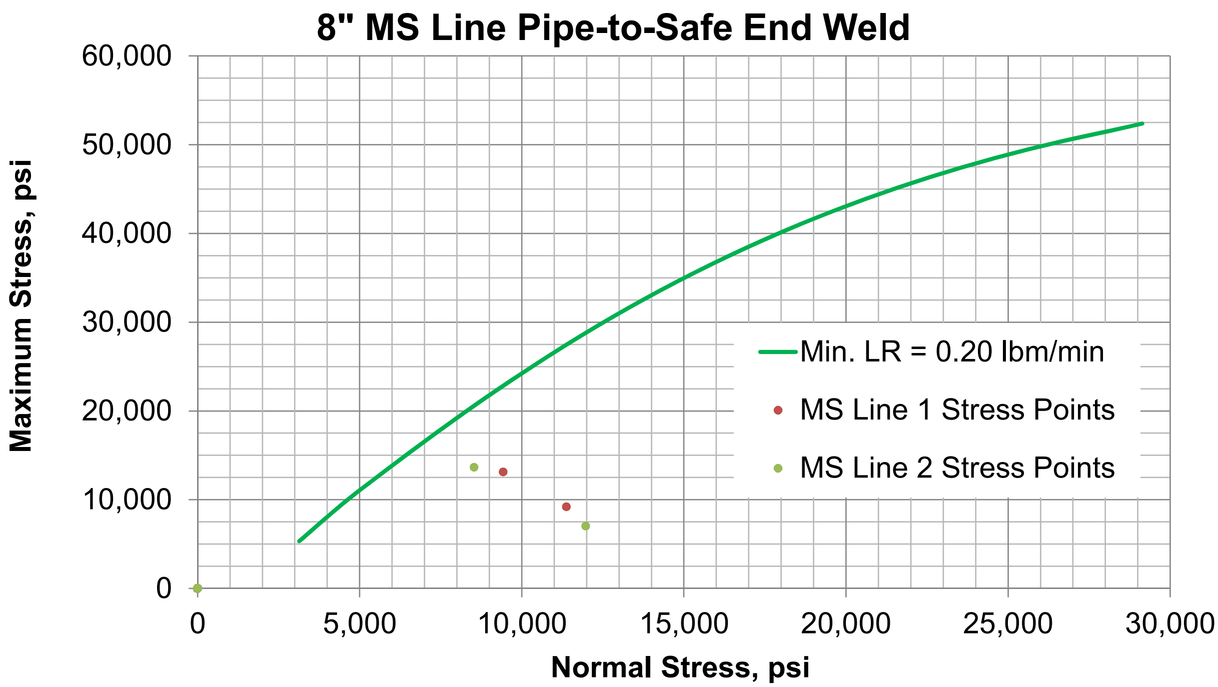


Figure 3.6-26: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 12 Straight Pipe Base Metal

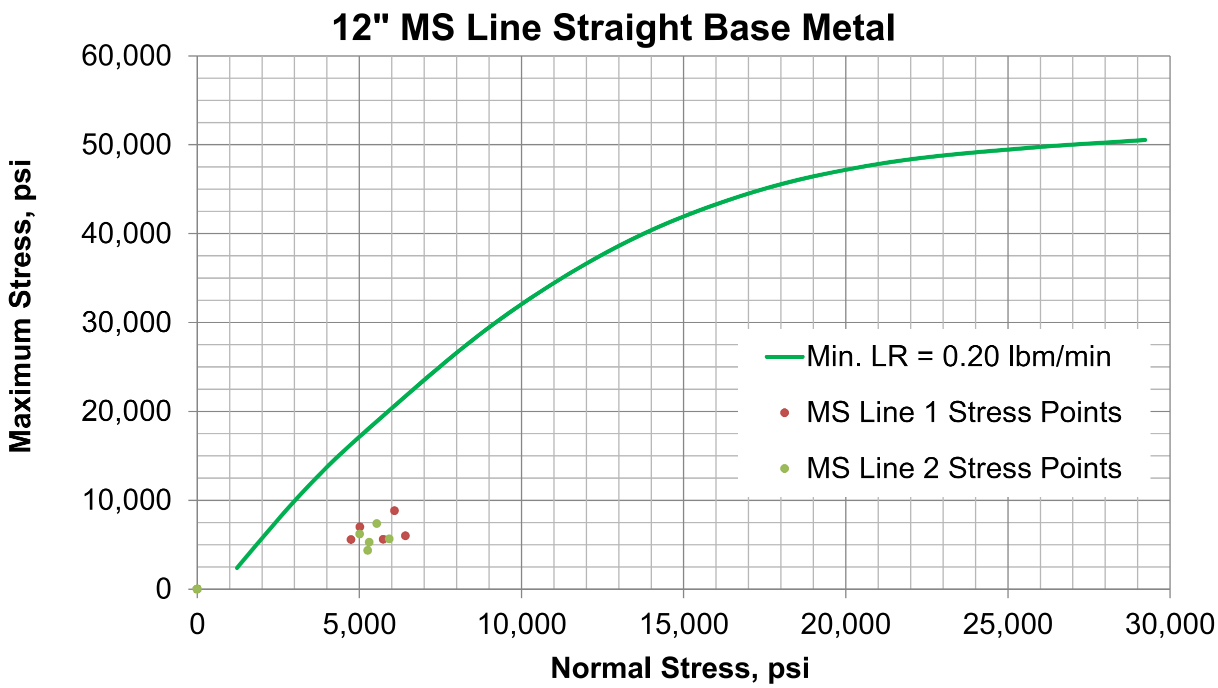


Figure 3.6-27: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 12 Pipe-to-Safe-End Weld

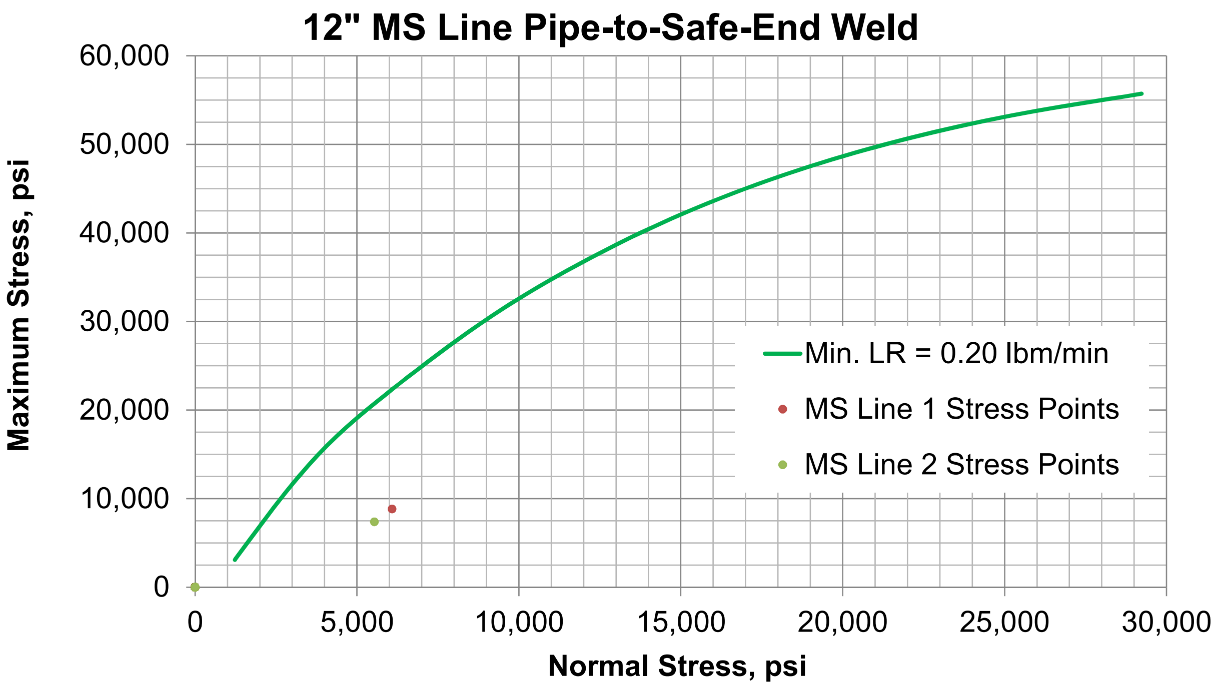


Figure 3.6-28: Smooth Bounding Analysis Curve for Main Steam System Nominal Pipe Size 8 Elbow Base Metal

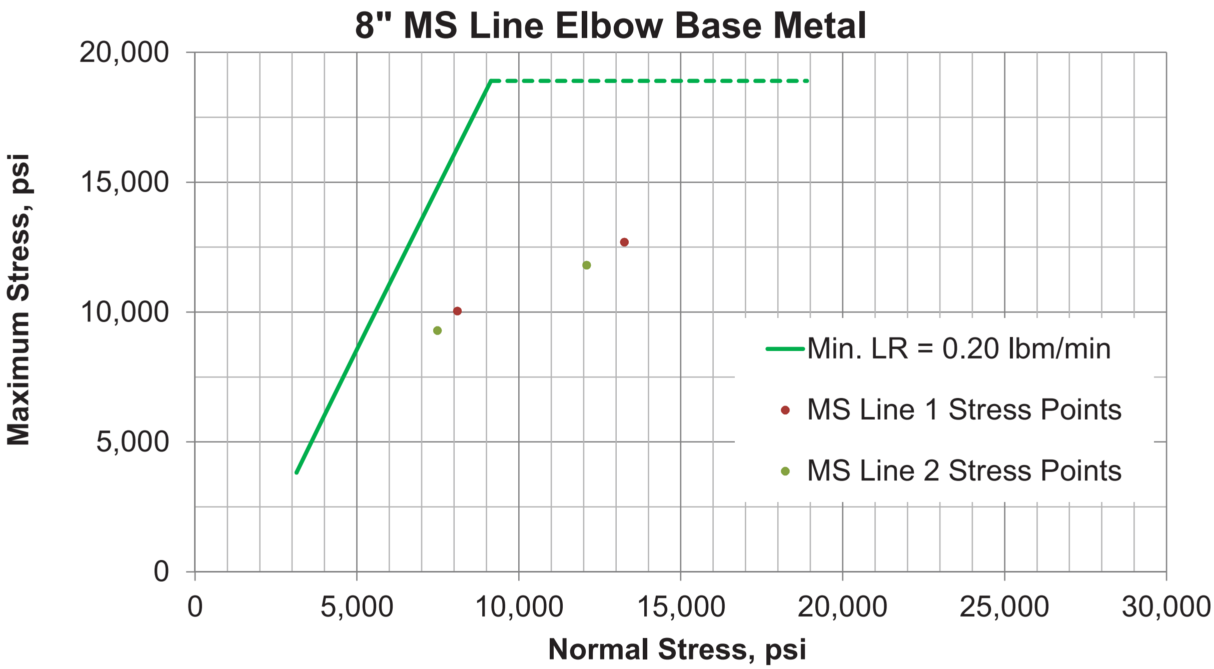


Figure 3.6-29: Smooth Bounding Analysis Curve for Nominal Pipe Size 4 Feedwater System Line Base Metal

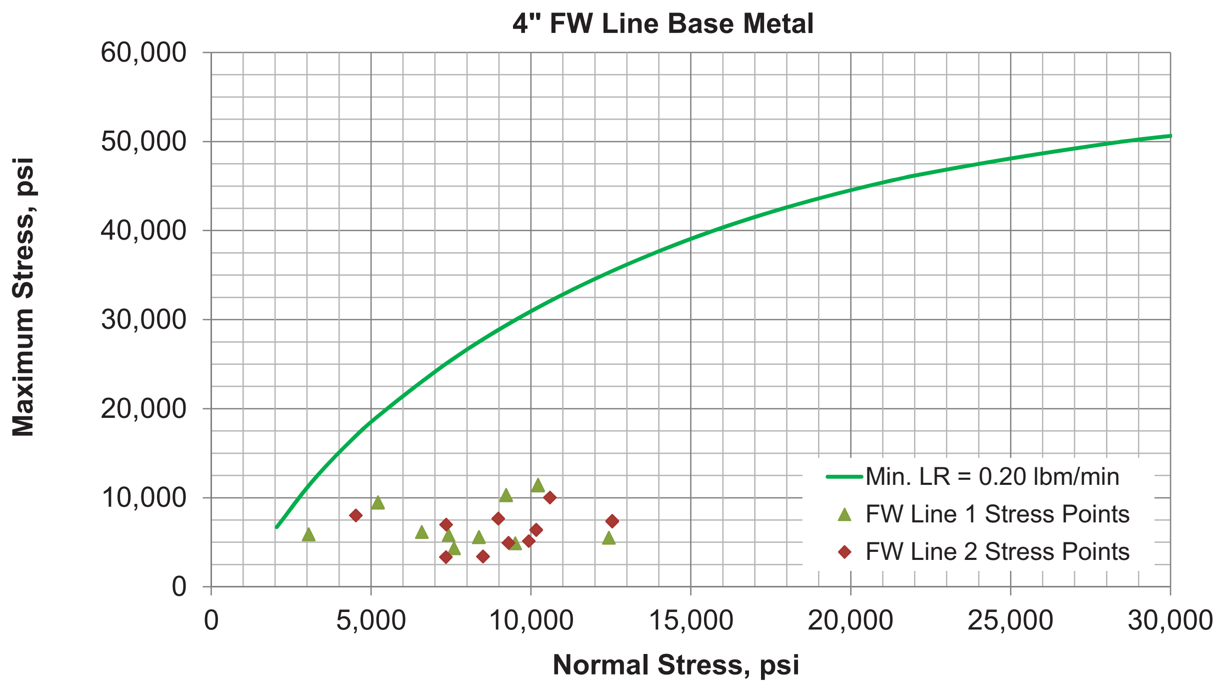


Figure 3.6-30: Smooth Bounding Analysis Curve for Nominal Pipe Size 4 Feedwater System Line Welds

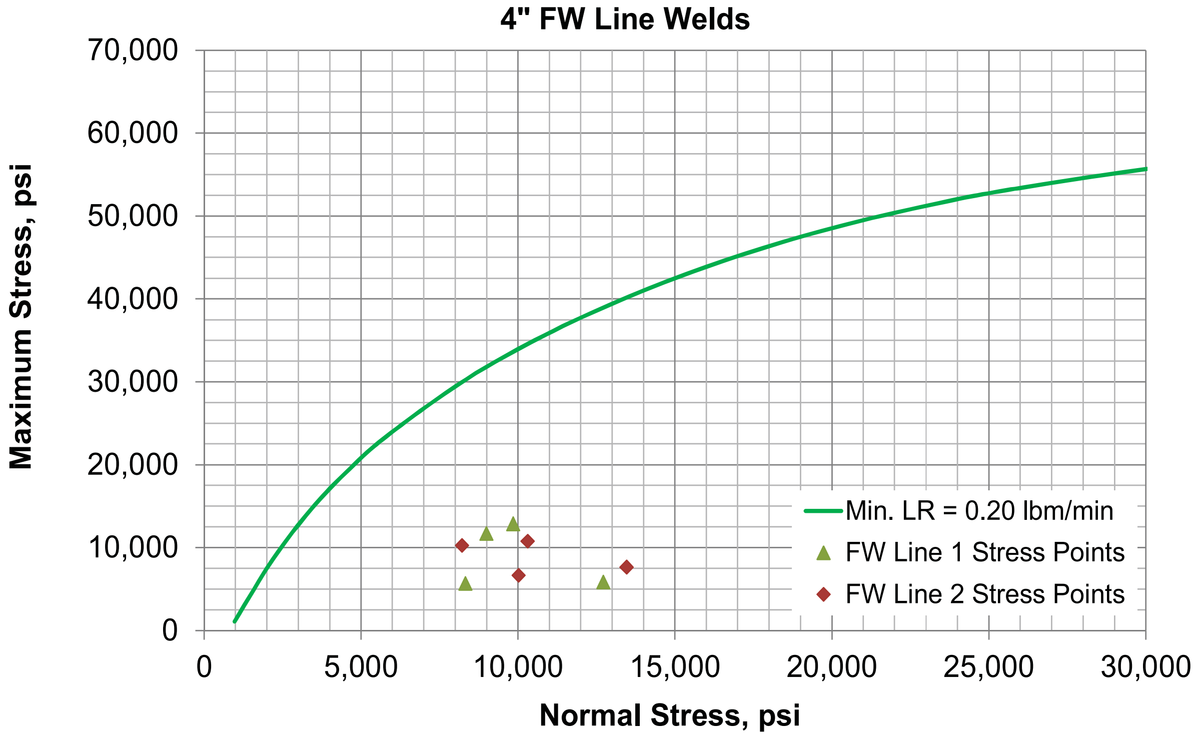


Figure 3.6-31: Smooth Bounding Analysis Curve for Nominal Pipe Size 5 Feedwater System Line Base Metal

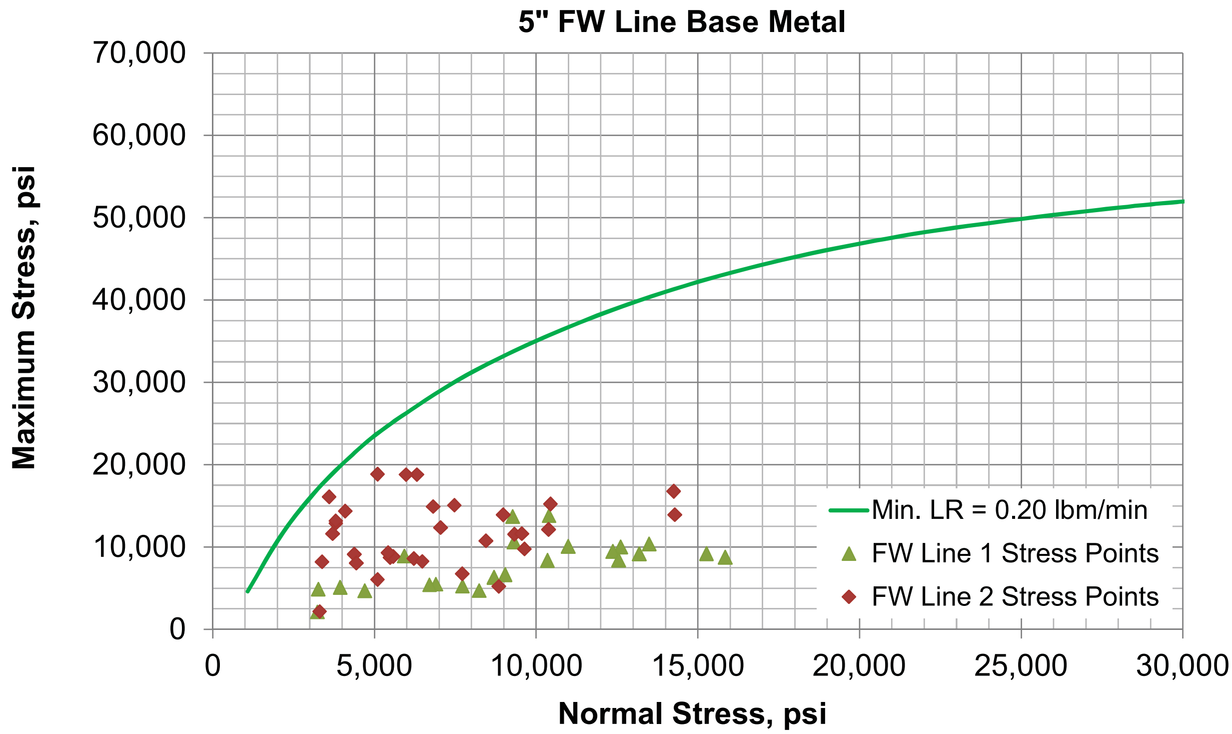


Figure 3.6-32: Smooth Bounding Analysis Curve for Nominal Pipe Size 5 Feedwater System Line Welds

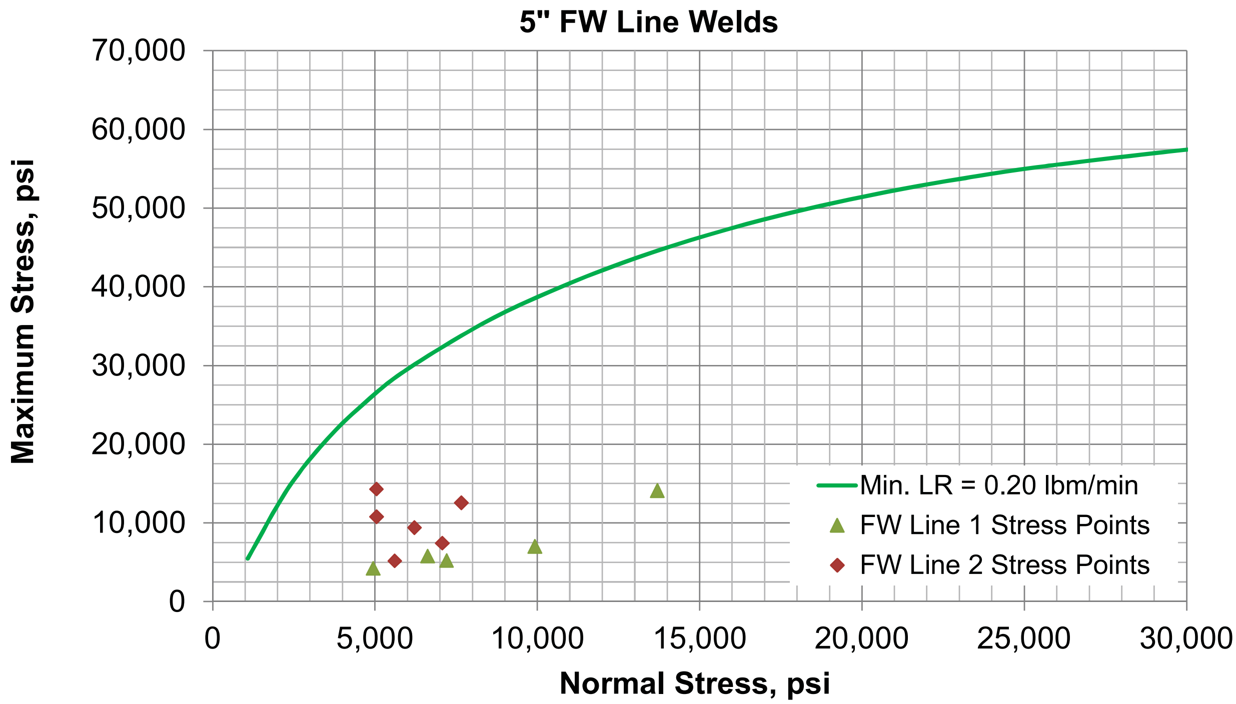


Figure 3.6-33: Typical Integral Jet Impingement Shield and Pipe Whip Restraint

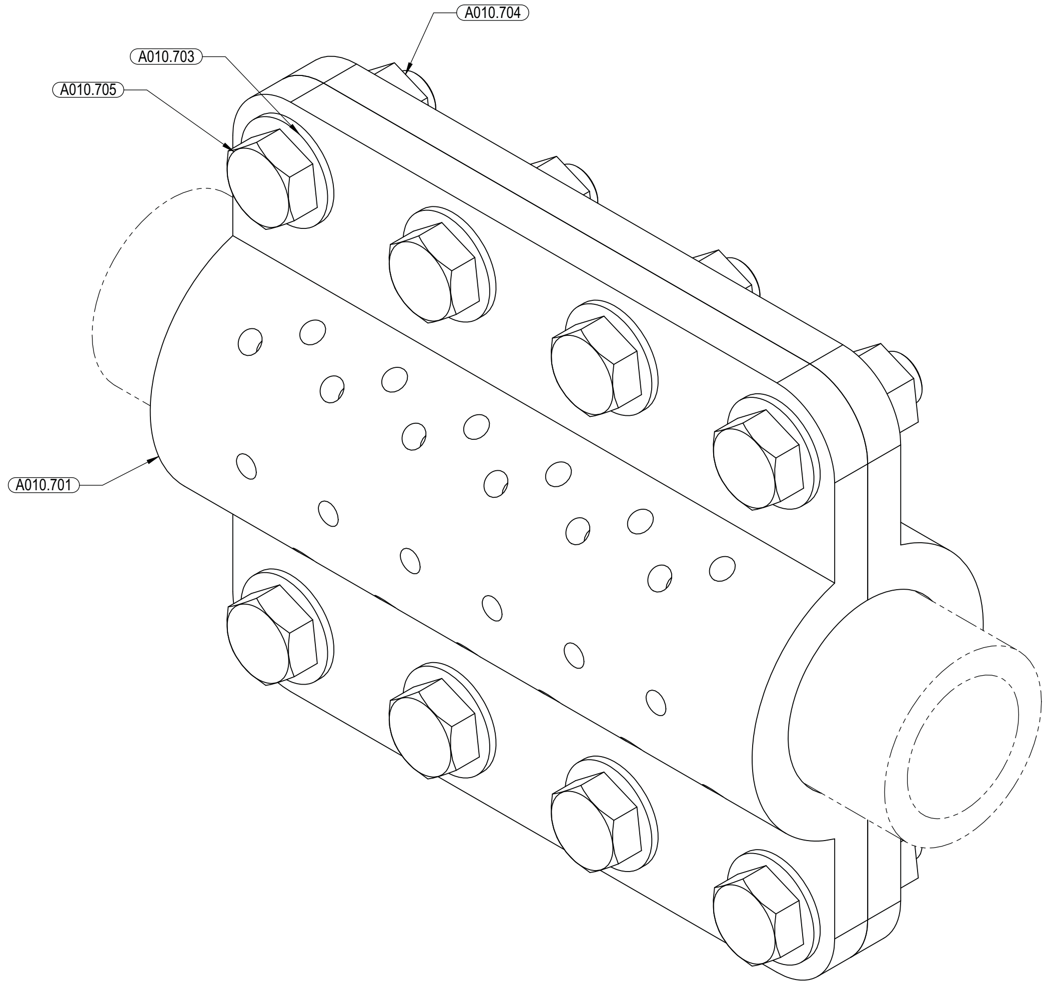


Figure 3.6-34: Cutaway View of Integral Jet Impingement Shield and Pipe Whip Restraint

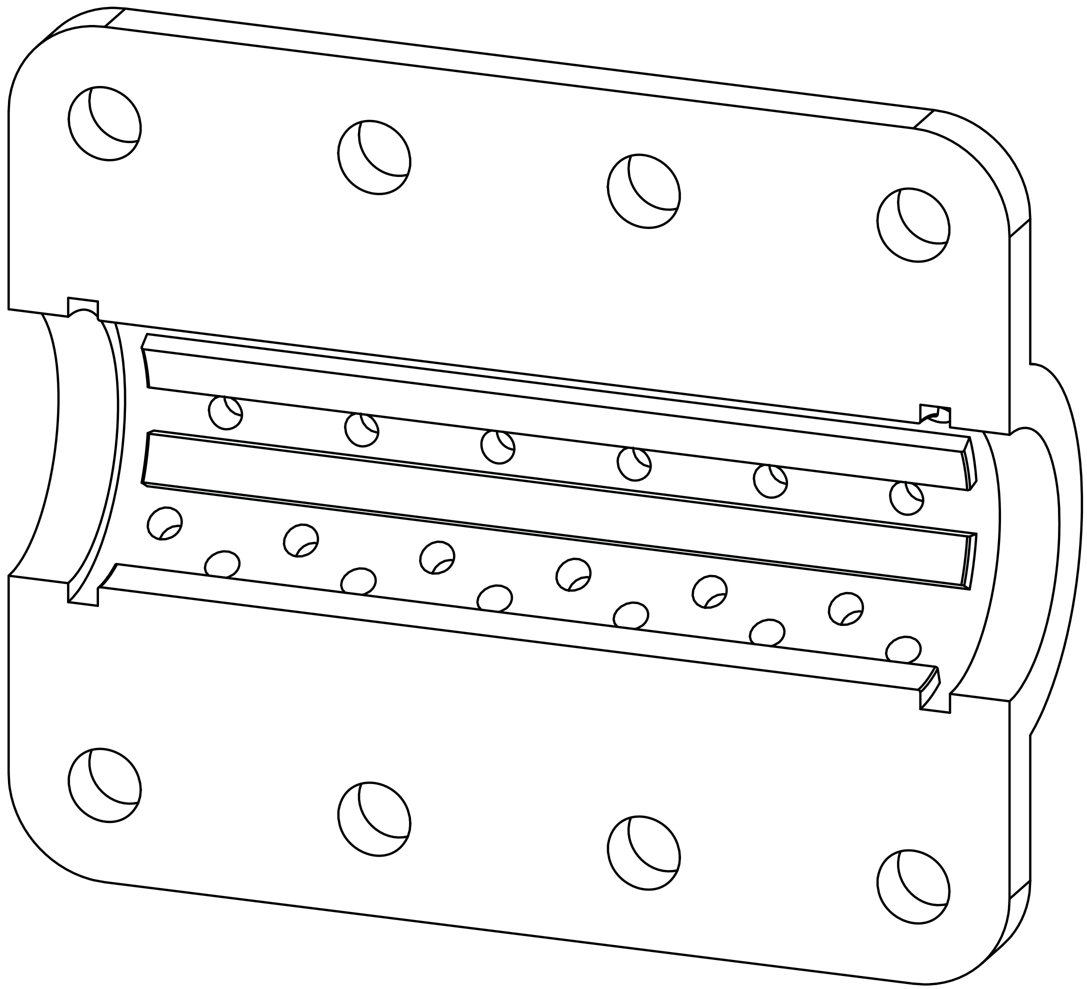


Figure 3.6-35: Disk-Type Jet from Circumferential Pipe Rupture at a Weld

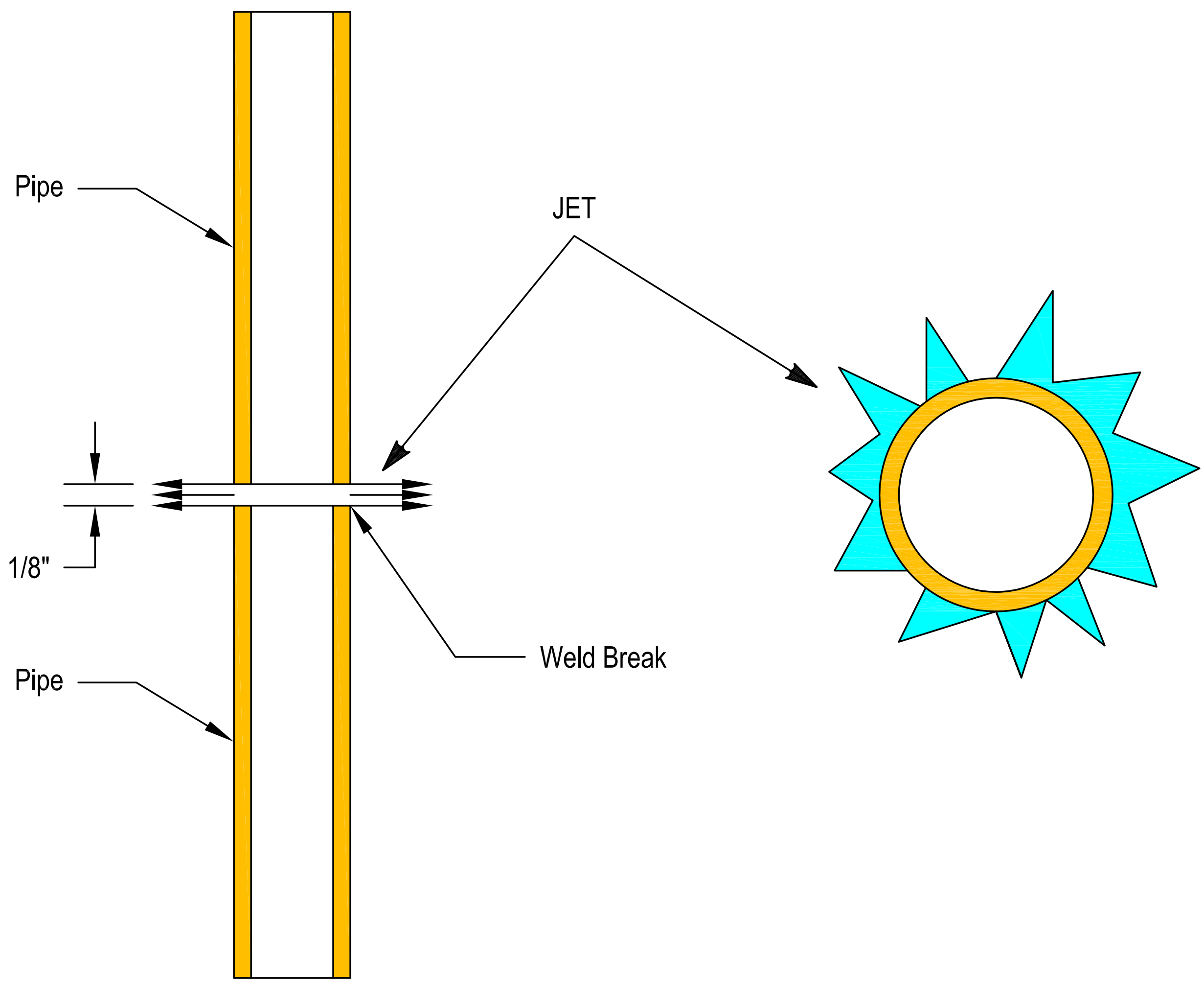


Figure 3.6-36: RCS pipe break total pressure drop along discharge centerline

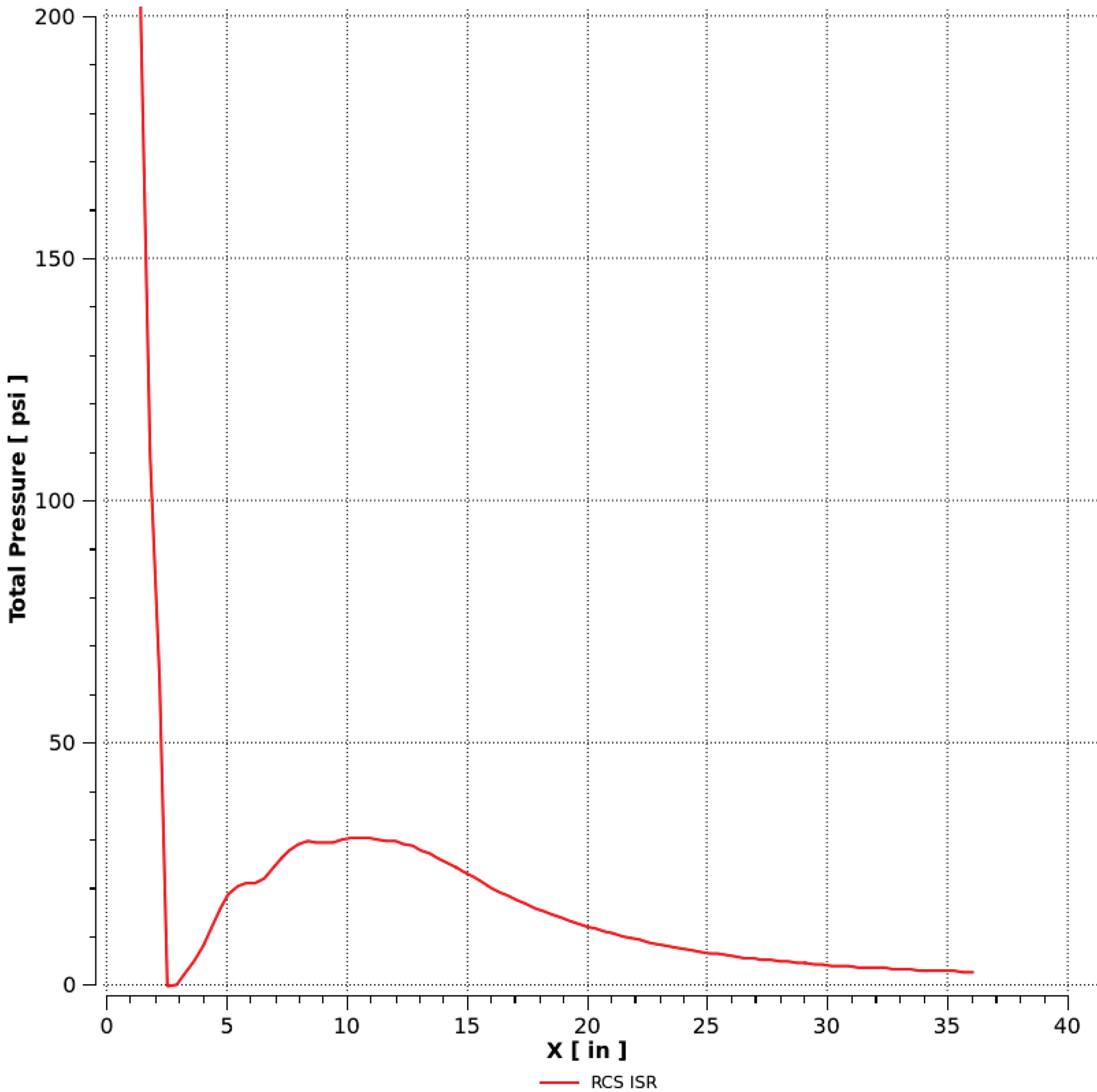


Figure 3.6-37: RCS pipe break total pressure graph at 5, 10, 15, and 20 inches radially from ISR

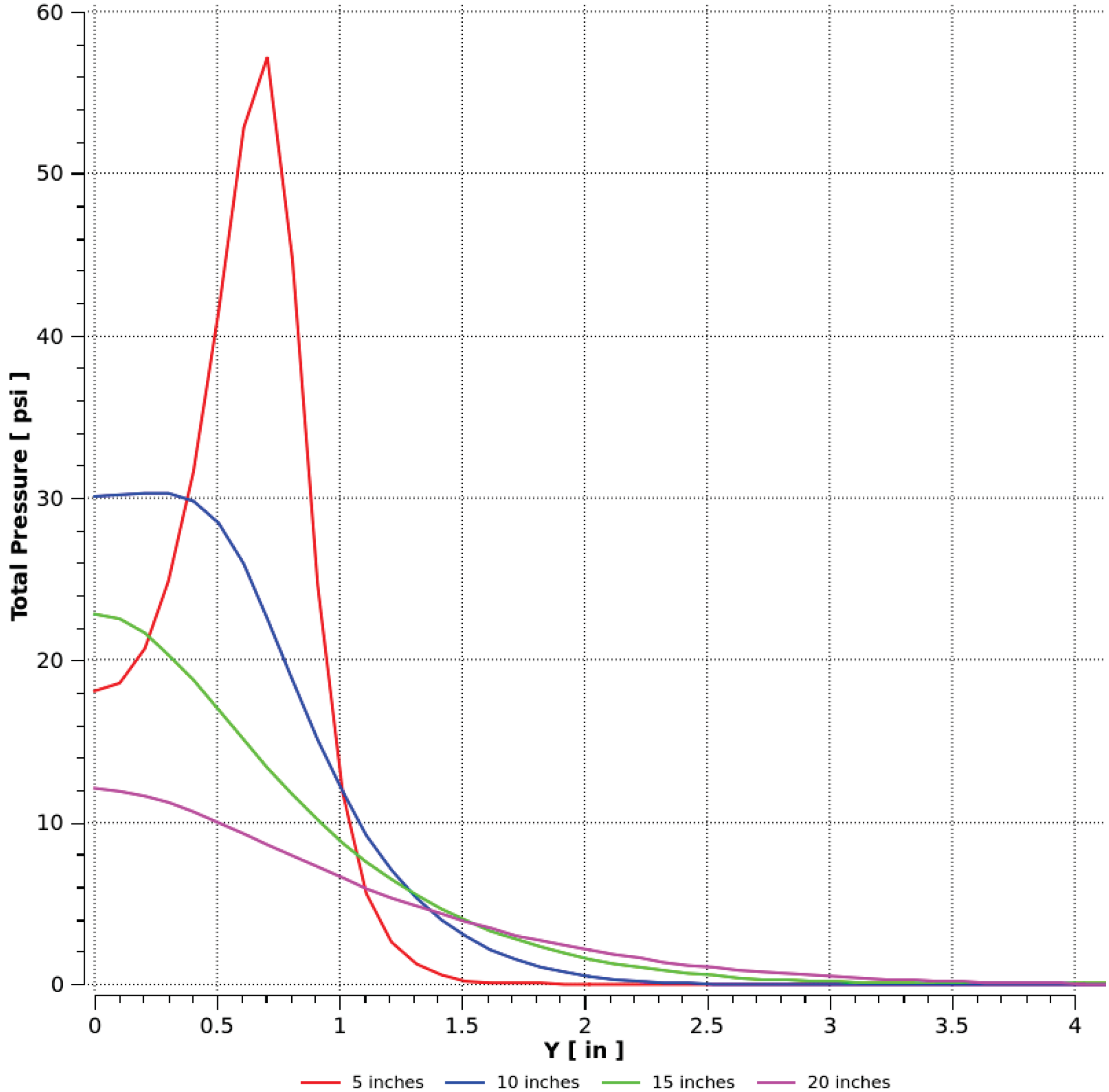


Figure 3.6-38: DHRS high temperature pipe break total pressure graph along discharge centerline

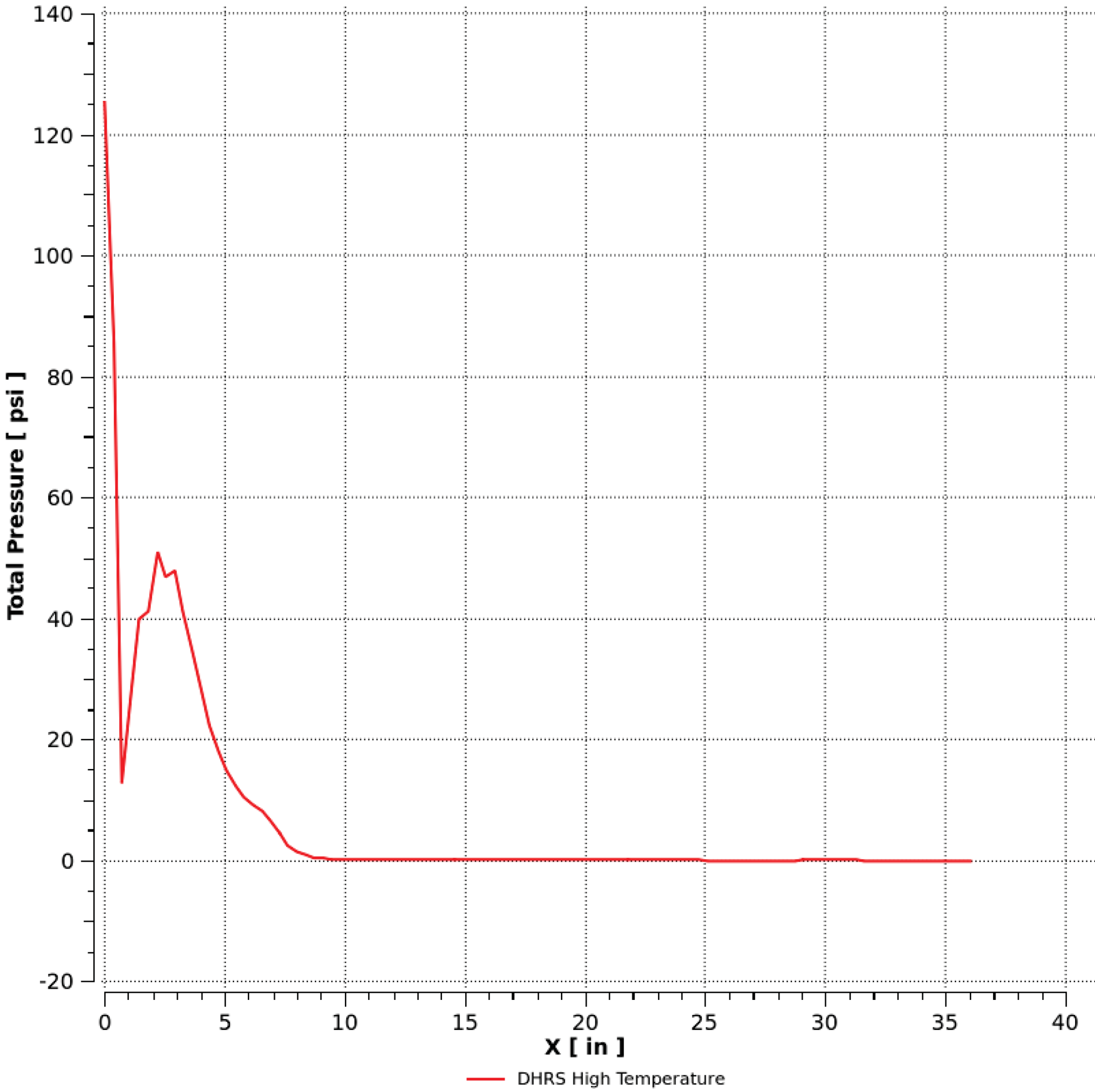


Figure 3.6-39: DHRS low temperature pipe break total pressure drop along discharge centerline

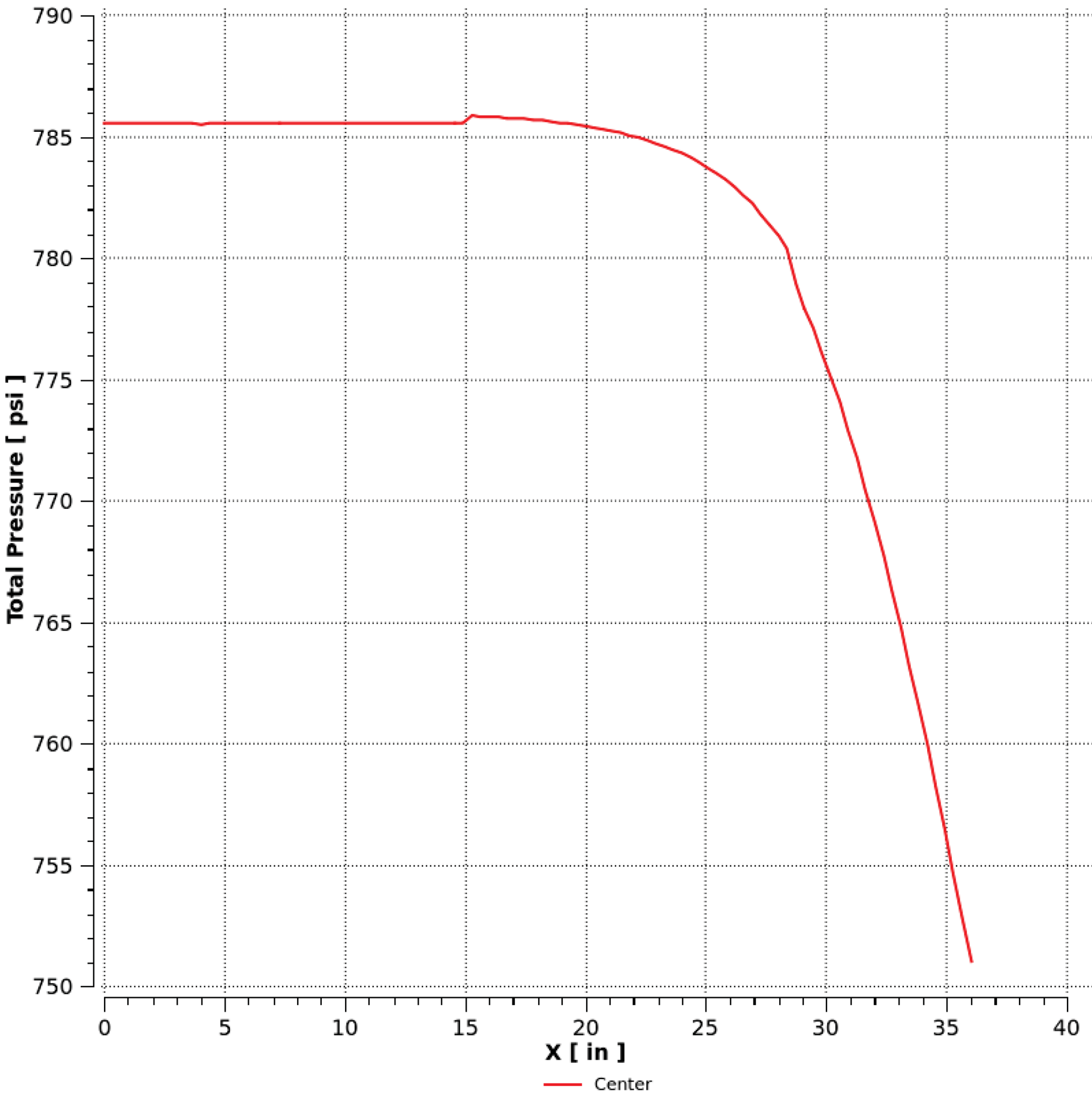


Figure 3.6-40: DHRS low temperature pipe break total pressure graph at 5, 10, 15, and 20 inches radially from ISR