December 31, 2022 Docket No. 52-050

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of the NuScale Standard Design Approval Application Part 2 – Final Safety Analysis Report, Chapter 3, "Design of Structures, Systems, Components and Equipment," Revision 0

REFERENCES:
1. NuScale letter to NRC, "NuScale Power, LLC Submittal of Planned Standard Design Approval Application Content," dated February 24, 2020 (ML20055E565)
2. NuScale letter to NRC, "NuScale Power, LLC Requests the NRC staff to conduct a pre-application readiness assessment of the draft, 'NuScale Standard Design Approval Application (SDAA),'" dated May 25, 2022 (ML22145A460)
3. NRC letter to NuScale, "Preapplication Readiness Assessment Report of the NuScale Power, LLC Standard Design Approval Draft Application," Office of Nuclear Reactor Regulation dated November 15, 2022 (ML22305A518)
4. NuScale letter to NRC, "NuScale Power, LLC Staged Submittal of Planned Standard Design Approval Application," dated November 21, 2022 (ML22325A349)

NuScale Power, LLC (NuScale) is pleased to submit Chapter 3 of the Standard Design Approval Application, "Design of Structures, Systems, Components and Equipment," Revision 0. This chapter supports Part 2, "Final Safety Analysis Report," (FSAR) of the NuScale Standard Design Approval Application (SDAA), as described in Reference 1. NuScale submits the chapter in accordance with requirements of 10 CFR 52 Subpart E, Standard Design Approvals. As described in Reference 4, the enclosure is part of a staged SDAA submittal. NuScale requests NRC review, approval, and granting of standard design approval for the US460 standard plant design.

From July 25, 2022 to October 26, 2022, the NRC performed a pre-application readiness assessment of available portions of the draft NuScale FSAR to determine the FSAR's readiness for submittal and for subsequent review by NRC staff (References 2 and 3). The NRC staff reviewed draft Chapter 3. NuScale is enclosing information in this submittal that: 1) closes gaps identified between the draft SDAA Chapter 3 and technical content generally expected by the NRC; and 2) resolves identified technical issues that may have adversely impacted acceptance, docketing, or technical review of the application. Enclosure 2 provides NuScale's responses to Reference 3 for Chapter 3 observations.

Enclosure 1 contains SDAA Part 2 Chapter 3, "Design of Structures, Systems, Components and Equipment," Revision 0, public version.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Mark Shaver at 541-360-0630 or at mshaver@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 31, 2022.

Sincerely,

Carrie Fosaaen
Senior Director, Regulatory Affairs
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Distribution: Brian Smith, NRC
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Enclosure 1: SDAA Part 2 Chapter 3, “Design of Structures, Systems, Components and Equipment,” Revision 0, (public)
Enclosure 2: Readiness Assessment Review Responses for Chapter 3

Enclosure 1:

SDAA Part 2 Chapter 3, “Design of Structures, Systems, Components and Equipment,” Revision 0 (public)

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NuScale US460 Plant
Standard Design Approval Application

Chapter Three

Design of Structures, Systems, Components and Equipment

Final Safety Analysis Report

Revision 0

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CHAPTER 3 DESIGN OF STRUCTURES, SYSTEMS, COMPONENTS AND EQUIPMENT

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

This section addresses design compliance with the General Design Criteria (GDCs) in 10 CFR 50, Appendix A.

In certain cases, the design implements a Principal Design Criterion (PDC) to address design-specific attributes of the NuScale US460 Standard Power Plant.

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). The NuScale QA Program provides confidence that the structures, systems, and components (SSC) that are required to perform safety-related and risk-significant functions are designed to perform the functions satisfactorily. NuScale's QA Program is described in the NuScale Quality Assurance Program Description.

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.

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 design conforms to GDC 1.

Relevant FSAR Chapters and Sections

Section 3.2 Classification of Structures, Systems, and Components

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)

Appendix 3C Methodology for Environmental Qualification of Electrical and Mechanical Equipment

Chapter 5 Reactor Coolant System and Connecting Systems

Chapter 6 Engineered Safety Features

Chapter 7 Instrumentation and Controls

Section 9.1 Fuel Storage and Handling

Section 9.3 Process Auxiliaries

Section 14.2 Initial Plant Test Program

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 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 site parameters are listed in Table 2.0-1.

Conformance or Exception

The design conforms to GDC 2.

Relevant FSAR Chapters and Sections

Chapter 2	Site Characteristics and Site Parameters
Section 3.2	Classification of Structures, Systems, and Components
Section 3.3	Wind and Tornado Loadings
Section 3.4	Water Level (Flood) Design
Section 3.5	Missile Protection
Section 3.7	Seismic Design
Section 3.8	Design of Category I Structures
Section 3.9	Mechanical Systems and Components
Section 3.10	Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment
Section 3.11	Environmental Qualification of Mechanical and Electrical Equipment
Section 3.12	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports
Appendix 3C	Methodology for Environmental Qualification of Electrical and Mechanical Equipment
Chapter 5	Reactor Coolant System and Connecting Systems
Chapter 6	Engineered Safety Features

Chapter 7	Instrumentation and Controls
Section 8.3	Onsite Power Systems
Section 9.1	Fuel Storage and Handling
Section 9.2	Water Systems
Section 9.3	Process Auxiliaries
Section 9.4	Air Conditioning, Heating, Cooling, and Ventilation Systems
Section 9.5	Other Auxiliary Systems
Section 10.3	Main Steam System
Section 10.4	Other Features of Steam and Power Conversion System
Chapter 15	Transient and Accident Analyses

3.1.1.3 Criterion 3-Fire Protection

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

Implementation in the NuScale Power Plant Design

Risk-significant or safety-related SSC are designed and located to minimize the probability and effect of fires and explosions. Noncombustible and fire resistance materials are used throughout the plant where fire is a potential risk to safety-related systems. Passive fire barriers compartmentalize the plant into separate areas or zones. Compartmentalization separates redundant, safety-related systems and components to ensure that a fire in one area does not prevent the redundant systems and components in an adjacent area from performing their safety functions. The primary purpose of these fire areas or zones is to confine the effects of fires to a single compartment or area, thereby minimizing the potential for adverse effects from fires on redundant risk-significant or safety-related SSC. Compartmentalization is achieved by using properly rated fire barriers, fire doors, fire dampers, and penetration seals to prevent the spread of fire between areas. Adequate equipment and cable separation meet the enhanced fire protection criteria as described in RG 1.189 Position 8.2 by the design of these divisions and subdivisions.

The fire protection system and equipment is designed in accordance with the guidance provided in Regulatory Guide 1.189, and applicable National Fire Protection Association codes. Consistency with RG 1.189 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 impair the safety capability of other SSC.

Conformance or Exception

This design conforms to GDC 3.

Relevant FSAR Chapters and Sections

Section 9.5 Other Auxiliary Systems

Appendix 9A Fire Hazard Analysis

Section 10.4 Other Features of Steam and Power Conversion System

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 effects of environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs), are considered in the design of safety-related and risk-significant SSC. The design protects against dynamic effects, including missiles, pipe whipping, and discharging fluids, that result from equipment failures and from events and conditions outside the NuScale Power Plant and prevents piping failure using break exclusion criteria.

Conformance or Exception

The design conforms to GDC 4.

Relevant FSAR Chapters and Sections

Section 3.3 Wind and Tornado Loadings

Section 3.4 Water Level (Flood) Design

Section 3.5 Missile Protection

Section 3.6 Protection against Dynamic Effects Associated with Postulated Rupture of Piping

Section 3.8 Design of Category I Structures

Section 3.9 Mechanical Systems and Components

Section 3.10 Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment

Section 3.11 Environmental Qualification of Mechanical and Electrical Equipment

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

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

Appendix 3C Methodology for Environmental Qualification of Electrical and Mechanical Equipment

Section 4.6 Functional Design of Control Rod Drive System

Chapter 5 Reactor Coolant System and Connecting Systems

Chapter 6 Engineered Safety Features

Chapter 7 Instrumentation and Controls

Section 8.3 Onsite Power Systems

Chapter 9 Auxiliary Systems

Chapter 10 Steam and Power Conversion System

Chapter 15 Transient and Accident Analyses

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

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

Implementation in the NuScale Power Plant Design

Safety-related and risk-significant SSC are not shared among the NuScale Power Modules (NPMs) with the exception of the control building, which houses the module protection system, and the reactor building which houses all NPMs, and the safety-related ultimate heat sink (UHS). The design of the UHS ensures that it can perform its safety functions, including in the event of an accident in one NPM, facilitating the orderly shutdown and cooldown of the remaining NPMs.

Conformance or Exception

The design conforms to GDC 5.

Relevant FSAR Chapters and Sections

Section 1.1 Introduction

Section 5.4 Reactor Coolant System Component and Subsystem Design

Section 6.2 Containment Systems

Section 6.3 Emergency Core Cooling System

Section 6.4 Control Room Habitability

Chapter 7 Instrumentation and Controls

Chapter 8 Electric Power

Chapter 9 Auxiliary Systems

Chapter 10 Steam and Power Conversion System

Chapter 15 Transient and Accident Analyses

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 plant is designed such that the reactor is 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 in conformance with GDC 10. The 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 design conforms to GDC 10.

Relevant FSAR Chapters and Sections

Section 3.9 Mechanical Systems and Components

Section 4.2 Fuel System Design

Section 4.3 Nuclear Design

Section 4.4 Thermal and Hydraulic Design

Chapter 7 Instrumentation and Controls

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 design conforms to GDC 11.

Relevant FSAR Chapters and Sections

Section 4.3 Nuclear Design

3.1.2.3 Criterion 12-Suppression of Reactor Power Oscillations

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

Implementation in the NuScale Power Plant Design

The reactor core is designed to ensure power oscillations that can result in conditions exceeding SAFDLs are not possible. Oscillations are evaluated at the beginning, middle, and end of the equilibrium cycle. The reactor core is to axially and radially stable, 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 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 design conforms to GDC 12.

Relevant FSAR Chapters and Sections

Section 4.3 Nuclear Design

Section 4.4 Thermal and Hydraulic Design

Section 15.9 Stability

3.1.2.4 Criterion 13-Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences,

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

Implementation in the NuScale Power Plant Design

Instrumentation and controls are provided to monitor variables and systems over their anticipated ranges for normal operations, AOOs, and postulated accident conditions to ensure adequate safety. The design of the safety-related instrumentation and control systems is based on independence, redundancy, predictability and repeatability, and diversity and defense-in-depth. 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 design conforms to GDC 13.

Relevant FSAR Chapters and Sections

Chapter 6 Engineered Safety Features

Chapter 7 Instrumentation and Controls

Chapter 9 Auxiliary Systems

Chapter 15 Transient and Accident Analyses

3.1.2.5 Criterion 14-Reactor Coolant Pressure Boundary

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

Implementation in the NuScale Power Plant Design

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

The primary and secondary water chemistry, along with the water chemistry for the pools forming the UHS, is monitored and controlled for chemical species that

can affect the RCPB integrity. Sampling and analysis of reactor coolant, secondary water, and pool water verify that key chemistry parameters are within prescribed limits and that impurities are properly controlled, which provides assurance that corrosion is mitigated and does not adversely affect the RCPB.

Conformance or Exception

The design conforms to GDC 14.

Relevant FSAR Chapters and Sections

Section 3.9 Mechanical Systems and Components

Section 3.10 Seismic and Dynamic Qualifications 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)

Chapter 5 Reactor Coolant System and Connecting Systems

Section 6.3 Emergency Core Cooling System

Section 9.3 Process Auxiliaries

Section 10.3 Main Steam System

Section 10.4 Other Features of Steam and Power Conversion 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 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.

The reactor safety valves provide RCPB overpressure protection and in accordance with the requirements of ASME Code, Section III Division 1, Subsection NB.

Conformance or Exception

The design conforms to GDC 15.

Relevant FSAR Chapters and Sections

Section 3.9 Mechanical Systems and Components

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

Chapter 5 Reactor Coolant System and Connecting Systems

Chapter 7 Instrumentation and Controls

Chapter 15 Transient and Accident Analyses

3.1.2.7 Criterion 16-Containment Design

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

Implementation in the NuScale Power Plant Design

The containment and associated systems are designed to establish an essentially leak-tight barrier against an uncontrolled release of radioactivity to the environment, and ensure 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 design conforms to GDC 16.

Relevant FSAR Chapters and Sections

Section 3.8 Design of Category I Structures

Section 6.2 Containment Systems

Chapter 7 Instrumentation and Controls

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 design supports an exemption from the criterion. The electric power systems are not necessary to perform safety functions. With electric power unavailable, safety-related SSC have sufficient capacity and capability to ensure (1) specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of AOOs and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Although not relied on to ensure plant safety-related functions are achieved, the designs of the onsite alternating current and direct current power systems include provisions for independence and redundancy.

Conformance or Exception

The design does not conform to GDC 17.

Relevant FSAR Chapters and Sections

Chapter 8 Electric Power

Chapter 15 Transient and Accident Analyses

3.1.2.9 Criterion 18-Inspection and Testing of Electric Power Systems

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

Implementation in the NuScale Power Plant Design

The design supports an exemption from the criterion. The electric power systems in the plant do not contain any safety-related or risk-significant SSC within the scope of GDC 18. Although not relied on to meet GDC 18, the plant design includes provisions for testing and inspection of power supply systems.

Conformance or Exception

The design does not conform to GDC 18.

Relevant FSAR Chapters and Sections

Chapter 8 Electric Power

3.1.2.10 Criterion 19-Control Room

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

Implementation in the NuScale Power Plant Design

The main control room (MCR) 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 loss-of-coolant accident (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 (HVAC) are normally provided to the MCR by the normal control room HVAC system (CRVS). Redundant smoke detectors and radiation detectors are provided in the outside air duct, upstream of both the CRVS filter unit 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 CRVS filter unit. When power is unavailable, or if high levels of radiation are detected in the control room envelope (CRE) supply duct, the CRVS air handling unit is stopped, the outside air intake is isolated, the bubble-tight CRE isolation dampers are closed, and the control room habitability system is actuated. The CRVS maintains the CRE habitable for at least 72 hours.

The MCR is designed with the ability to place the reactors in safe shutdown before an MCR evacuation event, and for safe shutdown to be maintained without further operator actions. Before evacuating the MCR, operators trip the reactors, initiate decay heat removal, and initiate containment isolation. These actions result in passive cooling that achieves safe shutdown of the reactors. Operators can also achieve safe shutdown of the reactors from outside the MCR in the I&C equipment rooms within the reactor building. The design supports an exemption from the portion of GDC 19 requiring the capability for subsequent cold shutdown from outside the control room. Following shutdown and initiation of passive cooling from either the MCR or the I&C equipment rooms, the design does not rely on operator action, instrumentation, or controls outside of the MCR to maintain a safe shutdown condition. The design includes remote stations for monitoring of the plant if the MCR is evacuated. There are no remote displays, alarms, or actions necessary to monitor and maintain the modules in a safe shutdown condition.

Conformance or Exception

The design conforms to PDC 19, as follows:

> A control room shall be provided from which actions can be taken to operate the plant safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.
>
> Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent as defined in 10 CFR 50.2 for the duration of the accident.
>
> Equipment at appropriate locations outside the control room shall be provided with a design capability for safe shutdown of the reactors, including necessary instrumentation and controls to maintain the modules in a safe shutdown condition.

Relevant FSAR Chapters and Sections

Section 5.4 Reactor Coolant System Component and Subsystem Design

Section 6.4 Control Room Habitability

Chapter 7 Instrumentation and Controls

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

Section 9.5 Other Auxiliary Systems

Appendix 9A Fire Hazard Analysis

Section 11.5 Process and Effluent Radiation Monitoring Instrumentation and Sampling

Section 12.3 Radiation Protection Design Features

Chapter 15 Transient and Accident Analyses

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 setpoints, trips the reactor to ensure analytical safety limits are not exceeded. 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 reactor coolant system (RCS).

Conformance or Exception

The design conforms to GDC 20.

Relevant FSAR Chapters and Sections

Chapter 7 Instrumentation and Controls

Chapter 15 Transient and Accident Analyses

3.1.3.2 Criterion 21-Protection System Reliability and Testability

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

Implementation in the NuScale Power Plant Design

The MPS incorporates the design principles of redundancy and independence to ensure high functional reliability. 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 does not prevent the protection function from being accomplished. The module protection system SSC are designed to be tested and calibrated while retaining the capability to accomplish their required safety function, permitting 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 design conforms to GDC 21.

Relevant FSAR Chapters and Sections

Chapter 7 Instrumentation and Controls

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 such that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions do not result in loss of the protection function. 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 does not prevent the safety function from being accomplished. 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 design conforms to GDC 22.

Relevant FSAR Chapters and Sections

Section 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 MPS has sufficient functional diversity to prevent the loss of a protection function, 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 power, or postulated adverse environments are experienced.

Conformance or Exception

The design conforms to GDC 23.

Relevant FSAR Chapters and Sections

Section 3.11 Environmental Qualification of Mechanical and Electrical Equipment

Appendix 3C Methodology for Environmental Qualification of Electrical and Mechanical Equipment

Section 4.6 Functional Design of Control Rod Drive System

Chapter 7 Instrumentation and Controls

Section 9.5 Other Auxiliary Systems

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

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.The MPS has physical, electrical, communication, and functional independence within the system and from nonsafety-related systems and components. The MPS has sufficient separation of the protection and the control systems to satisfy reliability, redundancy, and independence requirements even with a component or channel failed or removed from service.

Conformance or Exception

The design conforms to GDC 24.

Relevant FSAR Chapters and Sections

Chapter 7 Instrumentation and Controls

3.1.3.6 Criterion 25-Protection System Requirements for Reactivity Control Malfunctions

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

Implementation in the NuScale Power Plant Design

The setpoints of the MPS ensure that reactor trip or engineered safety feature actuation occurs before the process reaches the analytical limit. The setpoints are chosen to ensure the plant can operate and experience expected operational transients without unnecessary trips or engineered safety feature actuations. Chapter 15 safety analyses demonstrate SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

Conformance or Exception

The design conforms to GDC 25.

Relevant FSAR Chapters and Sections

Section 4.3 Nuclear Design

Section 4.6 Functional Design of Control Rod Drive System

Chapter 7 Instrumentation and Controls

Chapter 15 Transient and Accident Analyses

3.1.3.7 Criterion 26-Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One of the systems all 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 design incorporates two independent reactivity control systems of different design principles to satisfy GDC 26: CRDS and the chemical and volume control system (CVCS) in conjunction with the boron addition system.

The CRDS uses control rods and is 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 a stuck rod, SAFDLs are not exceeded. The CRDS is designed such that core reactivity can be safely controlled and that sufficient negative reactivity exists to bring the core to subcritical conditions following reactor trip, with appropriate margin for malfunctions such as a stuck rod, and SAFDLs are not exceeded. Motive power supply to the CRDS is removed on a reactor trip signal or loss of power, which cause the control rod assemblies (CRAs) by means of gravity alone.

The CVCS operates in conjunction with the boron addition system as the second reactivity control system. The CVCS is capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes by controlling the soluble boron concentration to compensate for fuel depletion during operation and reactivity changes from xenon burnout, to ensure SAFDLs are not exceeded.

The CVCS in conjunction with the boron addition system is capable of holding the reactor core subcritical under cold conditions.

Conformance or Exception

The design conforms to GDC 26.

Relevant FSAR Chapters and Sections

Section 3.9 Mechanical Systems and Components

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

The reactivity control systems are designed with a combined capability, in conjunction with supplemental boron recirculation during emergency core cooling system (ECCS) operation, of reliably controlling reactivity changes to ensure that under postulated accident conditions and with appropriate margin for a stuck rod the capability to cool the core is maintained.

The CRDS is designed such that core reactivity can be safely controlled and that sufficient negative reactivity exists to bring the core to subcritical conditions following reactor trip, with appropriate margin for malfunctions such as a stuck rod, and the capability to cool the core is maintained under postulated accident conditions. Motive power supply to the CRDS is removed on a reactor trip signal or loss of power, which causes the CRAs by means of gravity alone.

Following a reactor trip during a design-basis event, including a postulated accident, the ECCS is designed to actuate automatically (unless overridden by operators). The ECCS operation causes supplemental boron to recirculate into the reactor vessel, which, in conjunction with negative reactivity insertion from the control rods, maintains subcriticality for at least 72 hours with appropriate margin for a stuck rod, thereby ensuring the capability to cool the core is maintained. GDC 27 is applicable to the CRDS design as it relates to reactivity control systems being designed with appropriate margin for reliably controlling reactivity

under postulated accident conditions. The CRDS and the ECCS supplemental boron addition function, have the combined capability to reliably control reactivity changes and maintain the core cooling capability under postulated accident conditions with appropriate margin for a stuck rod. The CRDS rapidly inserts the CRAs via gravity upon a reactor trip signal or loss of power. The safety analyses demonstrate that, with a stuck rod, the capability to cool the core is maintained.

Conformance or Exception

The design conforms to GDC 27.

Relevant FSAR Chapters and Sections

Section 3.9 Mechanical Systems and Components

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

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 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 and disturbance of the core and reactor vessel internals due to a postulated reactivity event. Section 15.4 addresses plant safety associated with the reactivity insertion rates.

Conformance or Exception

The design conforms to GDC 28.

Relevant FSAR Chapters and Sections

Section 4.3 Nuclear Design

Section 4.6 Functional Design of Control Rod Drive System

Chapter 7 Instrumentation and Controls

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 ensure a high probability of performing the required safety-related functions in the event of an 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 by dropping the CRAs using gravity on a reactor trip signal or loss of power. The safety-related reactor trip function of the CRDS is initiated by MPS through the RTS. The MPS performs a reactor trip when plant parameters exceed the reactor trip setpoint. The reactor is placed in a subcritical condition with any assumed credible failure of any single active component.

Conformance or Exception

The design conforms to GDC 29.

Relevant FSAR Chapters and Sections

Section 3.9 Mechanical Systems and Components

Section 4.6 Functional Design of Control Rod Drive System

Chapter 7 Instrumentation and Controls

Section 9.3 Process Auxiliaries

Chapter 15 Transient and Accident Analyses

3.1.4 Fluid Systems

3.1.4.1 Criterion 30-Quality of Reactor Coolant Pressure Boundary

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

Implementation in the NuScale Power Plant Design

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

The containment evacuation system supports multiple 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 monitoring for level and radiation change. The capabilities of the detection methods are consistent with the guidance in Regulatory Guide 1.45.

Conformance or Exception

The design conforms to GDC 30.

Relevant FSAR Chapters and Sections

Section 3.2 Classification of Structures, Systems, and Components

Section 3.10 Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment

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 6.3 Emergency Core Cooling 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, stead state and transient stresses, and (4) size of flaws.

Implementation in the NuScale Power Plant Design

The design of the RPV and pressure retaining components associated with the RCPB have sufficient margin to ensure the RCPB behaves in a non-brittle manner and minimizes the probability of rapidly propagating fracture and gross rupture of the RCPB.

Conformance or Exception

The design conforms to GDC 31.

Relevant FSAR Chapters and Sections

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

Section 5.2 Integrity of Reactor Coolant Boundary

Section 5.3 Reactor Vessel

Section 6.1 Engineered Safety Feature Materials

Section 6.3 Emergency Core Cooling System

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 that are part of the RCPB are designed and provided with access to permit periodic inspection and testing requirements for ASME BPVC, Section III Division 1, Subsection NB Class 1 pressure-retaining components in accordance with ASME BPVC, Section XI Division 1 (Reference 3.1-5) pursuant to 10 CFR 50.55a(g). Equipment that may require inspection or repair is placed in an accessible position to minimize time and radiation exposure during refueling and maintenance outages.

An RPV material surveillance program is not necessary. The design supports an exemption from the requirements of 10 CFR 50.60, which includes the material surveillance requirements of 10 CFR 50, Appendix H. General Design Criterion 32 is satisfied because a material surveillance program is not necessary for the RPV materials.

Conformance or Exception

The design conforms to GDC 32.

Relevant FSAR Chapters and Sections

Section 5.2 Integrity of Reactor Coolant Boundary

Section 5.3 Reactor Vessel

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 design supports an exemption from GDC 33. The CVCS provides reactor coolant makeup during normal operation for small leaks in the RCPB, but is not relied upon to assure SAFDLs are not exceeded during a design-basis event. The RPV and CNV are designed to retain sufficient RCS inventory that, in conjunction with safety actuation setpoints to isolate CVCS from the RCS and operation of 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 design does not conform to GDC 33.

Relevant FSAR Chapters and Sections

Section 6.3 Emergency Core Cooling System

Section 8.2 Offsite Power System

Section 8.3 Onsite Power Systems

Section 9.3 Process Auxiliaries

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 decay and residual heat removal safety function is performed by the DHRS in conjunction with the containment isolation function of the main steam isolation valves, the main steam isolation bypass valves, and the feedwater isolation valves.

The DHRS is a closed-loop, passive condenser design that utilizes natural circulation flow from the steam generators to transfer 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 and main steam and feedwater containment isolation valves actuate upon loss or an interruption of electrical power.

The design supports an exemption from the electric power provisions of GDC 34.

Conformance or Exception

The design conforms to PDC 34, as follows:

> 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 ensure that the system safety function can be accomplished, assuming a single failure.

Relevant FSAR Chapters and Sections

Section 5.4 Reactor Coolant System Component and Subsystem Design

Section 8.2 Offsite Power System

Section 8.3 Onsite Power Systems

Section 9.5 Other Auxiliary Systems

Chapter 10 Steam and Power Conversion System

Chapter 15 Transient and Accident Analyses

3.1.4.6 Criterion 35-Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be 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 detections, 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 ECCS provides adequate passive heat removal following a loss of reactor coolant events.

The ECCS is enclosed inside containment and consists of two reactor vent valves located on the head of the RPV and two reactor recirculation valves located on the side of the RPV. All four valves are closed during normal operation and open when the system is actuated. 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 ensures 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. The ECCS does not rely on electrical power to perform its safety function. The valves are the only active components in the ECCS and actuate on stored energy. After the actuation, the valves do not require a subsequent change of state or electrical 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.

The design supports an exemption from the electric power provisions of GDC 35.

Conformance or Exception

The design conforms to PDC 35, as follows:

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 ensure that the system safety function can be accomplished, assuming a single failure.

Relevant FSAR Chapters and Sections

Section 4.2 Fuel System Design

Section 6.3 Emergency Core Cooling System

Section 8.2 Offsite Power System

Section 8.3 Onsite Power Systems

Chapter 15 Transient and Accident Analyses

3.1.4.7 Criterion 36-Inspection of Emergency Core Cooling System

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

Implementation in the NuScale Power Plant Design

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

Conformance or Exception

The design conforms to GDC 36.

Relevant FSAR Chapters and Sections

Section 6.3 Emergency Core Cooling System

3.1.4.8 Criterion 37-Testing of Emergency Core Cooling System

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

Implementation in the NuScale Power Plant Design

The ECCS is designed with the capability to support periodic pressure and functional testing 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 design conforms to GDC 37.

Relevant FSAR Chapters and Sections

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

Containment heat removal is an inherent characteristic ensured 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 CNV to the UHS. The design configuration of the CNV and UHS provides the ability to remove containment heat rapidly during accident conditions to reduce containment pressure and temperature, and maintain these conditions for an indefinite period with no reliance on active components or electrical power.

The design supports an exemption from the electric power provisions of GDC 38.

Conformance or Exception

The design conforms to PDC 38, as follows:

> 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 ensure that the system safety function can be accomplished, assuming a single failure.

Relevant FSAR Chapters and Sections

Section 6.2 Containment Systems

Section 8.2 Offsite Power System

Section 8.3 Onsite Power Systems

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 ASME BPVC, Section XI Division 1. The design permits appropriate periodic examination of the CNV to ensure continuing integrity and capability for heat transfer; that is, 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 design conforms to GDC 39.

Relevant FSAR Chapters and Sections

Section 6.2 Containment Systems

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 design supports an exemption from GDC 40. The NPM passive containment cooling does not include or require active components to provide the containment heat removal function, thus periodic testing specified by GDC 40 does not apply. Preoperational scaled testing of the passive containment heat removal function for LOCA conditions was performed and used to validate the computer code used to demonstrate 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 design does not conform to GDC 40.

Relevant FSAR Chapters and Sections

Section 6.2 Containment Systems

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

There are no containment atmosphere cleanup systems necessary 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.

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.

Passive autocatalytic recombiners are provided to control combustible gas concentrations in accordance with 10 CFR 50.44.

The design supports an exemption from the electric power provisions of GDC 41.

Conformance or Exception

The design conforms to PDC 41, as follows:

> Systems to control fission products, hydrogen, oxygen, and other substances that may release 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 ensure 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 ensure that its safety function can be accomplished, assuming a single failure.

Relevant FSAR Chapters and Sections

Section 6.2 Containment Systems

Section 6.5 Fission Product Removal and 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 subject to GDC 42.

Conformance or Exception

GDC 42 is not applicable to the design.

Relevant FSAR Chapters and Sections

Section 6.5 Fission Product Removal and Control Systems

3.1.4.14 Criterion 43-Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup 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 systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions are 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 systems, the transfer between normal and emergency power sources, and the operation of associated systems.

Implementation in the NuScale Power Plant Design

The design does not include containment atmosphere cleanup systems subject to GDC 43.

Conformance or Exception

GDC 43 is not applicable to the design.

Relevant FSAR Chapters and Sections

Section 6.5 Fission Product Removal and 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 UHS comprises the reactor pool, refueling pool, and spent fuel pool, and functions as a cooling water medium for the NPMs within the reactor pool and the stored spent fuel assemblies within the spent fuel pool. Because the NPMs are partially immersed in the UHS, no intermediate system is required to transfer heat from the NPMs to the UHS, which occurs either through the decay heat removal heat exchangers or through the containment vessel walls. Stored spent fuel assemblies are located in the UHS. To meet the intent of PDC 44, the requirements are applied to the UHS and the systems that ensure that the UHS is able to perform its safety function.

The Reactor Building provides a seismically-qualified enclosure that contains the water in the UHS. The pool leakage detection system provides indication of leakage from the pool walls and the pool liner on the floor of the UHS. Redundant level instrumentation provides another indication of leakage.

The pool cooling and cleanup system (PCWS) maintains UHS level and temperature during normal operation. The UHS maintains the core temperature at acceptably low levels following an accident, including a LOCA, that results 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 design supports an exemption from the electric power provisions of GDC 44.

Conformance or Exception

The design conforms to PDC 44, as follows:

> 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 ensure that the system safety function can be accomplished, assuming a single failure.

Relevant FSAR Chapters and Sections

Section 5.4 Reactor Coolant System Component and Subsystem Design

Section 8.2 Offsite Power System

Section 8.3 Onsite Power Systems

Section 9.2 Water Systems

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 pools that comprise the UHS are accessible for inspections.

Conformance or Exception

The design conforms to GDC 45.

Relevant FSAR Chapters and Sections

Section 9.2 Water Systems

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 periodic pressure or functional testing to ensure the structural and leaktight integrity of components or the operability and performance of the UHS. Verification of structural and leaktight integrity, and operability of the UHS, is accomplished by maintaining pool level, and by monitoring for leaks with the pool leakage detection system.

Conformance or Exception

The design conforms to GDC 46.

Relevant FSAR Chapters and Sections

Section 9.2 Water Systems

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 a design-basis LOCA with sufficient margin such that the design leak rates are not exceeded. 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 design conforms to GDC 50.

Relevant FSAR Chapters and Sections

Section 3.8 Design of Category I Structures

Section 6.2 Containment Systems

Section 8.3 Onsite Power 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 include sufficient margin to provide assurance that the containment pressure boundary does not undergo brittle fracture and the probability of rapidly propagating fracture is minimized under operating, maintenance, and postulated accident conditions. The ferritic containment pressure boundary materials satisfy the fracture toughness criteria for ASME BPVC Section III Division 1, Class 1 and 2 components.

Conformance or Exception

The design conforms to GDC 51.

Relevant FSAR Chapters and Sections

Section 6.2 Containment Systems

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 design supports an exemption from GDC 52. The CNV design allows testing and inspection, other than as anticipated by GDC 52, to ensure CNV leakage integrity.

The CNV is designed so that periodic local leak rate testing can be conducted in accordance with 10 CFR 50, Appendix J, thus ensuring that the allowable leakage rate values are not exceeded.

Conformance or Exception

The design does not conform to GDC 52.

Relevant FSAR Chapters and Sections

Section 6.2 Containment Systems

3.1.5.4 Criterion 53-Provisions for Containment Testing and Inspection

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

Implementation in the NuScale Power Plant Design

The CNV is designed to allow for sufficient access for inservice inspection of vessel welds and penetrations, and surveillance testing of containment isolation valves (CIVs) and penetration assemblies pursuant to ASME BPVC, Section XI Division 1 and "Standards and Guides for Operation and Maintenance of Nuclear Power Plants," ASME OM-2017 (Reference 3.1-6).

Conformance or Exception

The design conforms to GDC 53.

Relevant FSAR Chapters and Sections

Section 3.8 Design of Category I Structures

Section 6.2 Containment Systems

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 closure times are designed to minimize release of containment atmosphere to the environment.

Conformance or Exception

The design conforms to GDC 54.

Relevant FSAR Chapters and Sections

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 design supports an exemption from GDC 55. 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, the RCS discharge line, and the reactor high point degasification line, are part of the RCPB and penetrate primary reactor containment. Two CIVs are

provided for each of these lines and are located outside the CNV. Each line features a dual valve assembly and leakage test fixture welded directly to a CNV top head nozzle safe-end to provide two containment isolation barriers in series. The isolation valves are Seismic Category I components and constructed in accordance with ASME BPVC, Section III, Division 1, Subsection NB.

Conformance or Exception

The design does not conform to GDC 55. Alternate containment isolation provisions for specified lines penetrating the CNV satisfy the purpose of the GDC.

Relevant FSAR Chapters and Sections

Section 5.2 Integrity of Reactor Coolant Boundary

Section 6.2 Containment Systems

Chapter 15 Transient and Accident Analyses

3.1.5.7 Criterion 56-Primary Containment Isolation

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

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

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

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

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

Implementation in the NuScale Power Plant Design

The design supports an exemption from GDC 56. The lines that connect directly to the containment atmosphere and penetrate primary reactor containment 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 dual valve assembly and leakage test fixture welded directly to a containment top head nozzle safe-ends to provide two containment isolation barriers in series. The isolation valves are Seismic Category I components and constructed in accordance with ASME BPVC Section III Division 1, Subsection NB and NC.

Conformance or Exception

The design does not conform to GDC 56. Alternate containment isolation provisions for specified lines penetrating the CNV satisfy the purpose of the GDC.

Relevant FSAR Chapters and Sections

Section 6.2 Containment Systems

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 design supports an exemption from GDC 57 for some lines. 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 main steam and feedwater line is a Seismic Category I, ASME BPVC, Section III Division 1, Subsection NC, Class 2 valve. As noted in Section 3.1.5.7, for the RCCWS return and supply lines, two CIVs are provided for each line in a single-body, dual valve. These valves are Seismic Category I, ASME BPVC, Section III Division 1, Subsection NB, Class 1 components.

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

Conformance or Exception

The design of some lines does not conform to GDC 57. Other lines conform to GDC 57.

Relevant FSAR Chapters and Sections

Section 5.4 Reactor Coolant System Component and Subsystem Design

Section 6.2 Containment Systems

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 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 the limitations of 10 CFR 20 and 10 CFR 50, Appendix I are not exceeded.

Conformance or Exception

The design conforms to GDC 60.

Relevant FSAR Chapters and Sections

Section 3.2 Classification of Structures, Systems, and Components

Section 9.1 Fuel Storage and Handling

Section 9.2 Water Systems

Section 9.3 Process Auxiliaries

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

Chapter 10 Steam and Power Conversion System

Chapter 11 Radioactive Waste Management

Chapter 15 Transient and Accident Analyses

3.1.6.2 Criterion 61-Fuel Storage and Handling and Radioactivity Control

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

Implementation in the NuScale Power Plant Design

The PCWS cools the spent fuel assemblies stored 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 PCWS 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 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 PCWS. Section 9.2.5 describes performance of the UHS for accident conditions.

The area around the spent fuel pool is serviced by the nonsafety-related Reactor Building HVAC 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 cannot 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 PCWS, 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 design conforms to GDC 61.

Relevant FSAR Chapters and Sections

Section 9.1 Fuel Storage and Handling

Section 9.2 Water Systems

Section 9.3 Process Auxiliaries

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

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 fuel storage and handling 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 design conforms to GDC 62.

Relevant FSAR Chapters and Sections

Section 9.1 Fuel Storage and Handling

3.1.6.4 Criterion 63-Monitoring Fuel and Waste Storage

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

Implementation in the NuScale Power Plant Design

Monitoring for the loss of decay heat removal capability and excessive radiation levels is provided in the fuel storage and radioactive waste systems and associated handling areas for both normal and accident conditions. Information on cooling system performance is provided by the temperature detectors on the inlets and outlets of the heat exchangers in the PCWS. The outlet temperature detectors have a high setpoint for an alarm that alerts operators to determine the cause and ensure adequate active cooling performance. Leakage from 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 PCWS 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 design conforms to GDC 63.

Relevant FSAR Chapters and Sections

Section 9.1 Fuel Storage and Handling

Section 9.3 Process Auxiliaries

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 design 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 MCR. 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 design conforms to GDC 64.

Relevant FSAR Chapters and Sections

Chapter 7 Instrumentation and Controls

Section 9.1 Fuel Storage and Handling

Section 9.2 Water Systems

Section 9.3 Process Auxiliaries

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

Chapter 10 Steam and Power Conversion 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, 2017 edition, Section III Division 1, Subsection NB, "Class 1 Components," New York, NY.

3.1-3 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section II, "Materials," New York, NY.

3.1-4 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 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, 2017 edition, Section XI Division 1, "Rules for Inservice Inspection of Nuclear Components," New York, NY.

3.1-6 American Society of Mechanical Engineers, *Standards and Guides for Operation and Maintenance of Nuclear Power Plants*, ASME OM-2017, 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 in the determination of the appropriate quality standards and the identification of applicable codes and standards. The SSC classifications are based on a consideration of both safety-related functions (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 (D-RAP).

Table 3.2-1 identifies the buildings associated with the site layout and their seismic classification. Table 3.2-2 identifies a list of Seismic Category I SSC that provide pressure integrity functions or their supports, for the reactor coolant pressure boundary (RCPB). Table 3.2-2 also provides the applicable Quality Assurance Program (QAP) requirements and quality group classification. Discussion of systems comprised of Seismic Category II and III SSC are provided in the applicable chapters.

Seismic and quality group classification criteria are described in Section 3.2.1 and Section 3.2.2, respectively. The D-RAP process and categorization criteria are described in Section 17.4. Descriptions of the risk-significant Design Reliability Assurance Program structures, systems, and components classifications are found in Table 17.4-1.

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

Safety-related SSC and risk-significant SSC are subject to the QAP requirements described in Section 17.5. Applicable portions of 10 CFR 50 Appendix B have been applied to some nonsafety-related SSC where specific regulatory guidance applies (e.g., Regulatory Guide (RG) 1.29 “Seismic Design Classification for Nuclear Power Plants”).

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 radioactive waste systems based on the guidance in RG 1.143 “Design Guidance For Radioactive Waste Management Systems, Structures, And Components Installed In

Light-Water-Cooled Nuclear Power Plants"). No augmented requirements are identified for the SSC in Table 3.2-2.

The principal codes and standards used for the design of safety-related and risk-significant SSC are in accordance with the guidance of RG 1.26 "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

3.2.1 Seismic Classification

Seismic classification of SSC is consistent with the guidance of RG 1.29, and the SSC that meet Staff Regulatory Guidance C.2 are designated Seismic Category II. Seismic classification uses the following categories: Seismic Category I, Seismic Category II, Seismic Category III, and Seismic Category RW-IIa. These categories are described in Section 3.2.1.1, Section 3.2.1.2, Section 3.2.1.3, and Section 3.2.1.4, respectively.

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

In addition to RG 1.29, seismic classification of SSC is also consistent with the guidance in RG 1.143 and RG 1.189 "Fire Protection For Nuclear Power Plants."

Regulatory Guide 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 Section 11.2, Section 11.3 and Section 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 "Instrument Sensing Lines," as discussed in Section 7.2.2 and in Section C.1.f of RG 1.29. The use of this guidance ensures 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 safe shutdown earthquake (SSE).

The design of fire protection systems in accordance with RG 1.189 is described in Section 9.5.1.

3.2.1.1 Seismic Category I

The SSC classified as safety-related are designed to be capable of performing their safety functions during and following an SSE. Therefore, these safety-related SSC, including their foundations and supports, are classified as Seismic Category I.

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

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

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

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

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

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

3.2.1.2 Seismic Category II

The design requirements in Staff Regulatory Guidance C.2 in RG 1.29 for protection of Seismic Category I SSC are applied as follows to SSC classified as Seismic Category II. The 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 will not cause such failure. These SSC are classified as Seismic Category II.

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 because they are not required to remain functional following an event. Additionally, nonsafety-related instrument lines from safety related pressure boundaries are required to maintain pressure integrity.

Seismic Category II SSC are subject to the pertinent QAP requirements of 10 CFR 50, Appendix B.

3.2.1.3 Seismic Category III

The SSC not classified as Seismic Category I or Seismic Category II are classified as Seismic Category III. This category includes both 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

Regulatory Guide 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 classification. Therefore SSC used for radioactive waste must satisfy both criteria. Chapter 11 describes application of design criteria to SSC that are used for radioactive waste.

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.

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 American Society of Mechanical Engineers (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 discussed in Section 7.2. Design and construction requirements for reactor vessel internals are discussed in Section 3.9. Design and construction requirements for steam generator supports and tube supports are discussed in Section 5.4.

The following subsections also describe the codes and standards applicable to supports for Quality Group A, B, C, and D components.

3.2.2.1 Quality Group A

Regulatory Guide 1.26 establishes that Quality Group A corresponds to the category of components presented in 10 CFR 50.55a(c)(1). This category is limited to components that are part of the RCPB, except for the portions excluded by 10 CFR 50.55a(c)(2) that are included in Quality Group B.

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

All portions of the RCPB are in Quality Group A.

3.2.2.2 Quality Group B

Regulatory Guide 1.26 establishes that 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 RCPB 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 (CIVs), 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 RCPB that cannot be isolated from that boundary during all modes of operation by two normally closed or automatically closable valves.

Quality Group B SSC meet the requirements for Class 2 components in Section III, Division 1 of the ASME BPVC and applicable conditions promulgated in 10 CFR 50.55a(b). Supports for Quality Group B SSC meet the requirements for Class 2 supports in Section III, Division 1, Subsection NC and NF of the ASME BPVC.

3.2.2.3 Quality Group C

Regulatory Guide 1.26 establishes that 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 RCPB 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 RCPB 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 0.1 rem to the whole body or its equivalent to any part of the body

Quality Group C SSC meet the requirements for Class 3 components in Section III, Division 1 of the ASME BPVC and applicable conditions promulgated in 10 CFR 50.55a(b). Supports for Quality Group C SSC meet the requirements for Class 3 supports in Section III, Division 1, Subsection ND and NF of the ASME BPVC.

3.2.2.4 Quality Group D

Regulatory Guide 1.26 establishes that Quality Group D applies to water and steam-containing components that are not part of the RCPB 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).

The SSC designated as Quality Group D meet the codes and standards for components identified as applicable for Quality Group D in Table 1 of RG 1.26.

3.2.3 References

3.2-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section III, “Rules for Construction of Nuclear Facility Components,” New York, NY.

Table 3.2-1: Seismic Classification of Building Structures

Structure	Seismic Category
Reactor Building (RXB) • Majority of the building is SC-I, but several areas of the building are classified as SC-II (Note 1)	I/II
Turbine Building	III
Control Building (CRB) • Area housing the main control room and associated facilities	I
CRB • Portions of the building where protection of SC-I areas is required	II
CRB • Areas not classified as SC-I or SC-II (Note 1)	III
Annex Building	III
Firewater Building	III
Diesel Generator Building	III
Radioactive Waste Building • Above-grade structure	III
Radioactive Waste Building • Below-grade structure (areas designated for storage or processing of radioactive waste)	RW-IIa
Central Utility Building	III
Site Plant Cooling Water Chem Feed Building	III
Site Utility Racks	III

Note 1: Seismic Classification for RXB and CRB are discussed in Section 3.7.2.

Table 3.2-2: Classification of Seismic Category I, Pressure-Retaining Mechanical Systems and Components

Systems and Components (Note 1)	ASME Code Class	QA Program Applicability (Note 2)	Quality Group (Note 3)
Containment System (CNTS) Location: RXB			
• CIVs for: -Chemical volume and control system (CVCS) pressurizer spray, inboard & outboard -Reactor pressure vessel high point degasification, inboard & outboard -CVCS injection & discharge, inboard & outboard • Containment isolation test fixture valve for: -CVCS injection & discharge, pressurizer spray, and reactor pressure vessel high point degasification	1	Q	A
Steam Generator System Location: RXB			
• Steam generator tubes • Feed plenum access ports • Integral steam plenum and caps	1	Q	A
Control Rod Drive System Location: RXB			
• Control rod drive mechanism pressure boundary	1	Q	A
In-Core Instrumentation System Location: RXB			
• In-core instrumentation stringer assembly (outer sheath only)	N/A	Q	N/A
Reactor Coolant System Location: RXB			
• All components	1	Q	A
Emergency Core Cooling System Location: RXB			
• Reactor vent valve • Reactor vent valve trip valve • Reactor recirculation valve • Reactor recirculation trip valve • Reset valve • Reset lines • Trip lines	1	Q	A

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

Note 2: QAP applicability codes are as follows:

- Q = indicates quality assurance requirements of 10 CFR 50 Appendix B are applicable in accordance with the QAP (Section 17.5).

Note 3: Section 3.2.2.1 through Section 3.2.2.4 contain the applicable codes and standards for each RG 1.26 Quality Group designation A, B, C, and D. A Quality Group classification per RG 1.26 is not applicable to supports.

3.3 Wind and Tornado Loadings

The Seismic Category I portions of the Reactor Building (RXB) and Control Building (CRB), as well as the RW-IIa portion of the Radioactive Waste Building (RWB), are evaluated for wind, tornado, and hurricane loads. The three buildings are enclosed structures. The design criteria for evaluating local damage to the Reactor Building and Control Building from tornado and wind is consistent with Regulatory Guide (RG) 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants" and RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants."

The RWB is Seismic Category III above grade, with some limited sections categorized as RW-IIa (High Hazard). The RW-IIa portion is analyzed for tornado missile protection per RG 1.143, "Design Guidance For Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

The RW-IIa portion of the RWB is designed using the same wind, tornado, and hurricane loads as specified for the Seismic Category I structures. This approach is consistent with the wind load specified in Table 2 of RG 1.143. The ASCE/SEI 7-05 Standard (Reference 3.3-1) is used for wind loads in this design. Tornado missiles from RG 1.76 are also used.

COL Item 3.3-1: An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to severe and extreme (tornado and hurricane) wind loads will not collapse and adversely affect the Seismic Category I portions of the Reactor Building or of the Control Building.

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 capability to perform their safety functions. This conformance is achieved by establishing design parameters that are representative of a reasonable number of potential plant site locations in the United States. Design parameters for severe wind loads are provided in Section 3.3.1.1 and design parameters for extreme wind loads are provided in Section 3.3.1.2.

The RW-IIa portions of the RWB have been evaluated for severe and extreme wind loads using the methodology in Section 3.3.1.3 and can withstand severe and extreme winds.

3.3.1 Wind Loadings

3.3.1.1 Design Parameters for Severe Wind

The design-basis operating wind speed is a 3-second gust at 33 feet above ground for exposure category C. The operating wind speed (V_W) is 190 mph. The wind speed is increased by an importance factor of 1.15 for the design of the Reactor Building, Control Building, and Radioactive Waste Building. These design parameters are based upon Reference 3.3-1.

3.3.1.2 Design Parameters for Extreme Wind (Tornado and Hurricane)

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 consistent with tornado parameters postulated in RG 1.76:

- Maximum tornado wind speed . . . . . 270 mph
- Translational speed . . . . . 55 mph
- Maximum rotational speed . . . . . 215 mph
- Radius of maximum rotational speed . . . 150 ft
- Pressure drop . . . . . 1.6 psi
- Rate of pressure drop . . . . . 0.9 psi/sec

For tornadoes, V_w is the resultant of the maximum rotational speed and the translational speed.

The wind speed for the design-basis hurricane is the highest wind speed postulated in Regulatory Position 1 of RG 1.221, which occurs in Figure 2 of the RG:

- Maximum hurricane wind speed . . . . . 290 mph

Section 3.5 describes hurricane and tornado wind-generated missiles.

3.3.1.3 Determination of Wind Forces

The maximum wind velocities for severe winds and for tornado and hurricane winds are converted into pressure-induced forces (q_z) based on the applicable maximum wind speed (V_w) and applied to structures.

Wind forces are determined in conformance with 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 Reference 3.3-1, Table 6-3, but not less than 0.87,

Height “z” is the building height.

For tornadoes, wind speed is not assumed to vary with height.

K_{zt} = topographic factor equal to 1.0,

K_d = wind directionality factor equal to 1.0,

V_w = maximum wind speed, and

I = importance factor equal to 1.15 for the Reactor Building, Control Building, and Radioactive Waste Building.

Design wind pressures and extreme wind pressures on the Reactor Building, Control Building, and RW-IIa portion of the Radioactive Waste Building are determined in conformance with Reference 3.3-1, Equation 6-17:

$p = qGC_p – q_i (GC_{pi})$ (lb/ft^2)

where,

G = gust factor equal to 0.85,

C_p = external pressure coefficient equal to 1.0,

GC_{pi} = internal pressure coefficient equal to 0.18 for severe and hurricane winds,

q = velocity pressure, and

q_i = internal velocity pressure.

3.3.1.4 Combination of Forces

After tornado wind effects, W_w, atmospheric pressure change effects, W_p, and missile impact effects, W_m, are determined, the combined effect is determined. The method for combining these effects and establishing the total tornado load on a structure is:

$W_t = W_p$

$W_t = W_w + 0.5\ W_p + W_m$

Where:

W_t = total tornado load

W_w = load from wind effect

W_p = load from atmospheric pressure change effect, which is the pressure drop from Section 3.3.1.2.

W_m = load from missile impact effect

The most adverse of the above combinations is used.

After hurricane wind effects, W_W, and missile impact effects, W_m, are determined, the combined effect is determined. The method of combining these effects and establishing the total hurricane load on a structure is:

$W_h = W_W + W_m$

Where:

W_h = total hurricane load

W_W = load from wind effect

W_m = load from missile impact effect.

3.3.2 References

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

3.4 Water Level (Flood) Design

Flooding of a nuclear power plant can come from internal and external sources. 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 plant's structures. In the absence of final pipe routing information, this section addresses the expected flooding hazards.

The design satisfies General Design Criterion 4 because structures, systems, and components (SSC) are designed to withstand 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 satisfies General Design Criterion 2 because SSC accommodate 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 is performed for 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 loss of equipment required to perform safety functions. These SSC are "equipment subject to flood protection."

Table 3.4-1 identifies zones that contain SSC subject to flood protection. Safety-related cable is either routed above the flood level or qualified for submergence. Table 3.4-1 does not include zones where cables are the only safety-related SSC. Mitigation of flooding in the identified zones is accomplished by 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 analyses of the RXB and CRB identify potential internal flooding sources inside each building due to postulated pipe ruptures and fire suppression activities. For each flood source, the limiting flood height is determined for each area that contains that source. Flood heights are calculated for both a circumferential break (for non-seismic piping) and a leakage crack (for seismically-qualified piping).

The flooding analyses for the RXB and CRB include:

- identification of potential sources of internal flooding
- division of flood-able areas of each building into flood zones

- calculating maximum steady-state flood heights based on identified sources for use in informing flood mitigation requirements
- determining flood mitigation design features

3.4.1.1 Considerations and Assumptions used in the Flooding Analyses

High and moderate energy piping greater than 1-inch nominal diameter are considered as potential flood sources in any zone or area where they pass. The postulated piping failure is either a full-circumferential break or through-wall leakage crack, depending on the pipe energy and seismic classification. The critical crack size is half the pipe diameter in length and half the wall thickness in width. For non-seismic tanks, the full tank volume is considered to be instantaneously released when ruptured.

Considerations:

- The pipe size, schedule, location, and maximum operating conditions for each potential source are documented.
- For each potential flood source, the limiting flood height is calculated for each area that contains that source.
- Flood heights are calculated for both a circumferential break (for Seismic Category III piping) and a leakage crack (for Seismic Category I and II piping).
- For determining maximum flood levels, a single flooding event at a time is analyzed and a single active failure in systems used to mitigate the consequence of a piping failure is considered.
- Steady-state flood levels are calculated using the total source release and the available floor area. The released flood water spreads out through door gaps or wall openings to any connected rooms in the same flood zone.
- Backflow through floor drains is determined by taking a single blockage at a time.
- To account for various equipment and other blockages in the room, a standard gross room area reduction of 20 percent is applied to all rooms.

Assumptions:

- The collapse of concrete interior floors is not a viable water flow path or cause for flood.
- Floor drains or water removal systems are not available for calculating maximum flood levels.
- Fire pumps automatically start on low pressure in the fire main with electric and diesel fire pumps in operation.
- Pipe rupture releases are calculated using the maximum system operating pressure and temperature and pressure at the source release point is maintained throughout the release.

For the CRB:

- The flood source is detected and isolated within 30 minutes after leak initiation.
- The fire suppression system is isolated after 60 minutes.
- The elevator is not considered to be a floodable area.
- There is an additional 250 gpm of flow from a manually operated fire hose for one hour.
- The stairs are treated as a floodable area for the grade-level elevation.

For the RXB:

- The flood source is detected and isolated within 40 minutes after leak initiation.
- The fire suppression system is isolated after 120 minutes.
- The elevator or stairs are not considered to be floodable areas.
- Tanks with a capacity of less than 100 gallons are not considered.
- There is an additional 250 gpm of flow from a manually operated fire hose for 120 minutes.
- Stairwells and other floor openings are not credited for flood relief. However, the bounding source for a flooding area could potentially be a result of a propagation from a higher level in the building. To account for this potentiality, limiting sources from higher elevations are considered for lower elevations if there is a propagation path between the flooding areas.

3.4.1.2 Reactor Building Flooding Analysis

Table 3.4-1 lists potential flooding sources in the RXB.

Containment flooding analysis is not necessary for the design because containment is flooded as part of normal shutdown in preparation for NuScale Power Module refueling as described in Section 9.3.7.

3.4.1.3 Control Building Flooding Analysis

Table 3.4-1 lists potential flooding sources in the CRB. There are no high-energy piping systems in the CRB. Potential flood sources in the CRB are from moderate-energy piping.

3.4.1.4 Flooding Outside the Reactor Building and the Control Building

Neither the RXB and CRB flood from external sources. The design external flood level is established as less than one foot below the plant grade elevation.

Water from tanks and piping that are seismic category III and not protected against tornadoes or hurricanes are potential flooding sources outside the

buildings. There are no large tanks or water sources near entrances to the RXB and CRB. 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: An applicant that references the NuScale Power Plant US460 standard design will confirm the final location of structures, systems, and components subject to flood protection. The final routing of piping, and site-specific tanks or water source tanks are placed in locations that would not cause flooding to the Reactor Building or Control Building.

COL Item 3.4-2: An applicant that references the NuScale Power Plant US460 standard design will develop the on-site program addressing the key points of flood mitigation. The key points to this program include the procedures for mitigating internal flooding events; the equipment list of structures, systems, and components subject to flood protection in each plant area; and providing assurance that the program reliably mitigates flooding to the identified structures, systems, and components.

COL Item 3.4-3: An applicant that references the NuScale Power Plant US460 standard design 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.

3.4.2 Flood Protection from External Sources

Portions of the RXB and CRB are Seismic Category I. The RW-IIa portions of the Radioactive Waste Building do not contain 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 a "Dry Site" as defined in Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants." Seismic Category I structures are protected from external floods and groundwater. Table 2.0-1 provides site parameter probable maximum flood and maximum groundwater elevations.

The yard is graded to slope away from these buildings.

The design does not use a permanent dewatering system.

COL Item 3.4-4: An applicant that references the NuScale Power Plant US460 standard design will determine the extent of waterproofing and damp proofing needed for the underground portion of the Reactor Building based on site-specific conditions. Additionally, the applicant will provide the specified design life for waterstops, waterproofing, damp proofing, and watertight seals. If the design life is less than the operating life of the plant, the applicant will describe how continued protection will be ensured.

The design establishes a design-basis flood level (including wave action) of one foot below the plant grade elevation. 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, ground water, and soil pressure are factored into the structural design as static and dynamic loads discussed in Section 3.8.4.

3.4.2.2 Probable Maximum Precipitation

The design uses bounding parameters for rain and snow. These site parameters are presented in Table 2.0-1.

The RXB and CRB roofs are designed to prevent buildup of standing water.

The RXB and CRB roofs, including the bounding rain and snow loads, are evaluated as part of 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

Section 3.7.2 describes how 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 portions of the RXB and Seismic Category I portions of the CRB.

Analysis shows the CRB is capable of withstanding effects of the probable maximum precipitation.

COL Item 3.4-5: An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to external flooding will not collapse and adversely affect the Reactor Building or Seismic Category I portion of the Control Building.

Table 3.4-1: Limiting Flooding Sources and Maximum Steady-State Flood Heights

Building	Zone	Limiting Sources	Break Type	Max Flood Height (ft)
RXB	{{ Withheld - See Part 9 }}	FP 8 (Stairwell Flow)	Circumferential Break	4.1
	{{ Withheld - See Part 9 }}	FP 8 (Drain Backflow)	Circumferential Break	15.0
	{{ Withheld - See Part 9 }}	FP 8	Circumferential Break	15.0
	{{ Withheld - See Part 9 }}	FP 8	Circumferential Break	12.8
	{{ Withheld - See Part 9 }}	FP 8 (Stairwell Flow)	Circumferential Break	8.4
	{{ Withheld - See Part 9 }}	FP 8	Circumferential Break	3.8
	{{ Withheld - See Part 9 }}	FP 8 (Stairwell Flow)	Circumferential Break	12.8
	{{ Withheld - See Part 9 }}	FP 8	Circumferential Break	6.6
	{{ Withheld - See Part 9 }}	Flood Barrier to C306	Circumferential Break	-
	{{ Withheld - See Part 9 }}	Flood Barrier to C319	Circumferential Break	-
	{{ Withheld - See Part 9 }}	FP 8	Circumferential Break	4.3
	{{ Withheld - See Part 9 }}	FP 4	Circumferential Break	3.1
	{{ Withheld - See Part 9 }}	FP 4	Circumferential Break	2.7
	{{ Withheld - See Part 9 }}	UW 1.25	Circumferential Break	0.7
	{{ Withheld - See Part 9 }}	FP 4	Circumferential Break	5.3
	{{ Withheld - See Part 9 }}	FP 4	Circumferential Break	5.4
CRB	{{ Withheld - See Part 9 }}	Utility Water	Circumferential Break	0.6
	{{ Withheld - See Part 9 }}	Fire Suppression	-	1.0
	{{ Withheld - See Part 9 }}	Fire Suppression	-	0.6
	{{ Withheld - See Part 9 }}	Fire Suppression	-	1.9
	{{ Withheld - See Part 9 }}	There are no flood sources in the MCR	-	-
	{{ Withheld - See Part 9 }}	Potable Water	Leakage Crack	0.2

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
RXB (continued)		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

The design achieves protection from external missiles by locating structures, systems, and components (SSC) that require missile protection inside Seismic Category I portions of the Reactor Building (RXB) or Seismic Category I portions of the Control Building (CRB).

The design complies with General Design Criteria (GDC) 2 and GDC 4 because SSC are designed to accommodate the effects of internally and externally generated missiles without losing the ability to perform their safety functions.

The Radioactive Waste Building (RWB) is classified as Seismic Category III for most of the above grade portion and as RW-IIa below ground. The RW-IIa portions of the RWB meet design criteria for missiles specified in Table 2 of Regulatory Guide (RG) 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

The buildings provide missile protection by

- having design features to prevent missile generation.
- 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.

Table 3.2-1 identifies the buildings associated with the site layout and their seismic classification. Table 3.2-2 lists Seismic Category I SSC that provide pressure integrity functions or their supports, for the reactor coolant pressure boundary. Table 17.4-1 provides a list of risk-significant SSC that have a safety function that might be relied upon following a missile-producing event.

3.5.1 Missile Selection and Description

Missile generation is assumed to occur during operating conditions.

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 probability of missile occurrence is P_1. This section discusses when missiles are not 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} and not credible.

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

Missiles from piping or valves designed in accordance with American Society of Mechanical Engineers (ASME) Section III, (Reference 3.5-1) and maintained in accordance with an ASME Section XI (Reference 3.5-2) inspection program are not credible because construction and maintenance to these standards provides reasonable assurance that structural integrity is maintained during normal and upset conditions.

Valve stems are not credible missiles if at least one feature (in addition to the stem threads) is included in their design to prevent ejection. Valve stems with backseats 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 a small amount of stored energy and are not credible missiles.

Thermowells and similar fittings attached to piping or pressurized equipment by welding are not 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 credible missiles. The quantity of high-energy fluid in these instruments is limited and does not result in missile generation. The connecting piping and tubing is made using welded joints or compression fittings for the tubing. Tubing is small diameter and has 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 can be stored in the buildings. Cylinders, bottles, or tanks containing pressurized gas are considered missile sources unless secured.

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

Procedures developed in accordance with Section 13.5.2 ensure portable pressurized gas cylinders or bottles are moved to a location where they are not a potential hazard to equipment subject to missile protection requirements, or seismically restrained to prevent them from becoming missiles.

3.5.1.1.3 Rotating Equipment

There are no reactor coolant pumps, turbine driven pumps, or other large rotating components inside safety-related structures. The main turbine generators are outside the RXB.

Catastrophic failure of rotating equipment, such as fans and compressors, leading to the generation of missiles is not 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

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 that reduce hydrogen production. Therefore, a hydrogen explosion in a battery compartment is not a credible missile source. The RWB waste management control room, battery, and battery charging rooms are each served by two dedicated 100-percent capacity recirculating fan cooling units.

3.5.1.1.5 Gravitational Missiles

The SSC that could fall and impact or adversely affect safety-related or risk-significant SSC are classified as Seismic Category II. Seismic Category II equipment is mounted to ensure there is no adverse interaction between Seismic Category I SSC and Seismic Category II SSC. These SSC are not credible missiles.

Section 9.1.5 evaluates the overhead heavy load handling system. The devices in this system are not credible missile sources.

Procedures developed in accordance with Section 13.5.2 ensure 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 ensure equipment brought into the building to perform maintenance and equipment undergoing maintenance in the Reactor Building or Control Building is restrained to prevent it from becoming a missile. 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.

There is no rotating equipment inside containment, and pressurized components are ASME Class 1 or 2 and therefore not credible missile sources as discussed in Section 3.5.1.1.

A control rod drive mechanism housing failure, sufficient to create a missile from a piece of the housing or that allows a control rod to be ejected rapidly from the core, is not credible. The control rod drive mechanism housing is a Class 1 appurtenance per Reference 3.5-1.

3.5.1.3 Turbine Missiles

Regulatory Guide 1.115, “Protection Against Turbine Missiles,” provides guidance for limiting SSC requiring protection from postulated turbine missiles to those listed in Appendix A of the RG, which it refers to as “essential SSC.”

The design employs a barrier approach for protecting essential SSC against the effects of turbine missiles. Essential SSC requiring protection from turbine missiles are located in the RXB or CRB, and are listed in Table 17.4-1.

The Turbine Generator Building layout in relation to the overall site layout is shown on Figure 1.2-1. As shown in Figure 3.5-1, the turbine generator rotor shafts are oriented perpendicular to the RXB and CRB. However, portions of the RXB fall within the low-trajectory hazard zone, making the turbines unfavorably oriented as defined by RG 1.115. The CRB is located outside the low-trajectory turbine missile zone and is not assessed for low-trajectory turbine missiles. Essential SSC in the RXB are protected from low-trajectory turbine missiles by the exterior steel-plate composite (SC) wall and the wall panel located inside the exterior wall.

3.5.1.3.1 Reactor Building

The exterior wall of the RXB (Figure 3.5-2) is the first barrier credited for protecting essential SSC in the building. A global analysis and a local impact analysis show the missile does not penetrate the exterior wall in most of cases. If penetration occurs, investigations confirm essential SSC are protected.

3.5.1.3.2 Bounding Turbine Missile Properties

The bounding turbine missile is defined as one-half of the last-stage portion of the turbine rotor with the blades attached traveling at a speed based on a destructive overspeed of 190 percent. The weight is determined by summing the weight of the semicircular steel rotor portion with the weight of its blades.

Section 10.2 contains turbine generator design details used in the analysis.

3.5.1.3.3 Methodology

The assessment of the RXB structure for the effects of turbine missiles uses two focus areas: local and global.

The local turbine missile barrier assessment evaluates penetration and perforation where the inner steel plate prevents scabbing.

The global turbine missile barrier analysis evaluates the ability of the building and the exterior walls of the RXB to withstand the missile impact loading.

3.5.1.3.4 Acceptance Criteria

Acceptance criteria for local damage are:

1) Penetration

The acceptance criteria for turbine missile barriers in Subsection II.1.A of the Standard Review Plan (SRP) 3.5.3 suggests using empirical equations such as the modified National Defense Research Council (NDRC) formula and procedures for concrete structure analysis and design (Reference 3.5-3). Although the methodologies referenced in these documents do not have definitive limits on size or mass, the research and tests that inform the methodology do not use missiles greater than 300 pounds or velocities greater than 500 fps. These methods were developed for a steel slug, a piece of steel pipe, and a wooden pole. These objects are relatively small and light compared to a projectile the size of half of a turbine rotor. As such, the NRDC formula is limited with respect to calculating penetration distance. As an alternative, the evaluation uses a finite element analysis for predicting penetration distance in concrete instead of the NDRC formula.

2) Perforation, Spalling, and Scabbing

The RXB south and east walls are credited as barriers and the thickness requirement to protect against perforation is considered. Spalling that occurs at the exterior face, where the missile strikes the RXB, is not a concern for SSCs because it occurs outside the building. If the wall is not fully penetrated then back-face scabbing on the interior face inside the building is not a concern because the debris is restrained by the steel faceplate. The SC walls are analyzed to determine thickness necessary to prevent penetration. The steel plate thickness required to prevent perforation is increased by 25 percent per N690 (Reference 3.5-9). The analysis shows perforation does not occur.

Acceptance criteria for global damage:

An equivalent static calculation is used to assess the global effects of the bounding turbine missile.

3.5.1.3.5 Local Analysis

A local evaluation is performed for a turbine missile strike on the RXB exterior east wall. Based on the site layout and orientation of the turbines, it is postulated that the east wall and a portion of the south wall of the RXB are vulnerable to a turbine missile strike.

Three locations on the east wall are selected for local evaluation. These locations consist of the shortest load path, the stiffest region of the wall, and the least stiff region of the wall. These bounding locations are evaluated using a finite element model.

The finite element model results show the RXB exterior east wall meets the acceptance criteria from Section 3.5.1.3.4 except in the region of the shortest load path. This region shows a potential penetration in the wall. Further evaluation shows there are no essential SSC in the impact region and the missile is contained by interior walls after penetration.

3.5.1.3.6 Global Analysis

A global analysis is performed for a turbine missile strike on the RXB. This analysis converts the turbine missile to a static force. The static force is calculated using two different approaches. One approach uses Bechtel Topical Report, BC-TOP-9A, (Reference 3.5-5). The second approach considers using the Effect of Fragments Striking Structural Elements (Reference 3.5-10). The second approach results in a higher force and is used in the design. The wall is analyzed using the second approach for shear and moment forces resulting from a static point load and is then checked against SC walls available strength calculated per American Institute of Steel Construction (AISC) N690 (Reference 3.5-9).

3.5.1.3.7 Finite Element Analysis

The local analysis is performed with finite element analyses created in LS-DYNA using three models. The models encompass plausible strike zones on the east wall.

The models are generated with a coarse mesh around the perimeter and a refined mesh in the zone of impact and around openings. The exterior wall of the RXB is an SC wall composed of rib ties, face plates, and tie bars. The models use the standard spacing and materials defined for SC walls. The failure criterion is steel rupture.

The finite element analyses are benchmarked using experimental data and results from SMiRT-24 (Reference 3.5-11). The benchmark model

demonstrates the same failure mode of the test SC walls and provides similar missile penetration depths.

3.5.1.3.8 Final Required Barrier Thickness

The empirical equations endorsed by SRP 3.5.3 are used to determine the concrete barrier thickness necessary to prevent missile perforation through the barrier and scabbing of concrete material off the back face of the wall. The modified NDRC equations are used for perforation and scabbing in Section 3.5.3.1.1.2 and Section 3.5.3.1.1.3, respectively. Penetration depth calculated from the finite element analysis is used as input x. Missile diameter is determined from the projected area of the missile on the concrete target. For the RXB, this event occurs when the center of gravity of the missile is aligned with the target and results in an equivalent diameter of 21.6 inches.

3.5.1.3.9 Results

The RXB is evaluated for local and global effects.

The local analyses show the RXB east wall provides adequate protection against the design-basis turbine missile except when the missile strikes the doorway on the east wall at the 100-ft elevation in such a way that it impacts the wall opening side wall first, instead of the wall front faceplate. In this instance, the SC wall panel located along column line RX-5 of the RXB (Figure 3.5-2) is qualified as a missile barrier such that Seismic Category I SSC located west of column line RX-5 are protected.

The global evaluation that converts the turbine missile into a static force, as described in Section 3.5.1.3.6, shows the wall meets Reference 3.5-9 strength requirements and is not compromised by a missile impact.

COL Item 3.5-1: An applicant that references the NuScale Power Plant US460 standard design will demonstrate the site-specific turbine missile parameters are bounded by the standard design analysis, or provide a missile analysis using the site-specific turbine generator parameters to demonstrate that barriers adequately protect essential structures, systems, and components from turbine missiles. Parameters to verify are: limiting turbine missile spectrum (rotor and blade material properties); turbine rotor design, geometry and number of blades; final design of the Reactor Building exterior wall; and location of the turbines with respect to the Reactor Building and Control Building.

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. Missiles used in the evaluation are assumed to: be capable of striking from various directions and conform to the Region I missile spectrum in Table 2 of RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," for tornado missiles; and to Table 1 and Table 2 of RG 1.221, "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 3.3.1 and represent a probability of exceedance events of 1 x 10^{-7} per year for many potential sites.

The selected missiles are

- 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 lb 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 altitudes less than 30 ft above grade levels within 1/2 mile of the plant structures.

- a rigid missile that tests penetration resistance, such as a 6-inch diameter Schedule 40 pipe.

  The "pipe" missile is 6.625 inches in diameter by 15 feet long with a weight of 287 lb 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 1-inch diameter solid steel sphere to test the configuration of openings in protective barriers.

  The "sphere" missile is 1 inch in diameter and weighs 0.147 lb and has 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 224 ft/s and 85 ft/s, respectively, in a hurricane.

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

COL Item 3.5-2: An applicant that references the NuScale Power Plant US460 standard design will confirm the design-basis automobile missile parameters for the reference plant of velocity and maximum altitude of impact will not be exceeded as a result of extreme wind conditions that may occur in the vicinity of the site.

3.5.1.5 Site Proximity Missiles (Except Aircraft)

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

COL Item 3.5-3: An applicant that references the NuScale Power Plant US460 standard design will evaluate site-specific hazards due to external events, such as turbine failures that can occur at nearby or co-located facilities, which may produce more energetic missiles than the design-basis missiles defined in Section 3.5.1.

3.5.1.6 Aircraft Hazards

As described in Section 2.2, the design does not postulate 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 in Section 19.5.

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

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

The RXB and CRB meet guidance in RG 1.13,"Spent Fuel Storage Facility Design Basis," RG 1.117, "Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants," and RG 1.221 for protection of SSC from wind, tornado, and hurricane missiles.

3.5.3 Barrier Design Procedures

There are a limited number of potential internal missiles and a limited number of targets. Barriers are installed if a missile-and-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).

Safety-related and risk-significant SSC are protected from missiles by ensuring the barriers have sufficient thickness to prevent perforation, spalling, or scabbing of barriers. Missile barriers are designed to withstand local and global effects of missile impact loadings. The barrier design discussed below addresses 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. Unless stated otherwise, it is assumed 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 effects of missile impact resulting in penetration, perforation, and scabbing of the concrete using the Modified NDRC formulas discussed in Reference 3.5-3 as described in the following paragraphs. Concrete barrier thicknesses calculated using the equations in this section for perforation and scabbing are increased by 20 percent per F7.2.1 and F7.2.2 of ACI 349 (Reference 3.5-4).

Concrete thicknesses to preclude perforation or scabbing from the design-basis hurricane and tornado pipe and sphere missiles are calculated for the Reactor Building, Control Building, and RW-IIa portion of the Radioactive Waste Building external walls and roof using the equations below. Interior walls within the RWB identified as missile barriers to protect RW-IIa SSC have been evaluated and provide protection against extreme wind missiles.

Additional design characteristics of the RXB and the CRB are in Appendix 3B.2.

3.5.3.1.1.1 Penetration and Spalling Equations

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

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

$$x = KNW\left(\frac{V}{1000d}\right)^{1.8} + d \text{ for } \frac{x}{d} \geq 2.0$$

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/in^2).

3.5.3.1.1.2 Perforation Equations

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

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

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

3.5.3.1.2 Steel Barriers

Steel missile barriers are evaluated for the effects of missile impact resulting in penetration. The Stanford Formula (Reference 3.5-8, Equation 6.203) is used to determine the minimum steel thickness for a barrier to prevent perforation of the missile through the barrier. The Ballistic Research Laboratory (BRL) Formula (Reference 3.5-5, Equation 2-7) can be used to determine the minimum steel thickness for a barrier to prevent perforation of the missile if the results are comparable to the Stanford Formula results or if they are validated by penetration testing.

In using the Stanford and BRL formulae for missile perforation, it is assumed the missile impacts normal to the plane of the wall on a minimum impact area and, in the case of reinforced concrete, does not strike the reinforcing. Due to the conservative nature of these assumptions, the minimum thickness required for missile shields is taken as the thickness just perforated. Steel barrier thickness determined by the BRL formula is increased by 25 percent, per Reference 3.5-5. There are no steel missile barriers used in the design.

3.5.3.1.3 Steel-Plate Composite Walls

The required steel plate thickness to prevent perforation of SC walls from missile impact is calculated using equations based on Bruhl (Reference 3.5-7). A three-step approach is presented based on the failure mechanism described in the reference.

Step 1. Select an initial concrete wall thickness. Tc is the SC wall concrete infill thickness.

Step 2. Compute the weight, W_{CP}, and residual velocity, Vr, of the concrete conical plug dislodged by the missile after penetrating the SC wall.

Step 3. Calculate the required thickness, tp, of the rear steel faceplate to prevent its tearing fracture and thus perforation of the SC wall due to the concrete plug projectile.

3.5.3.2 Overall Damage Prediction

For impactive and impulsive loads, the design considers localized and global effects.

The forcing functions to determine the structural responses are Nuclear Engineering and Design, “Full-Scale Tornado-Missile Impact Tests,” (Reference 3.5-6) for the triangular impulse formulation of the design-basis steel pipe missile. Reference 3.5-5 is used for the design-basis automobile missile. The solid sphere missile is too small to affect structural response of the RXB and the CRB, and is not evaluated for its contribution to overall structural response.

Design for impulsive and impactive loads is in accordance with Reference 3.5-4 for concrete structures and NuScale Topical Report, TR-920-71621, Building Design and Analysis Methodology for Safety-Related Structures," (Reference 3.5-12), for SC walls, with the modifications listed below.

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

Concrete

Structural concrete members 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 Reference 3.5-4.

In Section F.3.5 of Reference 3.5-4, 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 Reference 3.5-4 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 Reference 3.5-4, 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 Reference 3.5-4.

3) The permissible ductility ratio should vary linearly from 1.0 to that given in F.3.3 or F.3.4 of Reference 3.5-4, for conditions between those 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.

The ductility factors from Section NB3.14, Table NB3.1 of Reference 3.5-9 are used in accordance with requirements of SRP 3.5.3.

In determining an appropriate equivalent static load, elasto-plastic behavior can be assumed with permissible ductility ratios as long as deflections do not result in loss of function of safety-related systems.

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, 2013 edition with no Addenda (subject to the conditions specified in paragraph (b)(1) of section 50.55a), Section III, "Rules for Construction of Nuclear Facility Components," New York, NY.

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

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

3.5-4 American Concrete Institute, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," ACI 349-06, Farmington Hills, MI.

3.5-5 Bechtel Power Corporation, "Design of Structures for Missile Impact," BC-TOP-9A, Revision 2, San Francisco, CA, September 1974.

3.5-6 Nuclear Engineering and Design, “Full-Scale Tornado-Missile Impact Tests,” Volume 45, Issue 1, March 1978, Pages 123-143.

3.5-7 Design of composite SC walls to prevent perforation from missile impact, Bruhl, J. C., et al., International Journal of Impact Engineering, 75 (2015) 75-87.

3.5-8 U.S. Reactor Containment Technology, Chapter 6, Volume 1, W.B. Cottrell and A.W. Savolainen, ORNL-NSIC-5, Oak Ridge National Laboratory, Oak Ridge, TN, 1965.

3.5-9 American Institute of Steel Construction, “Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities,” ANSI/AISC N690, 2018.

3.5-10 Williamson, R.A., and R.R. Alvy, Impact Effect of Fragments Striking Structural Elements, Holmes and Narver, Inc., Orange, CA, 1973.

3.5-11 Kim, J.M., et al., “Resistance of SC Walls Subjected to Missile Impact: Part 3. Small-Scale Tests,” International Association for Structural Mechanics in Reactor Technology Transactions, SMiRT-24, August 2017.

3.5-12 NuScale Power, LLC, “NuScale Topical Report: Building Design and Analysis Methodology for Safety-Related Structures,” NP-TR-0920-71621-NP-A, Revision 1.

Figure 3.5-1: Plan View of Partial NuScale Power Plant US460 Standard Design Showing Turbine Missile Trajectory

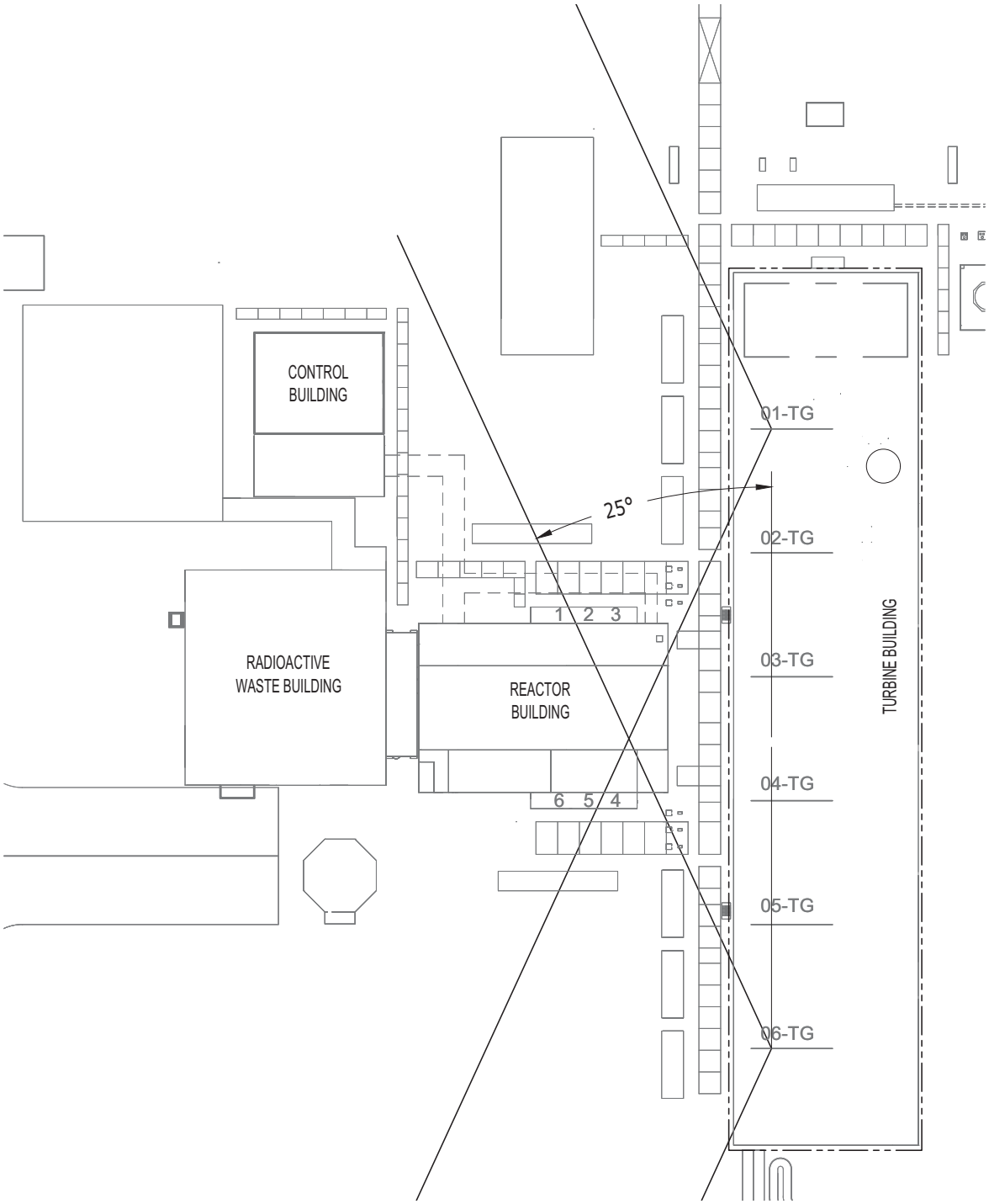
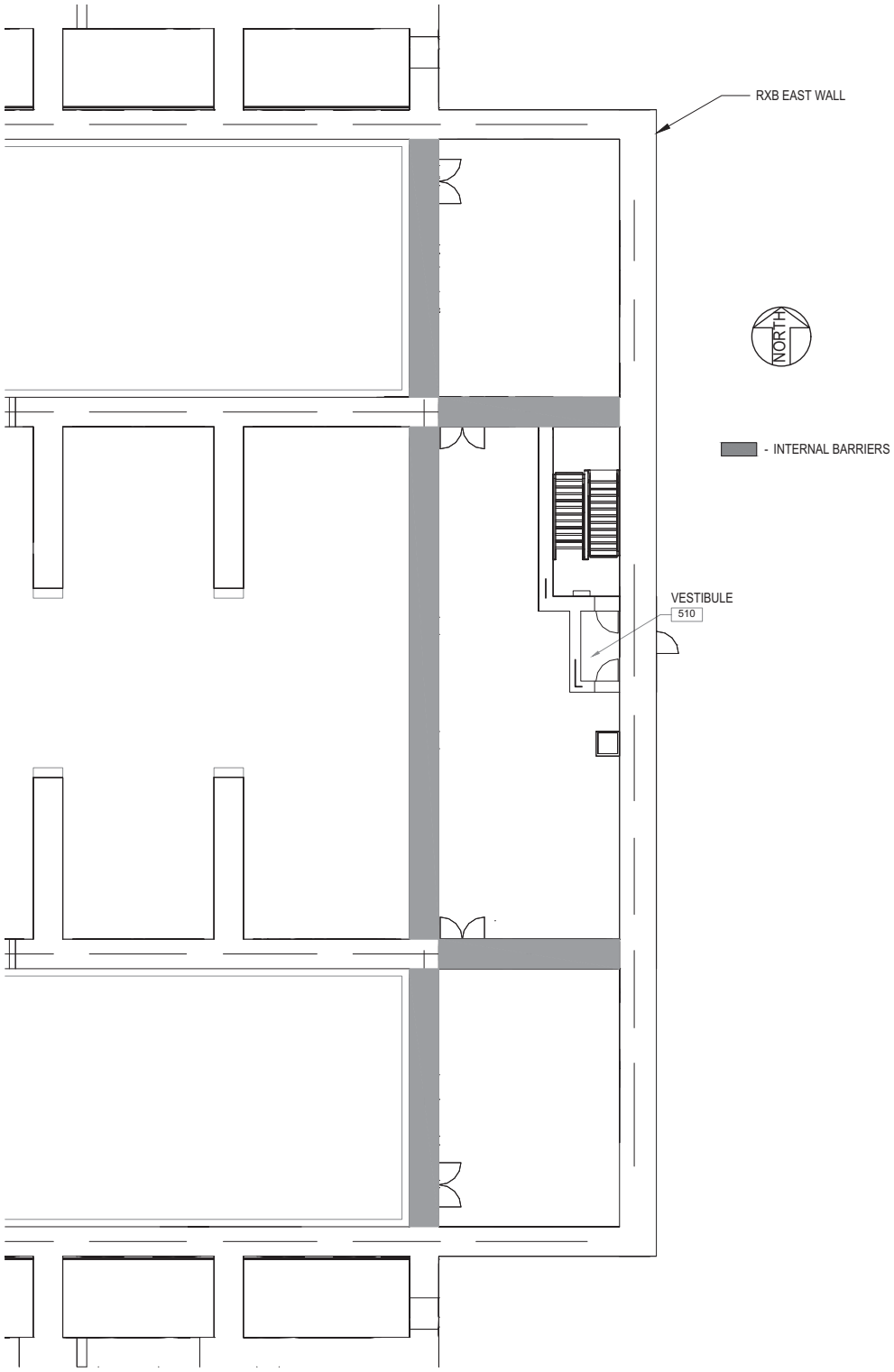


Figure 3.5-2: Section View of Reactor Building Showing Turbine Missile Barriers



3.6 Protection against Dynamic Effects Associated with Postulated Rupture of Piping

This section describes the design bases and measures to protect essential structures, systems, and components (SSC) inside and outside containment against the effects of postulated pipe ruptures. Figure 3.6-1 depicts the steps in the process for evaluation of potential line ruptures. The methodology applicable to identification and assessment of pipe rupture hazards addresses determination of postulated rupture locations, characteristics of ruptures, and assessment of the possible dynamic external effects of ruptures. Details of the analyses are provided in technical report TR-121507, "Pipe Rupture Hazards Analysis"

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 SSC are not unacceptably affected.

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

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

The piping systems that are considered include the high- and moderate-energy systems located inside and outside of the containment vessel (CNV). However, ruptures need not be postulated in lines that are NPS 1 and smaller.

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

Moderate-energy lines are evaluated for through-wall leakage cracks and for non-seismic lines, also evaluated for full circumferential breaks. Ruptures in moderate energy lines are analyzed for flooding and environmental effects.

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

This section primarily addresses the external dynamic effects of HELBs. Flooding is discussed in Section 3.4. Environmental effects are discussed in Section 3.11.

3.6.1.1 Identification of High- and Moderate-Energy Piping Systems

The areas of the plant that contain high- and moderate-energy lines or essential SSC are considered in five groups. Each is discussed in a separate section:

- inside the CNV (Section 3.6.1.1.1)

- outside the CNV under the bioshield (Section 3.6.1.1.2)
- in the Reactor Building (RXB), (outside the bioshield) (Section 3.6.1.1.3)
- in the Control Building (CRB) (Section 3.6.1.1.4)
- onsite (outside the RXB and CRB) (Section 3.6.1.1.5)

Table 3.6-1 provides a list of high- and moderate-energy piping systems inside the CNV and outside the CNV (under the bioshield) as these are the areas for which the evaluations are completed. Bounding evaluations and discussions of the other areas are also included, but are preliminary. Inspections, Tests, Analyses, and Acceptance Criteria confirms that the final pipe rupture hazards analysis demonstrates the acceptability of the dynamic and environmental effects associated with postulated ruptures in high-energy and moderate-energy piping systems within the NPM.

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

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

The containment flooding and drain system (CFDS) is a single open pipe inside containment that is normally isolated and not pressurized during operation. This line is identified as the CNTS flooding and drain line both inside and outside the CNV.

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

3.6.1.1.1 Inside the Containment Vessel

The high-energy lines inside the CNV are: main steam, feedwater, RCS injection, RCS discharge, high point degasification, PZR spray supply, and DHRS condensate return. The RCCWS supply and return lines are moderate energy and the CFDS line is open to the CNV environment and not considered high- or moderate-energy. (Table 3.6-1 shows high- and moderate-energy fluid system piping). The ECCS includes several small hydraulic lines inside containment that run between the ECCS main valves, the trip/reset valves,

and the RCS injection line. These high-energy ECCS lines are excluded from consideration as they are smaller than NPS 1.

3.6.1.1.2 Outside the Containment Vessel (Under the Bioshield)

The high-energy lines and the moderate-energy lines (RCCWS, CFDS, and the containment evacuation system (CES)) continue outside containment to the NPM disconnect flange (Table 3.6-1).

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

3.6.1.1.3 In the Reactor Building (Outside the Bioshield)

Within the RXB but outside the area under the bioshield, the high-energy lines with piping larger than NPS 1 include the MSS, FWS, MHS, and CVCS lines.

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

Once they exit the area under the bioshield, the high-energy CVCS lines run vertically downward in a pipe chase to the CVCS heat exchanger rooms and associated CVCS rooms.

Moderate-energy lines are routed throughout the RXB.

3.6.1.1.4 In the Control Building

There are no high-energy lines in the CRB, with the exception of the piping associated with the high pressure breathing air storage bottles in the control room habitability system.

3.6.1.1.5 Onsite (outside the Reactor Building and Control Building)

Outside of the RXB and CRB there are three high-energy lines: MSS, FWS, and extraction steam, and multiple moderate-energy lines.

Outside the RXB and CRB, the only essential or other protected SSC are located in underground safety-related tunnels, where they are separated from HELB effects.

3.6.1.2 Identification of Essential Structures, Systems, and Components

Essential SSC are primarily associated with the NPM, either inside the CNV or mounted on the top of the CNV head. Essential SSC in the design are a subset of

SSC classified as safety-related. Systems with safety-related SSC are listed below:

- Reactor core system including reactor fuel assembly
- In-core instrumentation system (ICIS)
- Reactor coolant system (RCS)
- Module protection system (MPS)
- Neutron monitoring system (NMS)
- Chemical and volume control system (CVCS)
- Control rod assembly (CRA) and control rod drive system (CRDS)
- Containment system (CNTS)
- Decay heat removal system (DHRS)
- Steam generator system (SGS)
- Emergency core cooling system (ECCS)
- Ultimate heat sink (UHS)
- Reactor Building (RXB) and Reactor Building components (RBCM)
- Control building (CRB)

In addition, post-accident monitoring (PAM) functionality for Type B and C variables (there are no Type A variables) is protected.

3.6.1.3 Characteristics of the NuScale Design

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

Specific examples of relevant design differences are:

- The response to HELB for a NuScale plant requires neither electric power nor injection of additional cooling water.
- The NPMs are partially immersed in a large pool of water that serves as the ultimate heat sink and does not require replenishment for at least 72 hours for design-basis events.
- Design-basis accidents do not require operator actions or re-establishing electric power for long-term cooling.
- Piping is small compared to the large reactors for which regulatory guidance was initially developed.

- The NPM containment is a pressure vessel designed and constructed to ASME Code Section III Class 1 requirements, versus a building in conventional LWRs.
- Piping of the NPM, including secondary system piping, is made of corrosion-resistant stainless steel.
- MSS and FWS piping inside the containment boundary and under the bioshield is designed to RCS design pressure and temperature.
- HELBs inside the CNV are limited to NPS 2 piping.
- The length of piping in which breaks must be postulated is minimal and the size of high-energy piping is small compared to current design reactors.
- The NPM containment is operated at a vacuum.
- Equipment and piping inside the NPM containment are not covered by insulation. This design is important for multiple reasons:
  - Jet impingement does not dislodge insulation that could lead to blockage of long-term-cooling recirculation.
  - Detection of small leakage cracks is not impeded by retention of moisture in insulation.
  - The bare piping is readily inspectable during refueling, because insulation does not need to be removed to observe deposits, discoloration, or other signs of degradation.
  - Corrosive substances (e.g., chlorides) cannot be trapped and held in contact with the piping surface.
- Essential components inside the NPM containment are qualified to be functional after exposure to saturated steam at containment design pressure of at least 1200 psia, requiring designs that are robust.
- The small NPM containment results in congestion that makes difficult the addition of traditional piping restraints and the separation of essential components from break locations, but whipping pipes in turn have a limited range of motion before encountering an obstacle.
- Containment isolation valves (CIVs) are outside of containment. Where two valves in series are required (i.e., for containment penetrations governed by GDC 55 and 56), both are in a single-piece valve body (i.e., no piping or welds between CIVs, precluding breaks in between). Except for the MS lines, there is also a containment isolation test fixture (CITF) valve between the CIVs and the CNV nozzles, but there is no physical piping (i.e., all valves are directly welded together and to the vessel safe ends).
- The RCS-connected lines (i.e., CVCS) each include an additional isolation valve outboard of the CIV body to preserve reactor coolant inventory in case of LOCAs outside containment. Injection and PZR spray include check valves, discharge includes an air operated valve, and high point degasification includes a solenoid valve.
- Active containment pressure suppression is not required.

- During a refueling, the NPM is disconnected from supporting systems by removal of piping spools, transported by crane to a refueling location, and disassembled. This process provides access for inspection to portions of the plant not normally accessible.
- Up to six NPMs are operating at the same time and in proximity, so the potential for a rupture in a system of one module to affect others is considered.
- The plant main control room is in a separate building that does not contain high energy piping, with the exception of the piping associated with the high pressure breathing air storage bottles in the control room habitability system.
- Dynamic effects of postulated HELBs that cause secondary breaks are evaluated, and protection for PAM capability (including augmented DC power) is provided by separation in different compartments within the building.

These unique characteristics affect choices about the means to address pipe ruptures.

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

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

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 rupture locations within a piping system.

The pipe break criteria of BTP 3-4 include the requirement that breaks be postulated at terminal ends.

General Design Criterion 4 allows dynamic effects associated with postulated pipe ruptures to be excluded from the design basis when analyses demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. This evaluation is referred to as LBB analyses and is discussed in Section 3.6.3. Similarly, breaks and leakage cracks may be excluded within the containment penetration area if criteria of BTP 3-4 B.1.(ii) are met.

3.6.2.1.1 Pipe Breaks Inside the Containment Vessel

A pipe rupture inside the CNV does not lead to containment bypass. Due to the tight configuration and the concentration of safety-related and essential

SSC inside the CNV, dynamic effects of pipe breaks are assessed for specific locations. The following strategies are employed for HELB inside containment:

- The SGS-MS, SGS-FW, and portions of the DHRS lines inside the CNV are included in the containment penetration area and evaluated using BTP 3-4 B.1.(ii) to eliminate the need to postulate terminal end breaks for these systems inside the CNV.
- The RCS injection, RCS discharge, PZR spray supply, and high-point degasification lines inside containment are NPS 2, ASME Class 1 stainless steel pipes. Due to their size, longitudinal breaks are not postulated. Circumferential breaks are postulated in accordance with BTP 3-4 Section B.1.(iii)(1). Breaks in Class 1 high-energy piping systems are postulated at the following locations:

  a) terminal ends (defined in Section 3.6.2.1)

  b) intermediate locations where the maximum stress range exceeds 2.4 S_m as calculated by equation (10) and either equation (12) or (13) of NB-3653 of Section III of the ASME Boiler and Pressure Vessel Code.

  c) intermediate locations where the cumulative usage factor exceeds 0.1, unless environmentally assisted fatigue is considered in which case the cumulative usage factor exceeds 0.4.

- The DHRS condensate lines inside containment run from each feedwater line, upstream of the feed plenum to the containment upper cylindrical shell penetration. These lines are NPS 2, and, along with MSS and FWS lines, are ASME Class 2. Due to their size, longitudinal breaks are not postulated. Circumferential breaks are postulated in accordance with BTP 3-4 Section B.1.(iii)(2) (except for the terminal end weld, which is included in the containment penetration area and evaluated per BTP 3-4 B.1.(ii)). Breaks in Class 2 high-energy piping systems are postulated at either terminal ends or at intermediate locations where stresses are calculated by the sum of equations (9) and (10) in NC-3653 of Section III of the ASME Boiler and Pressure Vessel Code to exceed 0.8 times the sum of the stress limits given in NC/ND-3653.

The RCCWS 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 high- and moderate-energy leakage cracks are bounded by the accident conditions inside the CNV. As a result, leakage cracks are not evaluated further inside containment. The CFDS line is open to the CNV environment and not considered high- or moderate-energy.

Final stress analysis is performed concurrent with fabrication of the first NPM. The postulated HELB locations are based upon the requirements of the ASME Section III Class 1, 2, and 3 Design Specification.

Inspections, Tests, Analyses, and Acceptance Criteria confirms that the final pipe rupture hazards analysis demonstrates the acceptability of the dynamic and environmental effects associated with postulated ruptures in high-energy and moderate-energy piping systems within the NPM.

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

The CIVs for the RCS injection, RCS discharge, PZR spray supply, and RPV high-point degasification lines are each dual, independent valves in a single body that is welded directly to a containment isolation test fixture (CITF) that is welded to a safe-end that is welded to the respective nozzle on the CNV head. These lines also include an additional isolation valve (check valve, air operated valve, or solenoid valve) outboard of the CIV body. The feedwater system CIV is similar, except there is a single isolation valve (in accordance with GDC 57) with a check valve as the outboard valve in the single piece body.

The MSS lines each have a single CIV. Between the CNV nozzle and the valve body is a safe-end tee to which the DHRS steam line attaches.

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

The containment penetration area is defined by regulatory guidance as the run of piping from the inside CIV to the outside CIV. This definition is not directly applicable to the design. The containment penetration area under the bioshield is defined as from the CNV safe-end to and including the weld connecting pipe to the outboard nozzle of the CIV. The DHRS piping located outside containment is within the containment penetration area.

For piping welds in the containment penetration area, provisions of BTP 3-4 Section B.1.(ii) are applied to preclude the need for breaks to be postulated, because they meet the design criteria of the Section III of the ASME Boiler and Pressure Vessel Code, Subarticle NE-1120 and the following criteria:

- Conservative stress and fatigue limits are met for these areas.
  - BTP 3-4 B.1.(ii)1.(a), (b), and (c) for ASME Class 1 piping (CVCS lines).
  - BTP 3-4 B.1.(ii).1.(d) and (e) for ASME Class 2 and 3, and B31.1 piping (MS, FW, and DHRS lines).
  - Although not explicitly considered in BTP 3-4 B.1.(ii), where piping is analyzed using the methods of NB-3200 rather than NB, NC, or ND-3600, the cumulative usage factor requirements of BTP 3-4 B.1.(ii)1.(b) are applied, while BTP 3-4 B.1.(ii).1.(a), (c), (d), and (e), which are based on stress equations in NB, NC, or ND-3600, are not applicable.
- There are no welded attachments for pipe supports.

- There is a minimum number of circumferential and no longitudinal welds in these lines in the containment penetration area.
- The length of the piping is the minimum practicable.
- There are no pipe anchors or restraints that require welding directly to the outer surface of the piping.
- Guard pipes are not used.
- The piping welds are included in the ISI program as described in Section 6.6, and the NPS 2 welds including and inboard of those of the pipe to outer nozzle welds of the CIVs are 100 percent volumetrically inspected, in addition to surface inspections as required by the ASME Boiler and Pressure Vessel Code Section XI.

The in-service inspection criteria of BTP 3-4 B.1.(ii)(7) are also applied to non-piping welds between valves, vessel safe-ends, and vessel nozzles that are located between the containment wall and break exclusion areas.

Outboard of the containment isolation valves, the CVCS NPS 2, RCS discharge, RCS injection, PZR spray supply, and high point degasification lines are ASME Class 3 lines to the next valve outboard of the CIV. The line transitions to ASME B31.1 outboard of the additional valve. At the first spool piece breakaway flange, the four lines become part of the CVCS. Piping under the bioshield, outboard of the containment penetration area including the refueling pipe spools, is designed to comply with BTP 3-4 Rev. 3 Paragraph B.1.(iii) to preclude breaks at intermediate locations by limiting stresses calculated by the sum of equations (9) and (10) in NC/ND-3653 of Section III of the ASME Boiler and Pressure Vessel Code to not exceed 0.8 times the sum of the stress limits given in NC/ ND-3653.

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

Decay Heat Removal System Lines

The DHRS is a closed loop system outside of the CNV that is entirely associated with a single NPM. Each NPM has two independent DHRS trains. Each train is associated with an independent steam generator (SG). The only active components in the DHRS are the DHRS actuation valves. The DHRS also relies on the MSS and FWS containment isolation valves to provide a closed loop system when it is activated. The DHRS is used to respond to transients including HELB outside containment. It is not used for normal shutdown, though the DHRS actuation valves are opened to allow slight circulation during wet layup of the SG. There is no flow through the DHRS system during normal operation. The DHRS is attached to the MSS line between the CNV and the MSS containment isolation valve. This portion of DHRS has two parallel actuation valves that are normally closed. These two

lines join into a single line that supplies the passive condenser. Each DHRS condenser is attached to the outside of the CNV. The condenser is designed as an ASME Class 2 component. A NPS 2 line exits the bottom of each DHRS condenser and penetrates the CNV. This line connects to the feedwater system inside containment. During operation, the DHRS is pressurized from the feedwater line. Section 5.4.3 provides additional discussion about the DHRS.

Breaks are not postulated in the DHRS piping outside containment in accordance with in BTP 3-4, B.1.(ii). Subject to certain design provisions, NRC guidance allows breaks associated with high-energy fluid systems piping in containment penetration areas to be excluded from the design basis. Though the DHRS piping extends beyond what would traditionally be considered a containment penetration area, this approach is chosen because the DHRS cannot be isolated from the CNV as there are no isolation valves.

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

3.6.2.1.2.1 Non-mechanistic Secondary Line Breaks in Containment Penetration Area

In accordance with BTP 3-3 B.1 (a)(1) for the NuScale design, the following considerations apply:

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

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

- The essential SSC in vicinity of MSS and FWS piping in the containment penetration area are CIVs, DHRS valves, and instrumentation cables and sensors.

- The CIVs are hydraulically held open against a stored energy device and shut upon a loss of power or a failure of the hydraulic line. The DHRS actuation valves similarly fail open.
- Failure of MSS and FWS piping is unlikely because:
  - Piping in the containment penetration area is made of stainless steel.
  - The physical length of MSS and FWS piping in the containment penetration area is zero (i.e., there are only valves and fittings).
  - MSS and FWS piping has a design pressure and temperature of 2200 psia and 650 degrees F, respectively, equivalent to the RCS piping.

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

Non-mechanistic breaks of MSS and FWS piping in the containment penetration area are evaluated, after consideration of the design differences from larger LWR plants. Comparing the typical PWR pipe MSS flow area to that of the design (NPS 30 to 38 vs NPS 12) yields a ratio of approximately one-thirteenth. On this basis, NuScale analyzes for environmental effects of an MSS non-mechanistic break with an area of 10.65 in^2, versus 1 ft^2 (144 in^2). The non-mechanistic FWS break size applied for the design is 5.87 in^2.

3.6.2.1.2.2 Break Exclusion

The design has both CIVs in a single valve body. There are no break locations between the valves. However, the welds between the valve body, containment isolation text fixture, and the CNV safe end are equivalent to those that break exclusion is intended to apply. Therefore, strict interpretation of the allowable extent of break exclusion would limit it to only a few welds in the design.The design extends this boundary inside and outside the CNV to include:

- The outboard weld at the CIV
- The SGS-MSS and SGS-FWS, and portions of the DHRS lines inside the CNV
- DHRS piping outside the CNV
- The ECCS main valve flange connections

Accordingly, the guidance of BTP 3-4 B.1.(ii) is used in piping design to ensure that breaks and leakage cracks can be excluded in the

containment penetration area. The BTP 3-3 non-mechanistic breaks of MSS and FWS piping are also addressed. For the remaining high-energy piping under the bioshield the analysis applies BTP 3-4 B.1.(iii) for ruptures and (v) for leakage cracks.

The length of piping and number of welds inside the CNV is limited. No primary or secondary piping other than about 65 feet of DHRS piping per train is within the break exclusion zone outside containment. The design pressure and temperature of MSS, FWS, and DHRS piping in the break exclusion zone is the same as for the RCS.

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

3.6.2.1.2.3 Leakage Cracks

Leakage cracks are excluded in containment penetration areas where the criteria of BTP 3-4 B.1.(ii) are satisfied.

For areas outside the containment penetration area, per BTP 3-4 Paragraph B.1.(v), leakage cracks are postulated in high-energy lines unless specific criteria are met. For Class 2 and 3 piping, the acceptance criterion is for the calculated stress to not exceed 0.4 times the sum of stress limits given in Subarticles NC/ND-3635. The BTP 3-4 B.3.(iii) specifies postulating leakage cracks with a flow area of one-half of a pipe diameter by one half pipe wall thickness in piping in the vicinity of essential SSC, regardless of system.

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

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

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

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

- blast, unconstrained pipe whip, and jet impingement caused by rupture of a main steam or feedwater pipe.

- dynamic subcompartment pressurization caused by main steam, feedwater, MHS, or CVCS breaks that are potentially limiting where they might occur in the building.
- multi-module impacts in common pipe galleries.

A break in a high-energy line in the RXB (outside of the bioshield) could potentially cause breaks or leakage cracks in smaller diameter or pipe schedule adjacent lines, including those of other NPMs, introducing an additional transient in a second NPM, and additional external effects. Therefore, in order to determine bounding dynamic effects and to ensure that the RXB structure is adequate for these scenarios, additional effects due to a break in conservatively selected smaller pipe are assumed. Once piping arrangements are finalized, the need for pipe whip restraints and barriers is determined to avoid additional breaks, if necessary.

Dynamic effects are addressed on a bounding basis and individual break locations are not specified. For flooding and environmental effects, discussed in Section 3.4 and Section 3.11 respectively, ruptures are postulated to occur anywhere on the line.

COL Item 3.6-1: An applicant that references the NuScale Power Plant US460 standard design will perform the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the reactor pool bay in the Reactor Building (RXB). This analysis includes an evaluation of multi-module impacts in common pipe galleries, and evaluations regarding subcompartment pressurization. The as-built Pipe Rupture Hazards Analysis (PRHA) will show that the analysis of RXB piping bounds the possible effects of ruptures for the routings of lines outside of the RXB, or will perform the PRHA of the high- and moderate-energy lines outside the buildings.

3.6.2.1.4 Pipe Breaks in the Control Building

There are no high-energy lines in the CRB, with the exception of the piping associated with the high pressure breathing air storage bottles in the control room habitability system. Flooding and environmental evaluations are described in Section 3.4 and Section 3.11, respectively.

3.6.2.1.5 Pipe Breaks Onsite (Outside the Reactor Building and Control Building)

There is no essential equipment outside of the RXB or CRB.

3.6.2.1.6 Types of Breaks

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

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

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

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

Longitudinal breaks are not applicable in the CNV, because piping is NPS 2 or meets break exclusion criteria. Also, longitudinal breaks are not considered under the bioshield, based on meeting criteria for not considering circumferential breaks. In the rest of the RXB, effects of longitudinal breaks (with break flow areas equal to the piping flow area) are bounded by circumferential breaks.

3.6.2.1.7 High- and Moderate-Energy Leakage Cracks

For high-energy lines, with the exception of those portions of piping exempted using the criteria in BTP 3-4 B.1.(ii) as described in Section 3.6.2.1.2 and

Section 3.6.2.7, leakage cracks are postulated at locations that result in the most severe environmental consequences unless otherwise selected by stress analysis. For lines where stress analysis is performed, postulated leakage crack locations are determined according to the criteria in BTP 3-4 B.1.(v). For ASME Code, Section III, Class 1 piping, leakage cracks are postulated at axial locations where the calculated stress range by Eq. (10) in NB-3653 exceeds 1.2Sm. For ASME Code, Section III, Class 2 and 3 piping, or nonsafety-class (i.e., non-ASME Class 1, 2, or 3), leakage cracks are postulated at axial locations where the calculated stress equal to the sum of Eq. (9) and Eq. (10) in NC/ND-3653 exceeds 0.4 times the sum of the stress limits given in NC/ND-3653.

For moderate-energy lines, leakage cracks are not postulated inside the containment or outside the containment under the bioshield. For the areas inside containment (described in Section 3.6.1.1.1) and outside containment under the bioshield (described in Section 3.6.1.1.2), the effects of leakage cracks in the moderate-energy RCCWS, CFDS, and CES are bounded by breaks in high-energy lines. In other areas of the plant, ruptures of moderate-energy lines are assumed at locations that result in the most severe environmental consequences. Environmental conditions are based upon the leakage cracks of the worst case (typically largest or hottest) line. Environmental effects are discussed in Section 3.11 and flooding analysis is described in Section 3.4.

3.6.2.2 Effects of High-Energy Line Breaks

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

3.6.2.2.1 Blast Effects

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

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

Key observations from this blast wave modeling are:

- A blast wave is weakly formed if the surrounding environment is at low pressure (less than 1 psia), as is the case inside the CNV. Buildup of pressure as blowdown progresses is not relevant, because the blast wave is a prompt and short-lived phenomenon.
- The severity of a blast depends on the amount of fluid that can escape within approximately one millisecond of break onset because the blast wave forms within that time.
- The high-energy, steam-filled lines are relatively small, which limits the severity of the blast pressure. The energy available to form the blast is less than one-seventeenth that for a typical large, light-water reactor.
- Blast waves are not significant for subcooled discharge, because liquid flashing occurs on time scales exceeding that of blast wave formation (Reference 3.6-8).
- A blast wave has well-defined and interrelated characteristics. For example, its peak pressure and speed decrease with distance from its origin.
- The pressure load applied by a blast wave is of short duration (i.e., an impulse load) and does not apply uniformly across large SSC at a given instant. Therefore, assuming the peak blast pressure is applied across the entire projected area of a component is inappropriate. The CFD analysis explicitly accounts for the time-varying pressure of the rapidly propagating blast wave.
- Reflection off surfaces can reinforce the pressure load, requiring consideration of plant-specific geometry. Angled or curved surfaces are loaded differently than a flat surface perpendicular to a line between the blast origin and surface. The pressures applied to surfaces by reflection can exceed the incoming wave pressure. For this reason, use of representative plant geometry is necessary. The CFD analysis includes the interaction of incident and reflected waves with each other and nearby surfaces, including how the shape and orientation of surfaces affect reflection.
- A small target has a lower peak pressure due to “clearing,” which is a phenomenon where some of the blast overpressure is relieved by bleeding off around the edge of the target. Because of both pressure-relieving clearing and the short load duration as a supersonic blast wave moves over them, small structures are not exposed to significant loading. The only SSC in the containment vessel or the Reactor Building that are large are the structures (e.g., CNV, RPV, and RXB walls and floors). The CFD analysis considers clearing.
- Several locations and directions of CVCS breaks in the CNV and MSS breaks in RXB pipe gallery are modeled. These locations and directions are selected to maximize blast pressure on nearby SSC (e.g., close to walls and corners) in order to bound final piping arrangements.

Blast analyses results show a maximum total force of 5980 lbf on a component in the CNV. In the RXB, the maximum total force is less than 10,000 lbf on a component, and bounded by jet impingement loads on the RXB walls. These forces are impulse loads that last only a few milliseconds or less.

In summary, three-dimensional CFD analysis of blast wave formation in the CNV and RXB is performed using modeling assumptions that bound the pressurization effects that occur for HELB in the plant. Blast wave force time histories are calculated for nearby SSC. The results show peak forces are low and bounded by the jet thrust forces that subsequently develop. The values are low because HELB are relatively small diameter and deposit only a small amount of mass and energy in the time it takes for a blast wave to form. The forces inside the CNV are low because the initial low ambient pressure does not support formation of a significant blast wave. The results also show the forces of the passing shock wave are of very short duration.

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

3.6.2.2.2 Pipe Whip

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

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

Inside the CNV, the largest pipe size subject to HELB conditions is NPS 2 and target SSC are robust (e.g., reactor vent valves (RVVs)). High-energy piping systems larger than NPS 2 are qualified for break exclusion inside the CNV. Outside the CNV, under the bioshield, piping satisfies the criteria of BTP 3-4 B.1.(ii) or (iii) to conclude that no breaks occur and that piping does not need to be evaluated for whip. However, nonmechanistic breaks of MSS and FWS lines and leakage cracks are considered. In the RXB outside the bioshield, main steam system, feedwater system, module heat up system, and chemical volume control system lines are subject to a postulated HELB, but there are only a limited number of SSC requiring protection.

The following considerations apply to evaluations of pipe whip:

- If the end is an RPV or CNV safe end, whip does not occur because the safe end and its nozzle are short, stiff, straight, and restrained by the component.

- In accordance with SRP Section 3.6.2, a pipe struck by another pipe of equal or smaller diameter and schedule (i.e., wall thickness) does not break or crack.
- The jet thrust necessary to cause pipe whip is determined. The calculation of thrust and jet impingement forces considers no line restrictions (e.g., a flow limiter) between the pressure source and break location, but does consider the absence of energy reservoirs, as applicable (e.g., the high-point vent pipe in the CNV is normally isolated).
- The length of pipe from the break location required for a plastic hinge is determined. If the jet thrust is insufficient to yield the pipe, then pipe whip at that break location is eliminated from further consideration.
- Pipe whip is considered to result in unrestrained motion of the pipe along a path governed by the hinge mechanism and the direction of the vector thrust of the break force. A maximum rotation of 90 degrees is assumed about a hinge. Pipe whip occurs in the plane defined by the piping geometry and configuration and initiates pipe movement in the direction of the jet reaction, as identified in BTP 3-3.

The RPV and CNV are robust structures with equivalent wall thicknesses exceeding that of the NPS 2 pipe that may whip inside the CNV.

In view of the SRP 3.6.2 provision for impact of a pipe on like-size or larger pipe, the RPV and CNV neither rupture nor crack if struck by a whipping NPS 2 pipe in the CNV. Because of the disparity in the thickness of the walls, the whipping pipe kinetic energy is absorbed in the bending and crushing of the pipe itself.

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

Ruptures above the NPM under the bioshield are excluded and there are few essential components exposed to whip elsewhere in the RXB. However, the RXB structural integrity must be assessed, so pipe whip impact force on concrete surfaces is determined. After break-opening, the steady-state jet thrust force, F_b, is:

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

where,

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

C_T = Thrust coefficient (unitless),

P_o = Internal system pressure (psia), and

A_e = Pipe flow area (in^2).

Applying the jet thrust force at the distance to the nearest restraint or anchor point determines the force available to overcome the pipe plastic bending moment and accelerate the pipe. If there is sufficient energy for whip to occur, the energy to yield the pipe is not deducted from its kinetic energy. Pipe whip evaluations have determined that impact on targets in the CNV (except for the CNV itself) is unlikely. Quantitative pipe whip impact evaluation is performed only for concrete walls and other structures in the RXB.

For an assumed pipe whip segment length and angle of travel that is bounding, whip of an MSS line (the highest-energy pipe in the RXB) is evaluated and does not perforate the SC wall with a thickness of 4 ft. and 3/4-in. thick steel face plates. Similarly, using the Sandia formula developed by Young (Reference 3.6-7), the FW line is evaluated to determine the penetration depth in the event of a strike against a building slab. The penetration depth is less than 56 percent of the slab depth. Given their smaller pipe size, chemical and volume control system pipe impacts are even less damaging.

3.6.2.2.3 Jet Impingement

Target SSC in the path of jets issuing from postulated breaks are assessed for the load imparted by the jet. In industry testing, single-phase steam jets with upstream pressures of 1200 psia are found to cause damage to pipe insulation at a distance of up to 25 times the pipe exit diameter (i.e., L/D = 25). However, insulation is fragile as evident from Reference 3.6-3, which reports types of insulation suffering damage due to impingement pressures as low as 4 psig.

NUREG/CR-6808 (Reference 3.6-4) Table 3-1 provides the impingement pressures found in testing to cause damage to various types of piping insulation used in U.S. pressurized water reactors. The damage pressures range from 4 to 40 psi for fibrous insulation to a high of 190 psi for two types of reflective metal insulation. Insulation is more fragile than the uninsulated solid metal surfaces of SSC inside the CNV. Therefore, jet impingement pressures need to be considerably above 190 psi to be of concern. As such, fewer uncertainties exist in predicting jet impingement effects on piping, and the relevant penetration distance is much shorter than 25 L/D.

Jet impingement testing was performed on electrical cable in support of the AP1000 assessment of debris generation. The conclusion was that cables at greater than or equal to 4 L/D from a jet simulating an AP1000 LOCA were not damaged. The results were given in terms of distance because of difficulty in accurately measuring impingement pressure. The NRC staff agreed with the conclusion.

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

For effects on the RXB structure, MSS breaks are limiting and are assumed to occur within 2 L/D of a wall, with no reduction in jet pressure with distance from the break. The maximum force of the jet results in a design-capacity ratio of wall for the jet impingement and reaction loading of 0.61.

An overview of the resistance to jet impingement is:

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

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

The impingement damage threshold of 190 psi is a sufficient measure of the structural integrity of components, but does not confirm functionality. Essential components inside the CNV are qualified for a CNV design condition of at least 1200 psia saturated steam. This value exceeds the 190 psi impingement acceptance threshold of 190 psia by a factor of more than six and is sufficient basis to consider functionality after jet impingement to be demonstrated.

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

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

- Liquid jets
- Two-phase jets
- Steam jets

As discussed for other effects, jet behavior and effects differ

- Inside the CNV: breaks are limited to NPS 2 RCS piping because SGS piping is classified as a “containment penetration area” under BTP 3-4. Only a degas line break would initially be steam, but spray line break reverse flow would almost immediately turn to steam. Other breaks such as injection line or spray line forward flow would be two-phase.
- In the NPM bay: no postulated breaks occur (nonmechanistic breaks do not require jet impingement evaluations) because piping satisfies break exclusion criteria of BTP 3-4 B.1.(ii) and (iii).
- In the RXB (outside the bioshield) piping arrangements are not finalized, so break locations and jet directions are assumed to be throughout in the rooms containing high-energy piping. The piping is limited to NPS 12 MSS, NPS 6 FWS, and NPS 3 CVCS and MHS piping at various pressures and temperatures. Main steam system jets are steam only, whereas FWS and CVCS breaks are liquid or two-phase.

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

Liquid jets

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

Two-phase jets

Two-phase jets are assessed using the methodology of NUREG/CR-2913 (Reference 3.6-6). A bounding approach is taken by identifying criteria for jet

formation in order to avoid the need to analyze individual break locations in the CNV and RXB.

- In the CNV

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

  The inputs for the NUREG/CR-2913 methodology are the system static thermodynamic conditions.

  Following the methodology, the relevant graph of Appendix A of NUREG/CR-2913 is selected to obtain target pressure and total force on the target for appropriate values of P_0, ΔT_0, or X_0, and distance to the target in L/D. For the CVCS breaks in the CNV, the thermodynamic conditions are 83.25 degrees K subcooling and 47 bar. The appropriate graph is Figure A.21, which shows pressures at specific points downstream in L/D and radially from the jet centerline in r/D. At the origin of the plot is the jet centerline at the break exit plane, and the shaded area at the lower left is the jet core (the region that has not yet begun to interact with the environment and in which fluid striking a target would experience full recovery of the fluid stagnation pressure). The letters A through C refer to the key for pressure (letters D and beyond for pressures above 5 bar are not plotted because they exist only near the jet core). For example, a letter C indicates pressure is 5.0 bar at 4 L/D and 0.75 r/D.

  The jet core is the region immediately downstream of a break in which the target pressure is the full stagnation pressure. Reference 3.6-4, Section 3.3.1.1 states that this region is significant only for jets involving subcooled stagnation conditions. Figure A.21 of NUREG/CR-2913 shows that the jet core dissipates within 3.72 L/D or about 6.3 inches for a thermodynamic condition similar to a chemical volume and control system HELB. This statement is viewed as conservative. Reference 3.6-1 Section 3.5.3.B notes that Sandia emphasizes the pipe exit core. The persistence of the core is attributed by Sandia to the time it takes for external pressure to penetrate the jet, and that the core length is longer than 0.5 D for subcooled and saturated water jets. Reference 3.6-1 notes, however, that test data are not consistent with the Sandia model, with only one or two test data sets exhibiting something like a liquid core while most data contradict the presence of a liquid core. Reference 3.6-1 concludes, “If a liquid core exists, it seems to be much smaller than indicated by Sandia.”

  Within 4 L/D or about 6.8 inches, the jet peak pressure has dropped to below 5.0 bar (72.5 psig). The A points representing 1.0 bar correspond to the edge of the jet. Only damage to fibrous insulation, at pressures as low as 4 psig, would be a concern beyond that. For the design, pressures at about 4 L/D are low enough to cause no damage to the hard components.

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

- In the RXB

  Similarly, for two phase jets in the RXB, the generic approach of a universal ZOI (i.e., a jet can reach any location in a room that houses a high-energy line) allows for breaks that bound final pipe routing and pipe whip considerations. Based on the discussion that follows for steam jets, FW, CVCS, and MHS pressure loading is not damaging.

Steam Jets

- In the CNV

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

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

  Applying the break exit pressure over this ZOI is an overestimation of the possible jet impingement loading. Therefore, the steam and two-phase jet pressure is assumed to decrease with distance proportional to the area of a jet that expands at a 30-degree half-angle to five pipe diameters and then at 10 degrees beyond that. A half-angle of 30 degrees is less than identified in the ANSI/ANS 58.2 Standard (Reference 3.6-2) and in other jet analyses for expansion into surroundings at normal atmospheric pressure. Thus, the jet pressure is below the 190 psi threshold for component damage at 2.3 L/D (3.9 in.). Although the NRC has challenged the general applicability of the ANSI/ANS Standard 58.2 spreading model, a half-angle of approximately 45 degrees or more is usually used. As the jet spreads more rapidly into the low-density CNV atmosphere, a 30-degree assumption is sufficiently conservative to bound actual jet impingement pressures due to local variation within the jet.

As noted, the jet core is only significant for subcooled jets. Reference 3.6-6 Section 3.6 discusses the core length L_c as ½D, one half of the pipe diameter for saturated stagnation conditions. It also notes that the length L_c depends on the time it takes a pressure wave to travel from the outer edge of the nozzle (i.e., break) to the jet center. Figure 4.3 of Reference 3.6-6 shows that for zero degrees subcooling L_c=½D. Thus, even if a jet core existed for a steam jet, its influence would be dissipated within ½D, which is too close for a jet impingement force to be of concern compared to pipe whip impact.

- In the RXB

  Other than the hydraulic actuator assembly skids, the only target SSC in the RXB are structural walls and nearby nonsafety-related pipes. The distance between a break and target SSC is not defined because RXB piping arrangements are not finalized. To verify suitability of the design of the RXB, bounding HELB scenarios have been identified.

  The MSS lines are much larger and contain more energy than other potential sources in the RXB. Demonstrating passing performance for MSS breaks provides confidence that final HELB analysis results are satisfactory. Therefore, a conservative approach is taken in which the jet impingement pressure is assumed to be the same as that at the break exit (i.e., no reduction for spreading with distance). For an MSS HELB, the break exit pressure is 700 psia, to which the thrust coefficient CT of 1.26 is applied. For an MSS break, which imposes the highest load of postulated HELBs in the RXB, the design-capacity ratio of the wall for jet impingement and reaction loading is 0.61. Because pipe rupture loads are localized, they have no effect on the overall structural integrity of the wall.

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

3.6.2.2.4 Dynamic Amplification and Resonance of Impingement Jet

Each of the following characteristics of postulated HELB is sufficient to ensure a resonance does not occur:

- A whipping pipe either 1) comes to rest against an object that intercepts a portion of the jet and distorts its axisymmetry or 2) flutters, causing a variation in the jet impingement angle and separation that prevents establishing synchronization of the transient waves.
- The break exit is distorted because of tearing as the break opens, which eliminates axisymmetry.
- No suitable (i.e., even, flat) impingement surfaces exist within the CNV. Relevant SSC are curved, which redirects reflected acoustic energy away from the break exit.

- The presence of a steam and water mixture in the jet acts to dampen pressure oscillations, preventing amplification.
- Splashing from the jet (and the jet from the opposite end of the break) interferes with the stability of the jet.

3.6.2.2.5 Subcompartment Pressurization

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

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

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

3.6.2.3 Protection Methods

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

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

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

3.6.2.3.1 Restraints, Barriers, and Shields

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

3.6.2.3.1.1 Pipe Whip Restraints

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

- Pipe whip restraints do not adversely affect structural margin of piping for other conditions:
  - Restraint design does not restrict thermal expansion and contraction.
  - The restraint design either: a) does not carry loads during normal operation or seismic events or b) the structural analysis includes a conservative load combination.
- Pipe whip restraints are located as close to the axis of the reaction thrust force as practicable. Pipe whip restraints are generally located so that a plastic hinge does not form in the pipe. If, due to physical limitations, pipe whip restraints are located so that a plastic hinge can form, the consequences of the whipping pipe and the jet impingement effect are further investigated. Lateral guides are provided where necessary to predict and control pipe motion. The Pipe Rupture Hazards Analysis technical report, TR-121507, provides further details.
- Generally, pipe whip restraints are designed and located with sufficient clearances between the pipe and the restraint, such that they do not interact and cause additional piping stresses. A design hot position gap is provided that allows maximum predicted thermal, seismic, and seismic anchor movement displacements to occur without interaction:
  - Exception to this general criterion may occur when a pipe support and restraint are incorporated into the same structural steel frame, or when a zero design gap is required. In these cases, the pipe whip restraint is included in the piping analysis and designed to the requirements of pipe support structures for all loads except pipe break, and designed to the requirements of pipe whip restraints when pipe break loads are included.
- In general, the pipe whip restraints do not prevent access required to conduct inservice inspection examination of piping welds. When the

location of the restraint makes the piping welds inaccessible for inservice inspection, a portion of the restraint is designed to be removable to provide accessibility.

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

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

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

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

3.6.2.3.1.2 Pipe Whip Barriers

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

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

3.6.2.3.1.3 Jet Impingement Shields

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

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

3.6.2.4 Guard Pipe Assembly Design Criteria for Piping in Containment Penetration Areas

Guard pipes are not used in the containment penetration area.

3.6.2.5 Analytical Methods to Define Forcing Functions and Response Models

Section 3.6.2.2 addresses conformance to this requirement.

3.6.2.6 Dynamic Analysis Methods to Verify Integrity and Operability

Section 3.6.2.2 addresses conformance to this requirement.

3.6.2.7 Implementation of Criteria Dealing with Special Features

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

In the design, each of two RVVs and two RRVs bolt directly to the reactor vessel. These four bolted-flange connections are classified as break exclusion areas. Because this configuration does not include a physical piping length, a majority of the BTP 3-4 B.1.(ii) criteria do not apply. However, these BTP 3-4 B.1.(ii) criteria generically involve design stress and fatigue limits and inservice inspection (ISI) guidelines, which are addressed for these bolted connections below.

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

Design Stress and Fatigue Limits

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

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

Faulty design is not a concern for the RVV and RRV flanges. The design features for flanged connections that affect the stresses in the bolts are primarily the

number and size of the bolts used, which are selected based on industry standards (ASME B16.5). The RVV and RRV flanged connections consist of Class 2500 NPS 5 and NPS 2 B16.5 flange configurations, respectively. In addition to conforming to an industry standard design, detailed analysis validates the design per ASME BPVC Section III, NB-3230, including a fatigue evaluation. The fatigue evaluation for these bolts utilizes the fatigue curve from ASME Section III, Division I, Mandatory Appendix I, Figure I-9.7. Figure I-9.7 was generated specifically for small diameter bolting made of SB-637 UNS N07718. Also, as required by NB-3230.3(c) for high strength bolting, a fatigue strength reduction factor of not less than 4.0 is applied to the bolts.

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

Unexpected modes of operation for piping systems in the nuclear industry generally involve thermal stratification, cycling, and striping. These situations do not apply to these valves. Unexpected vibration is another common concern, however, the RVVs and RRVs are within the scope of the Comprehensive Vibration Assessment Program (CVAP). As described in TR-121353, "NuScale Comprehensive Vibration Assessment Program Technical Report," (Reference 3.6-9) the CVAP ensures that structural components of the NPM exposed to fluid flow are precluded from the detrimental effects of flow-induced vibration (FIV).

Other degradation mechanisms that have contributed to past piping failures are addressed below. Included is an explanation why these mechanisms are less likely to occur in the RVV and RRV valves than in a typical piping system:

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

Inservice Inspection

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

Threaded Fastener Design

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

Leakage Detection

Leakage monitoring is provided by two means: the change in pressure within the CNV, and collected condensate from the containment evacuation system. Even under a scenario where leakage occurs due to one or more postulated bolt breaks, containment leakage monitoring systems are sensitive to a leak rate as low as 0.05 gallons per minute. Containment is a relatively small closed volume and is maintained at a pressure of less than 1 psia during normal operation. High containment pressure is also a safety actuation signal that initiates a reactor trip.

3.6.3 Leak-Before-Break Evaluation Procedures

For the NuScale Power Plant LBB methodology is not used. The design eliminates postulated breaks in those portions of the main steam and feedwater line by applying BTP 3-4 B.1(ii) containment penetration area design criteria. The LBB is also not applied to RCS-connected piping because of its small size (i.e., NPS 2).

3.6.4 High Energy Line Break Evaluation

3.6.4.1 Postulation of Pipe Breaks in Areas Other than Containment Penetration

Where break locations are selected without the benefit of stress calculations, breaks are postulated at the piping welds to each fitting, valve, or welded attachment. Breaks in non-ASME Class piping are addressed in Section 3.6.2.1.7. Additionally, in accordance with BTP 3-4, B.1.(iii)(4), if a structure is credited with separating a high-energy line from an essential SSC, that separating structure is designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect on the regardless that the criteria described in BTP 3-4, B.1.(iii)(1) through (3) might not require the postulation of a break at that location.

3.6.4.2 NuScale Power Module Piping System Parameters

High-energy piping systems (i.e., CVCS, MSS, FWS, and DHRS) are evaluated for HELB inside and outside the CNV. Although the DHRS condenser is manufactured from piping products, 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.

3.6.4.3 NuScale Power Module Piping Material

The high-energy piping systems of the NPM are manufactured using ASME SA-312, TP304 with a maximum carbon content of 0.03 percent and which are taken from ASME Section II, Materials. The low carbon content of the stainless steel exhibits the higher strength associated with the straight grade of TP304 SS with the low carbon content of TP304L. Thus, design pressure uses the strength properties from the straight TP304 SS grade at design temperature of 650 degrees F. Note that S_A is calculated with a 1.0 stress range reduction factor, f.

3.6.5 References

3.6-1 Marklund, Jan-Erik, Studsvik Energiteknik AB, “Evaluation of Free Jet and Jet Impingement Tests with Hot Water and Steam,” Studsvik/NR-85/54, May 21, 1985.

3.6-2 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, LaGrange Park, IL.

3.6-3 U.S. Nuclear Regulatory Commission, “Boiling Water Reactor ECCS Suction Strainer Performance Issue No. 7 - ZOI Adjustment for Air Jet Testing,” BWROG Meeting, July 20, 2011, Agencywide Document Access and Management System (ADAMS) Accession No. ML11203A432.

3.6-4 U.S. Nuclear Regulatory Commission, “Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance,” NUREG/CR-6808, February 2003.

3.6-5 Corradini, M., Advisory Committee on Reactor Safeguards, letter to Victor McCree, U.S. Nuclear Regulatory Commission, April 12, 2018, ADAMS Accession No. ML18102A074.

3.6-6 U.S. Nuclear Regulatory Commission, “Two-Phase Jet Loads,” NUREG/CR-2913, January 1983.

3.6-7 Sandia National Laboratories, "Penetration Equations," SAND-97-2426, Albuquerque, NM, October 1997.

3.6-8 Liu, J., et al., “Investigation on Energetics of Ex-vessel Vapor Explosion Based on Spontaneous Nucleation Fragmentation,” *Journal of Nuclear Science and Technology*, (2002): 39:1:31-39.

3.6-9 NuScale Power, LLC, "Comprehensive Vibration Assessment Program (CVAP) Analysis Technical Report," TR-121353, Revision 0.

Table 3.6-1: High- and Moderate-Energy Lines in the Containment Vessel and NuScale Power Module Bay

Code	**Abbr**.	Description	Energy
CNV			
A013	CNT	Containment System	N/A[1]
A014	SG	Steam Generator	High
A022	CRDS	Control Rod Drive System	Moderate
A030	RCS	Reactor Coolant System	High
B020	ECC	Emergency Core Cooling System	High / N/A[2]
B030	DHR	Decay Heat Removal System	High
NPM Bay			
A013	CNT	Containment System	High / Moderate[3]
B010	CVCS	Chemical and Volume Control System	High
B030	DHR	Decay Heat Removal System	High
B170	PCW	Pool Cooling and Cleanup Systems	Moderate
B190	CE	Containment Evacuation System	N/A[4]
B191	CFD	Containment Flooding and Drain System	Moderate[5]
B200	RCCW	Reactor Component Cooling Water System	Moderate
C010	MS	Main Steam System	High
C020	FW	Condensate and Feedwater System	High
C161	IA	Instrument and Control Air System	Moderate

Notes:

1. Includes CNT-CFDS line only. This line is open to the CNV environment (i.e., cannot be pressurized) and is considered to not be high or moderate energy.
2. ECCS tubing is smaller than NPS 1 and excluded from pipe rupture analysis; however, the ECCS main valves are considered high energy piping per TR-121507. The ECCS also contains NPS 2 lines that are open to the CNV environment (i.e., cannot be pressurized) and are not considered high- or moderate-energy.
3. Includes high energy CNTS-MS, CNTS-FW, CNTS-Injection, CNTS-Discharge, CNTS-PZR Spray, and CNTS Degasification lines, and moderate energy CNTS-RCCW Supply, CNTS-RCCW Return, CNTS-CFDS, and CNTS-CE lines.
4. This line operates below atmospheric pressure (i.e., vacuum) and is not considered high or moderate energy.
5. The peak water temperature during draining is 200°F. Higher temperatures could only occur if needed for post-accident combustible monitoring.

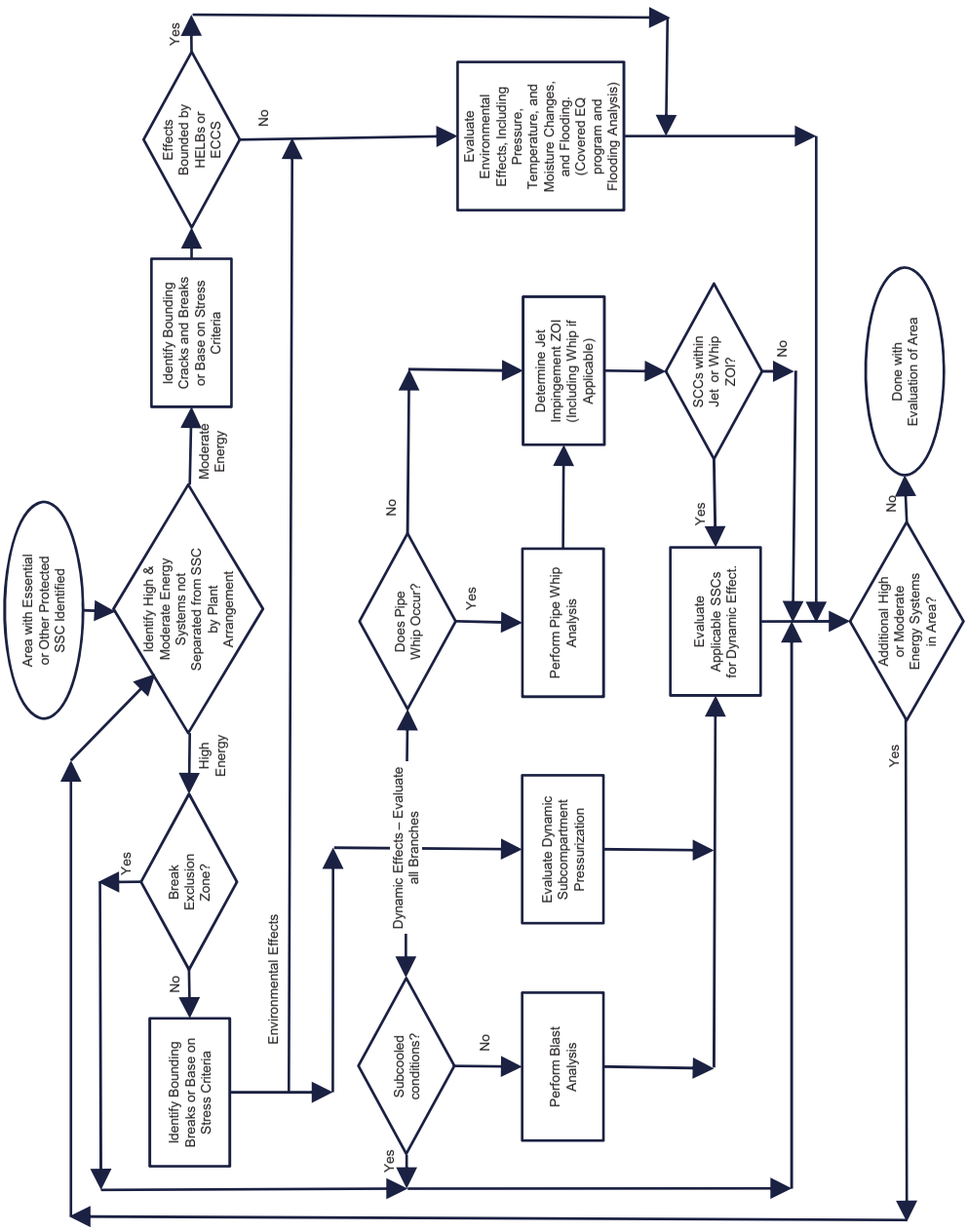


Figure 3.6-1: Flowchart of Methodology for Evaluation of Line Breaks

3.7 Seismic Design

Section 3.7.1 describes design parameters for seismic analysis. Section 3.7.2 describes the seismic analysis of the two site-independent Seismic Category I structures: portions of the Reactor Building (RXB) and portions of Control Building (CRB). Section 3.7.3 provides the seismic analysis of subsystems. Section 3.7.4 presents the instrumentation system for measuring effects of an earthquake.

Technical reports TR-121515 and TR-121517 (References 3.7.1-14 and 3.7.1-15, respectively) provide the seismic analysis of the NuScale Power Module (NPM). The NPM includes the reactor vessel, containment vessel, and associated structures, systems, and components (SSC).

The design complies with General Design Criterion 2 and 10 CFR 50, Appendix S because SSCs are designed to withstand effects of earthquakes without loss of capability to perform their safety functions. Software used for performing seismic analysis of SC-I SSCs conforms with the requirements for computer software in accordance with the NuScale Quality Assurance Program Description (QAPD) (Reference 3.7.1-12). In order to ensure the design is acceptable without modification at most sites, site-independent structures are designed using the Section 2.0 enveloping site parameters. With respect to earthquake design, two generic earthquake spectra and multiple generic soil profiles are used for the design of the site-independent Seismic Category I portions of the RXB and portions of the CRB.

The following is a discussion of the terms used in Section 3.7 and Section 3.8. These definitions are consistent with definitions in 10 CFR 50, Appendix S, Interim Staff Guidance ISG-001 (Reference 3.7.1-1), and other regulatory guidance documents.

Ground motion response spectra (GMRS) are site-specific ground motion response spectra characterized by horizontal and vertical response spectra determined as free-field motions on the ground surface or as free-field outcrop motions on the uppermost in-situ competent material using performance based procedures.

Safe shutdown earthquake (SSE) ground motion is the vibratory ground motion for which safety-related SSC are designed to remain functional. The SSE for a site is a smoothed spectra developed to envelop the GMRS. The SSE is characterized at the free ground surface.

Operating basis earthquake (OBE) ground motion is the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation remain functional. The operating basis earthquake ground motion is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input.

The OBE for the NuScale Power Plant is established as one-third of the SSE. Therefore, the OBE is not a design-basis ground motion for Seismic Category I structures and no specific analysis is required.

Foundation input response spectra (FIRS) is the performance-based site-specific seismic ground motion at the foundation level in the free field.

The ground motion response spectra, safe shutdown earthquake ground motion, operating basis earthquake, and foundation input response spectra are developed by the applicant. For evaluation of the site-independent RXB and CRB, the certified seismic design response spectra (CSDRS) (described below) is used instead of the FIRS.

Certified seismic design response spectra (CSDRS) are site-independent seismic design response spectra developed for design of the Seismic Category I and II Structures. The NuScale CSDRS consists of multiple sets of spectra, identified as the CSDRS and the CSDRS-high frequency (CSDRS-HF). The CSDRS are applied as an outcrop motion in the free-field at the foundation level of each building. Section 3.7.1.1.1.1 discusses development of the CSDRS.

Certified seismic design response spectra - high frequency (CSDRS-HF) is a seismic design spectra developed to envelop the GMRS of most hard rock sites. The CSDRS-HF has less low frequency (below ~10 Hz) and more high frequency (above ~10 Hz) content than the CSDRS. Section 3.7.1.1.1.2 discusses development of the CSDRS-HF.

3.7.1 Seismic Design Parameters

3.7.1.1 Design Ground Motion

3.7.1.1.1 Design Ground Motion Response Spectra

The CSDRS is a broad spectra (similar to Regulatory Guide [RG] 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” Revision 2) that is intended to encompass the GMRS at most selected sites. The CSDRS is used as a design basis for Seismic Category I SSC to withstand the effects of earthquakes without loss of the capability to perform their safety functions. However, the CSDRS does not bound hard rock sites in the central and eastern United States. To improve the range of acceptable locations, site-independent Seismic Category I structures, RXB, and CRB are also evaluated using a spectra that has more content above 10 Hz than the CSDRS. This spectra is identified as the CSDRS-HF. These spectra are described in more detail below.

3.7.1.1.1.1 Certified Seismic Design Response Spectra

Response spectra were developed to envelope most sites except for the highly seismic west coast sites and the central and eastern United States hard rock sites subject to higher frequency earthquakes. The response spectra are smooth broadband geometric mean spectra that were developed based upon expert panel recommendations and comparison to available industry data providing SSEs at existing and proposed reactor sites. The vertical component was developed independently of the horizontal component (i.e., the vertical component is not a ratio of the

horizontal component). The CSDRS bounds the RG 1.60 spectra anchored at 0.1g.

While similar, this spectra is not scaled from the RG 1.60 horizontal and vertical spectra. Instead, additional control points are established below 3.5 Hz and the control points above 3.5 Hz were shifted to higher frequencies. In addition, the vertical control point at 3.5 Hz was shifted to 4.5 Hz. Table 3.7.1-1 provides the horizontal and vertical control points for the CSDRS at 5 percent damping. Figure 3.7.1-1 compares the horizontal CSDRS at 5 percent damping against RG 1.60 spectra scaled to 0.1g. Figure 3.7.1-2 provides the same comparison in the vertical direction. Although not developed as a ratio, the vertical spectrum is two-thirds or more of the horizontal spectrum.

There are three components to the CSDRS. The two horizontal components, identified as north-south and east-west, are equivalent. The three components: north-south, east-west, and vertical are mutually orthogonal. All three components are developed at 5 percent damping. The horizontal components have a peak ground acceleration (PGA) of 0.5g and the vertical component has a PGA of 0.4g.

3.7.1.1.1.2 Certified Seismic Design Response Spectra - High Frequency

In order to address the high frequency, hard rock sites, a second response spectra was developed. The CSDRS-HF was developed based on expert panel recommendations and comparison with available hard rock high frequency response spectra data.

Like the CSDRS, the CSDRS-HF has three mutually orthogonal components (north-south, east-west, and vertical), with the horizontal components equivalent. The vertical component was not scaled from the horizontal component. It was also developed independently. Above 2 Hz, the vertical component is two-thirds or more of the horizontal spectra. Above 50 Hz, the vertical component is larger than the horizontal component. Table 3.7.1-2 provides the horizontal and vertical control points for the CSDRS-HF at 5 percent damping. Figure 3.7.1-3 compares the horizontal CSDRS and CSDRS-HF at 5 percent damping. Figure 3.7.1-4 provides the same comparison for the vertical direction.

3.7.1.1.1.3 Site Applicability

The CSDRS and CSDRS-HF can be compared against the preliminary GMRS data presented in the Nuclear Regulatory Commission (NRC) Memorandum "Support Document for Screening and Prioritization Results Regarding Seismic Hazard Re-Evaluations for Operating Reactors in the Central and Eastern United States" (Reference 3.7.1-2). By inspection, it can be seen that the CSDRS and CSDRS-HF provide a reasonable envelope for site conditions.

3.7.1.1.2 Design Ground Motion Time History

Six sets of time histories (each set consists of two horizontal and one vertical time history) were developed. Five of the time history sets conform with the CSDRS and the sixth set conforms with the CSDRS-HF. Each time history set was developed in accordance with ASCE/SEI 43-05, Section 2.4 (a) through (f) (Reference 3.7.1-3). This approach aligns with NRC Design Specific Review Standard 3.7.1 Subsection II.1B, Option 1, Approach 2. The CSDRS time histories are based upon the 1992 Landers earthquake, the 1989 Loma Prieta earthquake, the 1999 Chi-Chi earthquake, the 1999 Kocaeli earthquake, and the 1940 Imperial Valley earthquake. The CSDRS-HF time histories are based upon the 1992 Landers earthquake.

3.7.1.1.2.1 Seed Time Histories

Each seed time history is selected from actual acceleration time histories available from the Pacific Earthquake Engineering Research Center (PEER) ground motion database (Reference 3.7.1-4). The selection is based upon the intensity, duration, frequency content of the earthquake recording, and the epicenter distance from the recording station.

The acceleration recordings selected as seeds are described briefly below.

Yermo This set of time histories was recorded at the Yermo Fire Station during the 1992 Landers Earthquake, which occurred on June 28, 1992 at 04:57 am (11:57 coordinated universal time [UTC]), with an epicenter near the town of Landers, California. It was a magnitude 7.3 moment magnitude scale (MMS) earthquake. The time step is 0.02 seconds and the duration is 43.98 seconds and the maximum PGA recorded is 0.245g.

Figure 3.7.1-5a provides the unmodified Yermo acceleration, velocity, and displacement time histories and the response spectra scaled to the CSDRS in the east-west direction. Figure 3.7.1-5c and Figure 3.7.1-5e provide the same information in the north-south and vertical directions.

Capitola Recorded at station 47125 Capitola during the 1989 Loma Prieta Earthquake striking the San Francisco Bay Area of California on October 17, 1989 at 5:04 pm (October 18, 1989 at 00:04 UTC). It was a magnitude 6.9 MMS earthquake. The time step size of the recording is 0.005 seconds and the duration is 39.95 seconds. The maximum PGA recorded is 0.541g.

Figure 3.7.1-6a provides the unmodified Capitola acceleration, velocity, and displacement time histories and the response spectra scaled to the CSDRS in the east-west

direction. Figure 3.7.1-6c and Figure 3.7.1-6e provide the same information in the north-south and vertical directions.

Chi-Chi Recorded at station TCU076 during the 1999 Chi-Chi Earthquake striking central Taiwan on September 21, 1999 at 1:47 am (September 20, 1999 at 17:47 UTC). This earthquake is also known as the 921 Earthquake because it occurred on September 21. It was a magnitude 7.6 MMS earthquake. The time step size of the recording is 0.005 seconds and the duration is 89.995 seconds. The maximum PGA recorded is 0.416g.

Figure 3.7.1-7a provides the unmodified Chi-Chi acceleration, velocity, and displacement time histories and the response spectra scaled to the CSDRS in the east-west direction. Figure 3.7.1-7c and Figure 3.7.1-7e provides the same information in the north-south and vertical directions.

Izmit This set of time histories was recorded at Station Izmit during the 1999 Kocaeli Earthquake, which occurred on August 17, 1999 at 3:02 am (00:02 UTC) in northwestern Turkey. It was a magnitude of 7.4 MMS. The time step size of this recording is 0.005 seconds and the duration is recorded as 29.995 seconds. The maximum PGA recorded is 0.22g.

Figure 3.7.1-8a provides the unmodified Izmit acceleration, velocity, and displacement time histories and the response spectra scaled to the CSDRS in the east-west direction. Figure 3.7.1-8c and Figure 3.7.1-8e provides the same information in the north-south and vertical directions.

El Centro This set of time histories was recorded at station 117 El Centro Array #9 during the 1940 Imperial Valley Earthquake. This earthquake occurred on May 18, 1940 at 8:37 pm (May 19, 1940 at 05:35 UTC) in the Imperial Valley in southeastern Southern California. It was a magnitude 6.9 MMS earthquake. The time step size is 0.01 seconds and duration of 39.99 seconds. The maximum PGA recorded is 0.313g.

Figure 3.7.1-9a provides the unmodified El Centro acceleration, velocity, and displacement time histories and the response spectra scaled to the CSDRS in the east-west direction. Figure 3.7.1-9c and Figure 3.7.1-9e provides the same information in the north-south and vertical directions.

Lucerne These time histories were recorded at the Lucerne station during the 1992 Landers Earthquake which occurred on June 28, 1992 at 04:57 am (11:57 UTC), with an epicenter

near the town of Landers, California. The 1992 Landers earthquake was a magnitude 7.3 MMS earthquake. The duration of this recording is 48.12 seconds and the time-step size is 0.005 seconds. The maximum PGA recorded is 0.818g. Although this is the same earthquake as Yermo, a different recording station was selected to better match the CSDRS-HF.

Figure 3.7.1-10a provides the unmodified Lucerne acceleration, velocity, and displacement time histories and the response spectra scaled to the CSDRS-HF in the east-west direction. Figure 3.7.1-10c and Figure 3.7.1-10e provides the same information in the north-south and vertical directions.

3.7.1.1.2.2 Generation of Certified Seismic Design Response Spectra and Certified Seismic Design Response Spectra - High Frequency Compatible Time Histories

The numerical methodology devised by Lilhanand and Tseng (Reference 3.7.1-5) and later improved by N.A. Abrahamson (Reference 3.7.1-6) is used to generate CSDRS and CSDRS-HF compatible time histories. The methodology modifies an existing acceleration time history so that its response spectrum closely matches a target response spectrum. The methodology is described in detail in the above-mentioned references and was implemented in computer program RspMatch2009 (Reference 3.7.1-7). Further improvement was incorporated in RspMatch2009 for calculation efficiency and convergence stability by using a new adjustment function, which allows the use of analytical integration and readily integrates to zero velocity and displacement without additional baseline correction. A brief description of the process to develop spectrum compatible time histories is given below:

1) The time history is interpolated, if the time step is not 0.005 sec, to have 0.005 sec time steps.

2) The time history is scaled to get the response spectrum close to the target CSDRS.

3) The scaled time history is entered into RspMatch2009 as the seed acceleration, and the CSDRS (or CSDRS-HF) is entered as a target spectrum.

4) RspMatch2009 is used to add wavelets to the acceleration time history.

5) The modified acceleration time history is loaded into SAP2000 and a response spectrum generated at 100 frequencies, which are equally spaced in logarithmic scale, per frequency decade.

6) The resulting response spectrum is compared to the acceptance criteria (described in Section 3.7.1.1.2.3 below).

If necessary, additional refining passes (steps 4, 5, and 6) are rerun.

Comparisons of the modified Yermo time histories to the CSDRS are provided in Figure 3.7.1-5b for the east-west direction, Figure 3.7.1-5d for the north-south direction, and Figure 3.7.1-5f for the vertical direction. The equivalent information is provided in Figure 3.7.1-6b, Figure 3.7.1-6d, and Figure 3.7.1-6f through Figure 3.7.1-10f for the other time histories.

3.7.1.1.2.3 Confirmation and Checking of the Modified Time Histories

Cross Correlation Coefficients of Time Histories

The cross correlation between two components of each set of modified time histories was calculated using the method described in ASCE/SEI 43-05. The cross correlation coefficients are summarized in Table 3.7.1-3. As shown in the table, no cross correlation coefficient is greater than 0.16. Thus, the time histories are statistically independent.

Time increment and Duration

The six seed time histories all have durations that exceed 20 seconds. The Nyquist frequency used for development of CSDRS and CSDRS-HF compatible time histories is 100 Hz, resulting in a time increment of 0.005 seconds. The Yermo recording was in time steps of 0.02 seconds and El Centro was in time steps of 0.01 seconds. These were converted to 0.005 second time steps by linear interpolation.

Strong Motion Duration

The strong motion duration is defined as the time between 5 percent and 75 percent Arias intensity. Arias intensity plots for the modified Yermo time histories are provided in Figure 3.7.1-5b for the east-west direction, Figure 3.7.1-5d for the north-south direction and Figure 3.7.1-5f for the vertical direction. The equivalent information is provided in Figure 3.7.1-6b, Figure 3.7.1-6d, and Figure 3.7.1-6f through Figure 3.7.1-10a for the other time histories.

The strong motion durations are summarized in Table 3.7.1-4. All strong motion durations are greater than six seconds with exception of the north-south component of the modified Izmit recording, which is 5.265 seconds.

As shown in Figure 3.7.1-11, the normalized Arias intensity time history for the north-south component of the Izmit time history shows significant shaking for several seconds after 75 percent intensity is reached. The vertical black dashed lines show the time of the 5 percent (at 1.36 seconds) and 75 percent (at 6.625 seconds) Arias intensities. The

vertical green dashed line indicates the 6 second duration is achieved at about 80 percent Arias intensity. Strong shaking starts before the 5 percent time and continues after the 75 percent time. Thus, the strong motion duration of this component of the Izmit time history is acceptable.

Comparison to Target Response Spectra

The response spectra of the five CSDRS compatible time history sets were generated by SAP2000 (Reference 3.7.1-8) at more than 600 frequencies (i.e., 200 frequencies per decade evenly distributed in the logarithmic frequency scale and the seven frequency control points used to define the CSDRS). The total number of frequencies used is 607. The response spectra of the five CSDRS compatible time histories are compared with the CSDRS to examine the degree of compatibility.

No frequency point in any of the CSDRS compatible time histories is greater than 30 percent above the CSDRS and no point is more than 10 percent below the target. In addition, there are no instances where consecutive points in a +/-10 percent frequency window fall below the target response spectrum. The comparison data are tabulated in Table 3.7.1-5. Figure 3.7-12, Figure 3.7.1-13, and Figure 3.7.1-14 provide a visual comparison of the average of the five CSDRS compatible time histories to the CSDRS. As can be seen in these figures, the average is equal to, or slightly above, the CSDRS target in all three directions.

For the comparison of the Lucerne time histories to the CSDRS-HF, the quantity of frequencies generated varied by direction and decade. With the exception of the decade from 0.1 to 1 Hz, which had 85 frequency points in the vertical direction, all decades had more than 100 frequencies generated. For the CSDRS-HF, the frequency range of interest is 10 - 100 Hz. In this decade 362 frequencies were generated in the vertical direction.

No frequency point in the Lucerne time histories is more than 30 percent above the CSDRS-HF, and no point is more than 10 percent below the target. In addition, above 0.23 Hz, there are no instances where consecutive points in a +/-10 percent frequency window fall below the CSDRS-HF spectrum. The comparison data are tabulated in Table 3.7.1-5.

Power Spectra Density

Power spectra density (PSD) curves were created to ensure there are no gaps in the spectra. The PSD is a measure of the distribution of power in an accelerogram as a function of frequency. The one-sided PSD

computed from an accelerogram is defined in terms of Fourier amplitudes of the time history, F(ω), by the relation:

$$PSD(\omega) = \frac{2|F(\omega)|^2}{2\pi T_{sm}} \qquad \text{Eq. 3.7-1}$$

where T_{sm} is the strong motion duration.

As can be seen in Figure 3.7.1-15 and Figure 3.7.1-16, there are no gaps in the PSDs for any time histories.

3.7.1.1.2.4 Results

Based upon the above discussion, the modified time histories are valid representations of earthquakes that match the CSDRS and CSDRS-HF.

The five CSDRS compatible time histories and the CSDRS-HF compatible time histories are used for the design of the buildings. The building analysis outputs are used for the design of the bioshield and the reactor building crane.

3.7.1.1.3 Site-Specific Design Ground Motion

COL Item 3.7-1: An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific safe shutdown earthquake.

COL Item 3.7-2: An applicant that references the NuScale Power Plant US460 standard design will provide site-specific time histories. In addition to the above criteria for cross correlation coefficients, time step and earthquake duration, strong motion durations, comparison to response spectra and power spectra density, the applicant will also confirm that site-specific ratios V/A and AD/V^2 (A, V, D, are peak ground acceleration, ground velocity, and ground displacement, respectively) are consistent with characteristic values for the magnitude and distance of the appropriate controlling events defining the site-specific uniform hazard response spectra.

COL Item 3.7-3: An applicant that references the NuScale Power Plant US460 standard design will include an analysis of the performance-based response spectra established at the surface and intermediate depth(s) that take into account the complexities of the subsurface layer profiles of the site and provide a technical justification for the adequacy of vertical to horizontal (V/H) spectral ratios used in establishing the site-specific foundation input response spectra and the performance-based response spectra for the vertical direction.

3.7.1.2 Percentage of Critical Damping Values

3.7.1.2.1 System and Component Damping

Analyses of Seismic Category I and Seismic Category II SSC use the damping values of RG 1.61, Revision 1, "Damping Values for Seismic Design of Nuclear Power Plants." The technical reports TR-121515 and TR-121517 discuss damping used for the NPM subsystem (Reference 3.7.1-14 and Reference 3.7.1-15, respectively).

3.7.1.2.2 Structural Damping

The report TR-0920-71621-P-A (Reference 3.7.1-16) provides analytical models with damping values and stiffness properties based on the actual stress state of the members under the most critical seismic load combination.

3.7.1.2.3 Soil Damping

The dynamic properties of the soil and rock materials (i.e., the shear modulus and damping ratio) are dependent on shear strain levels induced during the shaking of an earthquake motion. The soil shear modulus decreases with the increase of soil shear strain, while the soil damping increases with the increase of soil shear strain. Soil degradation and damping functions were developed from 1993 Electric Power Research Institute data (Reference 3.7.1-9). These functions are shown in Figure 3.7.1-17 and Figure 3.7.1-18. For the half-space soil or rock, the shear wave velocities are assumed independent of the shear strain and the low-strain stiffness and strain-compatible damping of the soil layer above the half-space is used.

The numerical values of the shear modulus degradation and damping ratio curves as functions of the shear strains are tabulated in Table 3.7.1-6, Table 3.7.1-7 and Table 3.7.1-8. Because this site response analysis is not for a site-specific design, it is assumed that the soil site has a cohesionless soil and the extent of soil degradation varies with depth as shown in Table 3.7.1-6 and Table 3.7.1-7. However, for a rock site with a shear-wave velocity of 3500 fps or greater, the rock degradation shown in Table 3.7.1-8 is used regardless of depth. The maximum soil damping is limited to 15 percent.

Section 3.7.1.3 discusses damping values for soils as part of the creation of strain compatible properties for the generic soil profiles.

3.7.1.3 Supporting Media for Seismic Category I Structures

The design of the site-independent Seismic Category I structures is based upon three generic soil profiles. These soil profiles are not intended to represent the different soil profiles that may be encountered at actual sites. Rather, they represent the range of conditions (soft soil, rock, and hard rock) that could likely be encountered at a site.

The analysis considers four soil/earthquake combinations. The two relatively soft profiles (soft soil and rock) are evaluated in combination with the CSDRS. The rock profile is also evaluated for soil separation effects with CSDRS. The hard rock profile is evaluated in combination with the CSDRS-HF.

3.7.1.3.1 Generic Soil Profiles

The soil profiles used for the seismic analysis were selected from a larger pool of profiles. These profiles were initially identified as soil Type 1 through soil Type 12. This nomenclature remains, even though several of the original profiles were discarded because they produced results that were similar to, or bounded by, other profiles. The rock profiles tend to control the results. However, a soft soil profile is retained to ensure that those soil configurations are acceptable. The design envelope created by evaluating a broad range of soil conditions is sufficient to account for sites with lower water levels. For stability analysis, assuming high groundwater is a more conservative approach. Finite element models of the double building (RXB and Radioactive Waste Building [RWB]) are surrounded by backfill that extends at least 22.5 ft from the exterior faces of the building models. The backfill has the same properties as the Soft Soil Profile [Type 11].

Soft Soil Profile [Type 11]

Initial soil properties versus depth (shear wave velocity, soil unit weight, and Poisson's ratio) are provided in Table 3.7.1-9. This soil profile is shown in Figure 3.7.1-19.

Rock Profile [Type 7]

Soil properties versus depth (shear wave velocity, soil unit weight, and Poisson's ratio) are provided in Table 3.7.1-10. This soil profile is shown in Figure 3.7.1-19.

Hard Rock Profile [Type 9]

Soil properties versus depth (shear wave velocity, soil unit weight, and Poisson's ratio) are provided in Table 3.7.1-11. This soil profile is shown in Figure 3.7.1-19.

3.7.1.3.2 Strain Compatible Soil Properties

To obtain the strain compatible soil properties, site response analyses are performed. In the case of the RXB, the site response analysis is performed with the time histories applied as outcrop motions at the base of the RXB (for example, on top of the 15th layer in Table 3.7.1-12). The site response analysis is performed and the strain compatible properties are obtained as well as the in-layer motions at the base of the foundation for input to the SSI analysis of the double building model.

In the case of the CRB, the site response analysis is performed with the time histories applied at the surface, due to the shallow depth of the CRB base, and the strain compatible properties are obtained. The time histories to be applied at the surface for the SSI analysis of the CRB are the same as the outcrop motions.

The thickness and shear wave velocity of a soil layer determines the maximum frequency of a seismic wave that can accurately be calculated to pass through that soil layer in the SSI analysis. A rule of thumb for the relationship among these three parameters is given by Eq. 3.7-2.

$$h \leq \frac{1}{5}\frac{V_s}{f_{pass}} \qquad \text{Eq. 3.7-2}$$

where,

f_{pass} is the maximum frequency that can pass through the soil layer,

V_S is the shear wave velocity, and

h is the layer thickness.

To ensure that high frequency motion is adequately transferred to the structure, layers of 6.25-ft thickness were used between the surface and the base of the RXB and 5-ft-thick layers were used to the 300 foot depth.

Initial detailed soil properties for the site response analysis were developed by using these thicknesses and interpolating the original data presented in Table 3.7.1-9, Table 3.7.1-12, Table 3.7.1-10, and Table 3.7.1-11, and incorporating the soil damping and shear modulus information presented in Table 3.7.1-6, Table 3.7.1-7 and Table 3.7.1-8. The low-strain shear wave velocities for the soil types are shown in Figure 3.7.1-19. The densities are shown in Figure 3.7.1-20.

For analysis, the water table is assumed to be at the grade level. For saturated soil, a P-wave velocity, V_P, of 5000 fps is used. The exception is when it must be adjusted to limit the Poisson's ratio to 0.48. The maximum soil damping is limited to 15 percent.

The site response analysis of the RXB is performed using computer program SHAKE2000 (Reference 3.7.1-10). The site response analysis of the CRB is performed using the computer program ACS SASSI (Reference 3.7.1-13). The nonlinear soil behavior is approximated by the equivalent linear technique described in "SHAKE, A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites," (Reference 3.7.1-11). The SHAKE2000 program and the SOIL module in ACS SASSI calculate one-dimensional analysis of a layered soil profile subjected to a seismic wave

propagating in the vertical direction. Only one acceleration component can be applied as the excitation input in each analysis.

The nonlinear soil properties are defined by soil shear strain dependent shear moduli and damping ratios for each layer. Using the strain-dependent shear modulus degradation curves and strain-dependent damping curves, the iterative procedure implemented in the computer program SHAKE2000 and SOIL module of ACS SASSI calculate the strain-compatible soil properties in terms of shear moduli (or shear wave velocities) and damping ratios for all layers.

The following steps were used to obtain a single set of the strain-compatible soil properties of a soil profile for all three excitation components:

Step 1. Perform initial shake analysis, using SOIL module of ACS SASSI for CRB and SHAKE2000 for double building, for the first S-wave excitation, designated as SV, using the east-west acceleration time history as the input motion. Soil property iteration is required. This step calculates the strain-compatible soil properties due to the first horizontal excitation.

Step 2. Perform initial shake analysis, using SOIL module of ACS SASSI for CRB and SHAKE2000 for double building, for the second S-wave excitation, designated as SH, using the north-south acceleration time history as the input motion. Soil property iteration is required. This step calculates the strain-compatible soil properties due to the second horizontal excitation.

Step 3. Average the strain-compatible properties obtained in Steps 1 and 2. This step calculates final strain-compatible soil properties applicable to the horizontal excitation components (i.e., east-west and north-south).

Step 4. Perform the final shake analyses, using SHAKE2000 for double building, for the SV and SH excitations using the averaged strain compatible soil properties obtained in Step 3. No iteration of soil properties is required. This step calculates the in-layer horizontal acceleration response time histories that are used as the horizontal input excitations (east-west and north-south) in the SSI analysis. This step is not needed for the CRB since, at the surface, the in-layer motions are the same as the outcrop motions.

Step 5. Perform the final shake analysis, using SHAKE2000 for double building, for the vertical excitation, designated as PV. The soil properties in terms of the P-wave velocities, V_P, of all layers are required for the vertical excitation analysis. The P-wave velocities are calculated as described below. The same strain-compatible soil damping ratios for all layers obtained in Step 3 are used. No iteration of soil properties is required. This step produces the

in-layer vertical site response time histories used in the SSI analysis. For the CRB, the P-wave velocities need to be calculated but the site response analysis does not need to be run.

For the calculation of site responses from the vertical excitation, the confined moduli (or P-wave velocities) are used in Step 5 instead of using the strain-compatible shear moduli (or S-wave velocities). The calculation of V_P is described below.

Calculate the shear wave velocity for each layer based on its strain-compatible shear modulus G and soil density ρ as:

$$V_s = \sqrt{\frac{G}{\rho}} \qquad \text{Eq. 3.7-3}$$

where G is the shear modulus calculated in Step 3, and ρ is the soil density calculated as the unit weight, γ, divided by gravity constant, g.

Calculate the P-wave velocities for each layer using the following formula:

$$V_p = V_s\sqrt{\frac{2(1-\nu)}{(1-2\nu)}} \qquad \text{Eq. 3.7-4}$$

where ν is the Poisson's ratio of the soil layer. The Poisson's ratio can be calculated for a pair of known V_S and V_P as follows:

$$\nu = \frac{2\left(\frac{V_s}{V_p}\right)^2 - 1}{2\left(\frac{V_s}{V_p}\right)^2 - 2} \qquad \text{Eq. 3.7-5}$$

A minimum P-wave velocity of 5000 fps is used because the soil layer is below the water table. In using the V_P of 5000 fps for a saturated soil, the Poisson's ratio should be recalculated for V_P of 5000 fps using Eq. 3.7-5. If the Poisson's ratio exceeds 0.48, the saturated V_P is recalculated using 0.48 for the Poisson's ratio in Eq. 3.7-4. The limit of 0.48 for the Poisson's ratio is a limitation of the SSI analysis.

Step 6. Perform final shake analysis using SHAKE2000 for double building, as described in Steps 4 and 5. Use final strain compatible properties to get the in-layer motion at the bottom of the RXB

basemat for input to double building soil-structure interaction analysis.

This step is not necessary for the CRB because the CSDRS is being input at the surface. The in-layer motion at the surface and the outcrop motion at the surface for the CRB are the same.

For each soil type, the strain-compatible properties associated with each of the five CSDRS compatible time histories are averaged so that a single set of soil properties may be used per soil type. These average strain-compatible soil properties are presented in Table 3.7.1-12 and Table 3.7.1-13. There is only one set of CSDRS-HF compatible time histories, so no averaging is performed. The strain-compatible properties for the rock profiles are presented in Table 3.7.1-14 for Soil Type 9.

Average V_S profiles are combined into a single plot, shown in Figure 3.7.1-21, for the CSDRS compatible profiles. The CSDRS-HF compatible V_S profiles are provided in Figure 3.7.1-22.

Figure 3.7.1-23 and Figure 3.7.1-24 illustrate the strain compatible damping for the soil types used with the five CSDRS compatible time histories. Figure 3.7.1-25 combines the average damping ratios for all soil types on a single plot. Figure 3.7.1-26 shows the strain compatible damping for Soil Type 9.

The average strain-compatible soil properties of the CSDRS compatible inputs have previously been shown in Figure 3.7.1-21 for shear wave velocities and Figure 3.7.1-25 for damping ratios.

3.7.1.3.3 Site-Specific Soil Profile

COL Item 3.7-4: An applicant that references the NuScale Power Plant US460 standard design will:

- develop a site-specific strain-compatible soil profile.
- confirm that the criterion for the minimum required response spectrum is satisfied.
- determine whether the seismic site characteristics fall within the seismic design parameters such as soil layering assumptions used in the standard design, range of soil parameters, shear wave velocity values, and minimum soil bearing capacity.

3.7.1.4 References

3.7.1-1 U.S. Nuclear Regulatory Commission, "Interim Staff Guidance on Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications," ISG-001, May 2008.

3.7.1-2 U.S. Nuclear Regulatory Commission, "Support Document for Screening and Prioritization Results Regarding Seismic Hazard Re-Evaluations for Operating Reactors in the Central and Eastern United States," Memorandum, Agencywide Documents Access and Management System (ADAMS) Accession No. ML14136A126, May 21, 2014.

3.7.1-3 American Society of Civil Engineers/Structural Engineering Institute, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," ASCE/SEI 43-05, Reston, VA.

3.7.1-4 Pacific Earthquake Engineering Research Center, PEER NGA Strong Motion Database, http://peer.berkeley.edu/nga/, University of California, Berkeley, CA, 2013.

3.7.1-5 Lilhanand, K. and W.S.Tseng (F.H. Wittmann, ed.), "Generation of Synthetic Time Histories Compatible with Multiple-Damping Response Spectra," Biennial international conference on structural mechanics in reactor technology (SMiRT-9), Lausanne, Switzerland, 1987.

3.7.1-6 Abrahamson, N.A., "Non-Stationary Spectral Matching," Seismological Research Letters, (1992): 63:1:30.

3.7.1-7 RspMatch2009 [Computer Program]. (2011). Lacey, WA: GeoMotions, LLC.

3.7.1-8 SAP2000 (Advanced Version 17.1.1) [Computer Program]. (2014). Berkeley, CA: Computers and Structures, Inc.

3.7.1-9 Electric Power Research Institute, "Guidelines for Determining Design Basis Ground Motions," EPRI #102293, Palo Alto, CA, 1993.

3.7.1-10 Ordonez, G. A., SHAKE2000, Version 9.98.0, "A Computer Program for the 1-D Analysis of Geotechnical Earthquake Engineering Problems," User's Manual, April 2013.

3.7.1-11 Schnabel, P.B., J. Lysmer, and H.B. Seed, "SHAKE, A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites," EERC Report No. 72-12, University of California, Berkeley, 1972.

3.7.1-12 NuScale Power, LLC, "Quality Assurance Program Description," MN-122626, Revision 0.

3.7.1-13 ACS SASSI Version 4.3.1, Ghiocel Predictive Technologies, Inc., Pittsford, New York.

3.7.1-14 NuScale Power, LLC, "US460 NuScale Power Module Seismic Analysis," TR-121515, Revision 0.

3.7.1-15 NuScale Power, LLC, "NuScale Power Module Short-Term Transient Analysis," TR-121517, Revision 0.

3.7.1-16 NuScale Power, LLC, "Building Design and Analysis Methodology for Safety-Related Structures," TR-0920-71621-P-A, Revision 1.

Table 3.7.1-1: Certified Seismic Design Response Spectra Control Points at 5 Percent Damping

Horizontal (north-south and east-west)		Vertical	
Frequency (Hz)	**Acceleration (g)**	**Frequency (Hz)**	**Acceleration (g)**
0.1	0.024	0.1	0.016
0.25	0.15	0.25	0.1
1	0.60	1	0.40
3.5	1.15	4.5	1.06
12	1.15	16	1.06
50	0.50	50	0.40
100	0.50	100	0.40

Note:
Log-log interpolation is used between the frequencies listed in the table.

Table 3.7.1-2: Certified Seismic Design Response Spectra - High Frequency Control Points at 5 Percent Damping

Horizontal (north-south and east-west)		Vertical	
Frequency (Hz)	**Acceleration (g)**	**Frequency (Hz)**	**Acceleration (g)**
0.1	0.01	0.1	0.01
0.2	0.05	0.3	0.04
0.3	0.08	0.5	0.09
0.5	0.12	1	0.1
1	0.16	2	0.18
1.8	0.25	3.8	0.24
3.7	0.35	4.6	0.29
5	0.43	11	0.76
8	0.9	20	1.0
20	1.5	30	1.2
25	1.6	50	1.3
30	1.5	100	0.52
50	1.0	-	-
100	0.52	-	-

Note:
Log-log interpolation is used between the frequencies listed in the table.

Table 3.7.1-3: Cross-Correlation Coefficients

Original Recording (target)	Modified Acceleration Component 1	Modified Acceleration Component 2	Cross-correlation Coefficient
Yermo (CSDRS)	North-south	East-west	0.0103
	East-west	Vertical	0.0159
	North-south	Vertical	0.0258
Capitola (CSDRS)	North-south	East-west	0.0277
	East-west	Vertical	0.0219
	North-south	Vertical	0.0862
Chi-Chi (CSDRS)	North-south	East-west	0.0951
	East-west	Vertical	0.0231
	North-south	Vertical	0.0811
Izmit (CSDRS)	North-south	East-west	0.0888
	East-west	Vertical	0.0473
	North-south	Vertical	0.0798
El Centro (CSDRS)	North-south	East-west	0.0071
	East-west	Vertical	0.0561
	North-south	Vertical	0.0490
Lucerne (CSDRS-HF)	North-south	East-west	0.0259
	East-west	Vertical	0.1162
	North-south	Vertical	0.0141

Table 3.7.1-4: Duration of Time Histories

Original Recording (Target)	Component	No. of Data Points	Time Step Size (sec)	Duration (sec)	T_{05} (sec)	T_{75} (sec)	Strong Motion Duration (T_{75} - T_{05}) (sec)
Yermo (CSDRS)	East-west	8802	0.005	44.005	13.075	22.245	9.170
	North-south	8802	0.005	44.005	12.945	21.140	8.195
	Vertical	8802	0.005	44.005	7.320	18.470	11.150
Capitola (CSDRS)	East-west	7992	0.005	39.955	4.135	10.910	6.775
	North-south	7992	0.005	39.955	3.975	10.875	6.900
	Vertical	7992	0.005	39.955	3.415	10.885	7.470
Chi-Chi (CSDRS)	East-west	13854	0.005	69.265	5.965	19.540	13.575
	North-south	13854	0.005	69.265	4.515	22.680	18.165
	Vertical	13854	0.005	69.265	3.295	18.995	15.700
Izmit (CSDRS)	East-west	6000	0.005	29.995	2.930	11.340	8.410
	North-south	6000	0.005	29.995	1.360	6.625	5.265*
	Vertical	6000	0.005	29.995	1.985	9.960	7.975
El Centro (CSDRS)	East-west	8004	0.005	40.015	2.095	16.545	14.450
	North-south	8004	0.005	40.015	1.575	10.895	9.320
	Vertical	8004	0.005	40.015	1.885	8.000	6.115
Lucerne (CSDRS-HF)	North-south	9625	0.005	48.12	7.510	16.240	8.730
	East-west	9625	0.005	48.12	7.665	16.185	8.520
	Vertical	9625	0.005	48.12	6.270	16.555	10.285

Note:
* This is acceptable as explained in Section 3.7.1.1.2.3

Table 3.7.1-5: Comparison of Response Spectra to Certified Seismic Design Response Spectra and Certified Seismic Design Response Spectra - High Frequency

Original Recordings	Component	Frequency Decade	Number of Freq. in Response Spectrum Calculation	Max. Difference below Target (%)	Max. Difference above Target[a] (%)	Max. Number of Consecutive Points below Target
Yermo (CSDRS)	North-south	three decades from 0.1 to 100 Hz	607	-3.6	+23.8	1
	East-west			-5.3	+26.3	4
	Vertical			-4.9	+22.4	7
Capitola (CSDRS)	North-south	three decades from 0.1 to 100 Hz	607	-8.6	+23.8	10[c]
	East-west			-4.3	+23.7	5
	Vertical			-7.0	+24.1	2
Chi-Chi (CSDRS)	North-south	three decades from 0.1 to 100 Hz	607	-7.4	+16.1	3
	East-west			-4.6	+30.0[b]	4
	Vertical			-4.6	+27.0	2
Izmit (CSDRS)	North-south	three decades from 0.1 to 100 Hz	607	-7.1	+21.0	3
	East-west			-5.2	+17.5	11[d]
	Vertical			-9.3	+17.8	5
El Centro (CSDRS)	North-south	three decades from 0.1 to 100 Hz	607	-6.6	+16.3	7
	East-west			-5.8	+27.9	14[e]
	Vertical			-7.2	+17.2	3
Lucerne (CSDRS-HF)	North-south	0.1 - 1 Hz	110	-6.51	+13.11	6
		1 - 10 Hz	215			
		10 - 100 Hz	271			
	East-west	0.1 - 1 Hz	132	-6.63	+13.07	19[f]
		1 - 10 Hz	148			
		10 - 100 Hz	221			
	Vertical	0.1 - 1 Hz	85[g]	-2.26	+13.65	6
		1 - 10 Hz	229			
		10 - 100 Hz	362			

Notes:

(a) The high values are obtained in low frequency range of lower than 0.2 Hz

(b) Actually 29.96 at 0.164 Hz

(c) Found at 0.145 Hz, the maximum below target is 5.2%; beyond frequency 0.162 Hz, the maximum number of below target value is 1

(d) Found at 0.12 Hz, the maximum below target is 5.2%; beyond frequency 0.135 Hz, the maximum number of below target value is 4

(e) Found at 0.22 Hz, the maximum below target is 3.9%; beyond frequency 0.254 Hz, the maximum number of below target value is 1

(f) Found at 0.123 Hz, maximum below target is 6.8%; also found at 0.21 Hz with maximum below target 6.7%; beyond 0.23 Hz the maximum number below target is five

(g) There are less than 100 points in the 0.1 Hz to 1 Hz decade. However, for the CSDRS-HF, the frequency range of interest is 10 Hz to 100 Hz. There are 362 analyzed frequencies in that decade

Table 3.7.1-6: Soil Shear Modulus Degradation and Strain-Dependent Soil Damping (0-120 ft)

1. Depth 0-20 ft			2. Depth 20-50 ft			3. Depth 50-120 ft		
Strain	**G/G_{max}**	**Damping (%)**	**Strain**	**G/G_{max}**	**Damping (%)**	**Strain**	**G/G_{max}**	**Damping (%)**
0.0001	1	1.5	0.0001	1	1.2	0.0001	1	1
0.0003	1	1.6	0.0003	1	1.2	0.0003	1	1
0.001	0.985	1.9	0.001	0.995	1.3	0.001	1	1.1
0.003	0.915	2.8	0.003	0.95	2	0.003	0.97	1.7
0.01	0.75	5.1	0.01	0.825	3.6	0.01	0.875	2.8
0.03	0.52	9	0.03	0.62	6.8	0.03	0.695	5.3
0.1	0.275	15.4	0.1	0.36	12.6	0.1	0.43	10.3
0.3	0.125	21.5	0.3	0.175	18.7	0.3	0.23	16.3
1	0.045	28	1	0.067	25	1	0.09	22.8

Table 3.7.1-7: Soil Shear Modulus Degradation and Strain-Dependent Soil Damping (120 ft-1000 ft)

4. Depth 120-250 ft			5. Depth 250-500 ft			6. Depth 500-1000 ft		
Strain	G/G_{max}	Damping (%)	Strain	G/G_{max}	Damping (%)	Strain	G/G_{max}	Damping (%)
0.0001	1	0.8	0.0001	1	0.8	0.0001	1	0.6
0.0003	1	0.8	0.0003	1	0.8	0.0003	1	0.6
0.001	1	0.9	0.001	1	0.8	0.001	1	0.6
0.003	0.975	1.3	0.003	0.985	1	0.003	0.99	0.8
0.01	0.905	2.2	0.01	0.93	1.8	0.01	0.95	1.3
0.03	0.755	4.3	0.03	0.805	3.4	0.03	0.86	2.4
0.1	0.495	8.8	0.1	0.56	7.3	0.1	0.65	5.5
0.3	0.28	14.3	0.3	0.335	12.5	0.3	0.41	10.2
1	0.115	21	1	0.15	19.2	1	0.2	16.7

Table 3.7.1-8: Strain-Dependent Soil Shear Moduli and Soil Damping Ratios for Gravel and Rock

7. Gravel (130+ ft)			8. Rock Average			
Strain	**G/G_{max}**	**Damping (%)**	**Strain**	**G/G_{max}**	**Strain**	**Damping (%)**
0.0001	1	3	0.0001	1	0.0001	0.4
0.0003	1	3	0.0003	1	0.001	0.8
0.001	1	3.3	0.001	0.9875	0.01	1.5
0.003	0.985	4	0.003	0.9525	0.1	3
0.01	0.82	6.5	0.01	0.9	1	4.6
0.03	0.57	10.1	0.03	0.81	-	-
0.1	0.32	16	0.1	0.725	-	-
0.3	0.14	22.5	1	0.55	-	-
1	0.05	27.5	-	-	-	-

Table 3.7.1-9: Soft Soil [Type 11] Parameters

Layer No.	Thickness (ft)	Depth (ft)	Shear Wave Velocity Vs (ft/s)	Unit Weight (pcf)	Poisson's Ratio
1	2	-2	703.3	120	0.35
2	3	-5	703.3	120	0.35
3	15	-20	703.3	120	0.35
4	20	-40	981.8	120	0.35
5	20	-60	1163.8	120	0.35
6	20	-80	1199	120	0.35
7	20	-100	1136	120	0.35
8	20	-120	1143	120	0.35
9	40	-160	1162	130	0.35
10	40	-200	1181	130	0.35
11	40	-240	1200	130	0.35
12	60	-300	8000	150	0.25
13	Halfspace	-300	8000	150	0.25

Table 3.7.1-10: Rock [Type 7] Parameters

Layer No.	Thickness (ft)	Depth (ft)	Shear Wave Velocity Vs (ft/s)	Unit Weight (pcf)	Poisson's Ratio
1	2	-2	5000	120	0.38
2	3	-5	5000	120	0.38
3	15	-20	5000	120	0.38
4	20	-40	5000	120	0.35
5	20	-60	5000	125	0.35
6	20	-80	5000	125	0.35
7	20	-100	5000	125	0.35
8	20	-120	5000	130	0.32
9	40	-160	5000	130	0.32
10	40	-200	5000	135	0.32
11	40	-240	5000	135	0.32
12	60	-300	5000	135	0.30
13	Halfspace	-300	5000	135	0.30

Table 3.7.1-11: Hard Rock [Type 9] Parameters

Layer No.	Thickness (ft)	Depth (ft)	Shear Wave Velocity Vs (ft/s)	Unit Weight (pcf)	Poisson's Ratio
1	2	-2	8000	150	0.25
2	3	-5	8000	150	0.25
3	15	-20	8000	150	0.25
4	20	-40	8000	150	0.25
5	20	-60	8000	150	0.25
6	20	-80	8000	150	0.25
7	20	-100	8000	150	0.25
8	20	-120	8000	150	0.25
9	40	-160	8000	150	0.25
10	40	-200	8000	150	0.25
11	40	-240	8000	150	0.25
12	60	-300	8000	150	0.25
13	Halfspace	-300	8000	150	0.25

Table 3.7.1-12: Average Strain-Compatible Properties for Certified Seismic Design Response Spectra for Rock [Type 7]

Layer No.	Depth(ft)	Layer Thickness (ft)	Damping Ratio	Unit Weight (pcf)	Vs (fps)	Poisson's Ratio	Vp (fps)
1	6.25	6.25	0.004	120	5000	0.38	11365
2	12.5	6.25	0.006	120	4993	0.38	11349
3	18.75	6.25	0.007	120	4980	0.38	11319
4	25	6.25	0.008	120	4971	0.36	10513
5	31.25	6.25	0.009	120	4956	0.35	10317
6	37.5	6.25	0.009	120	4939	0.35	10282
7	43.75	6.25	0.01	123	4928	0.35	10258
8	50	6.25	0.01	125	4918	0.35	10237
9	56.25	6.25	0.01	125	4907	0.35	10215
10	62.5	6.25	0.011	125	4898	0.35	10197
11	68.75	6.25	0.011	125	4890	0.35	10180
12	75	6.25	0.011	125	4883	0.35	10165
13	80	5	0.011	125	4876	0.35	10151
14	85	5	0.012	125	4870	0.35	10138
15	90	5	0.012	125	4864	0.35	10125
16	95	5	0.012	125	4858	0.35	10113
17	100	5	0.012	125	4853	0.35	10102
18	105	5	0.012	130	4852	0.32	9431
19	110	5	0.012	130	4847	0.32	9422
20	115	5	0.012	130	4843	0.32	9412
21	120	5	0.013	130	4838	0.32	9403
22	125	5	0.013	130	4834	0.32	9395
23	130	5	0.013	130	4829	0.32	9386
24	135	5	0.013	130	4825	0.32	9379
25	140	5	0.013	130	4821	0.32	9371
26	145	5	0.013	130	4818	0.32	9364
27	150	5	0.013	130	4814	0.32	9357
28	155	5	0.013	130	4811	0.32	9351
29	160	5	0.013	130	4808	0.32	9345
30	165	5	0.013	135	4809	0.32	9347
31	170	5	0.013	135	4806	0.32	9342
32	175	5	0.013	135	4803	0.32	9336
33	180	5	0.013	135	4801	0.32	9331
34	185	5	0.013	135	4798	0.32	9326
35	190	5	0.013	135	4796	0.32	9322
36	195	5	0.013	135	4794	0.32	9317
37	200	5	0.013	135	4791	0.32	9312
38	205	5	0.014	135	4789	0.32	9308
39	210	5	0.014	135	4787	0.32	9304
40	215	5	0.014	135	4785	0.32	9300
41	220	5	0.014	135	4783	0.32	9296
42	225	5	0.014	135	4781	0.32	9292
43	230	5	0.014	135	4779	0.32	9288
44	235	5	0.014	135	4777	0.32	9285
45	240	5	0.014	135	4775	0.32	9282
46	245	5	0.014	135	4774	0.30	8930
47	250	5	0.014	135	4772	0.30	8927
48	255	5	0.014	135	4770	0.30	8924

Table 3.7.1-12: Average Strain-Compatible Properties for Certified Seismic Design Response Spectra for Rock [Type 7] (Continued)

Layer No.	Depth(ft)	Layer Thickness (ft)	Damping Ratio	Unit Weight (pcf)	Vs (fps)	Poisson's Ratio	Vp (fps)
49	260	5	0.014	135	4768	0.30	8920
50	265	5	0.014	135	4766	0.30	8917
51	270	5	0.014	135	4765	0.30	8914
52	275	5	0.014	135	4763	0.30	8911
53	280	5	0.014	135	4762	0.30	8908
54	285	5	0.014	135	4760	0.30	8905
55	290	5	0.014	135	4759	0.30	8903
56	295	5	0.014	135	4757	0.30	8900
57	300	5	0.014	135	4756	0.30	8897
		Halfspace	0.014	135	5000	0.30	9354

Table 3.7.1-13: Average Strain-Compatible Properties for Certified Seismic Design Response Spectra for Soft Soil [Type 11]

Layer No.	Depth(ft)	Layer Thickness (ft)	Damping Ratio	Unit Weight (pcf)	Vs (fps)	Poisson's Ratio	Vp (fps)
1	6.25	6.25	0.045	120	625	0.48	3187
2	12.5	6.25	0.101	120	487	0.48	2481
3	18.75	6.25	0.149	120	371	0.48	1891
4	25	6.25	0.074	120	712	0.48	3632
5	31.25	6.25	0.08	120	739	0.48	3770
6	37.5	6.25	0.092	120	702	0.48	3581
7	43.75	6.25	0.084	120	805	0.48	4106
8	50	6.25	0.082	120	867	0.48	4421
9	56.25	6.25	0.063	120	933	0.48	4759
10	62.5	6.25	0.066	120	932	0.48	4754
11	68.75	6.25	0.068	120	943	0.48	4806
12	75	6.25	0.071	120	929	0.48	4739
13	80	5	0.074	120	919	0.48	4683
14	85	5	0.083	120	832	0.48	4240
15	90	5	0.085	120	824	0.48	4200
16	95	5	0.087	120	817	0.48	4163
17	100	5	0.088	120	810	0.48	4129
18	105	5	0.089	120	812	0.48	4141
19	110	5	0.09	120	807	0.48	4112
20	115	5	0.092	120	801	0.48	4082
21	120	5	0.093	120	795	0.48	4054
22	125	5	0.066	130	917	0.48	4674
23	130	5	0.067	130	911	0.48	4645
24	135	5	0.068	130	906	0.48	4617
25	140	5	0.07	130	899	0.48	4585
26	145	5	0.072	130	893	0.48	4552
27	150	5	0.073	130	886	0.48	4517
28	155	5	0.075	130	879	0.48	4483
29	160	5	0.076	130	873	0.48	4451
30	165	5	0.075	130	890	0.48	4539
31	170	5	0.077	130	885	0.48	4511
32	175	5	0.078	130	879	0.48	4484
33	180	5	0.079	130	875	0.48	4459
34	185	5	0.08	130	870	0.48	4436
35	190	5	0.081	130	865	0.48	4413
36	195	5	0.082	130	861	0.48	4389
37	200	5	0.083	130	856	0.48	4364
38	205	5	0.082	130	874	0.48	4458
39	210	5	0.083	130	870	0.48	4435
40	215	5	0.084	130	865	0.48	4413
41	220	5	0.085	130	861	0.48	4391
42	225	5	0.086	130	857	0.48	4370
43	230	5	0.087	130	854	0.48	4352
44	235	5	0.088	130	850	0.48	4333
45	240	5	0.089	130	846	0.48	4313
46	245	5	0.008	150	7945	0.25	13762
47	250	5	0.008	150	7936	0.25	13745
48	255	5	0.009	150	7925	0.25	13726

Table 3.7.1-13: Average Strain-Compatible Properties for Certified Seismic Design Response Spectra for Soft Soil [Type 11] (Continued)

Layer No.	Depth(ft)	Layer Thickness (ft)	Damping Ratio	Unit Weight (pcf)	Vs (fps)	Poisson's Ratio	Vp (fps)
49	260	5	0.009	150	7910	0.25	13700
50	265	5	0.009	150	7894	0.25	13672
51	270	5	0.01	150	7880	0.25	13649
52	275	5	0.01	150	7869	0.25	13629
53	280	5	0.01	150	7859	0.25	13611
54	285	5	0.01	150	7850	0.25	13597
55	290	5	0.01	150	7844	0.25	13586
56	295	5	0.01	150	7839	0.25	13578
57	300	5	0.01	150	7837	0.25	13573
		Halfspace	0.01	150	8000	0.25	13856

Table 3.7.1-14: Strain-Compatible Properties for Certified Seismic Design Response Spectra - High Frequency for Hard Rock [Type 9]

Layer No.	Depth (ft)	Layer Thickness (ft)	Damping Ratio	Unit Weight (pcf)	Vs (fps)	Poisson's Ratio	Vp (fps)
1	6.25	6.25	0.003	150	8000	0.250	13856.4
2	12.5	6.25	0.005	150	8000	0.250	13856.4
3	18.75	6.25	0.006	150	8000	0.250	13856.4
4	25	6.25	0.006	150	7992.2	0.250	13842.9
5	31.25	6.25	0.007	150	7982	0.250	13825.3
6	37.5	6.25	0.007	150	7974.1	0.250	13811.6
7	43.75	6.25	0.007	150	7967.8	0.250	13800.6
8	50	6.25	0.007	150	7962.7	0.250	13791.7
9	56.25	6.25	0.008	150	7958.5	0.250	13784.5
10	62.5	6.25	0.008	150	7955.2	0.250	13778.8
11	68.75	6.25	0.008	150	7952.6	0.250	13774.2
12	75	6.25	0.008	150	7949	0.250	13768
13	80	5	0.008	150	7946	0.250	13762.9
14	85	5	0.008	150	7944.1	0.250	13759.5
15	90	5	0.008	150	7940.8	0.250	13753.8
16	95	5	0.009	150	7936.5	0.250	13746.5
17	100	5	0.009	150	7932.9	0.250	13740.2
18	105	5	0.009	150	7929.9	0.250	13734.9
19	110	5	0.009	150	7927.4	0.250	13730.6
20	115	5	0.009	150	7925.4	0.250	13727.1
21	120	5	0.009	150	7923.1	0.250	13723.2
22	125	5	0.009	150	7920.1	0.250	13717.9
23	130	5	0.009	150	7917.3	0.250	13713.2
24	135	5	0.009	150	7914.8	0.250	13708.9
25	140	5	0.009	150	7912.6	0.250	13705
26	145	5	0.009	150	7910.6	0.250	13701.6
27	150	5	0.009	150	7908.8	0.250	13698.5
28	155	5	0.009	150	7906	0.250	13693.6
29	160	5	0.009	150	7903.4	0.250	13689.1
30	165	5	0.009	150	7901.1	0.250	13685.1
31	170	5	0.009	150	7899.1	0.250	13681.6
32	175	5	0.009	150	7897.3	0.250	13678.5
33	180	5	0.009	150	7895.8	0.250	13675.9
34	185	5	0.009	150	7894.5	0.250	13673.6
35	190	5	0.009	150	7893.4	0.250	13671.8
36	195	5	0.009	150	7892.6	0.250	13670.4
37	200	5	0.009	150	7892	0.250	13669.4
38	205	5	0.009	150	7891.7	0.250	13668.8
39	210	5	0.009	150	7891.6	0.250	13668.6
40	215	5	0.009	150	7891.6	0.250	13668.7
41	220	5	0.009	150	7891.9	0.250	13669.1
42	225	5	0.009	150	7890.9	0.250	13667.4
43	230	5	0.009	150	7890.1	0.250	13666.1
44	235	5	0.009	150	7889.6	0.250	13665.3
45	240	5	0.009	150	7889.5	0.250	13665
46	245	5	0.009	150	7889.7	0.250	13665.3
47	250	5	0.009	150	7890.1	0.250	13666

Table 3.7.1-14: Strain-Compatible Properties for Certified Seismic Design Response Spectra - High Frequency for Hard Rock [Type 9] (Continued)

Layer No.	Depth (ft)	Layer Thickness (ft)	Damping Ratio	Unit Weight (pcf)	Vs (fps)	Poisson's Ratio	Vp (fps)
48	255	5	0.009	150	7890.7	0.250	13667
49	260	5	0.009	150	7890.6	0.250	13666.9
50	265	5	0.009	150	7890.3	0.250	13666.4
51	270	5	0.009	150	7890.1	0.250	13666.1
52	275	5	0.009	150	7890	0.250	13665.9
53	280	5	0.009	150	7889.4	0.250	13664.9
54	285	5	0.009	150	7888.9	0.250	13664
55	290	5	0.009	150	7888.1	0.250	13662.6
56	295	5	0.009	150	7887.2	0.250	13661.1
57	300	5	0.009	150	7886.5	0.250	13659.8
		Halfspace	0.009	150	8000	0.250	13856.4

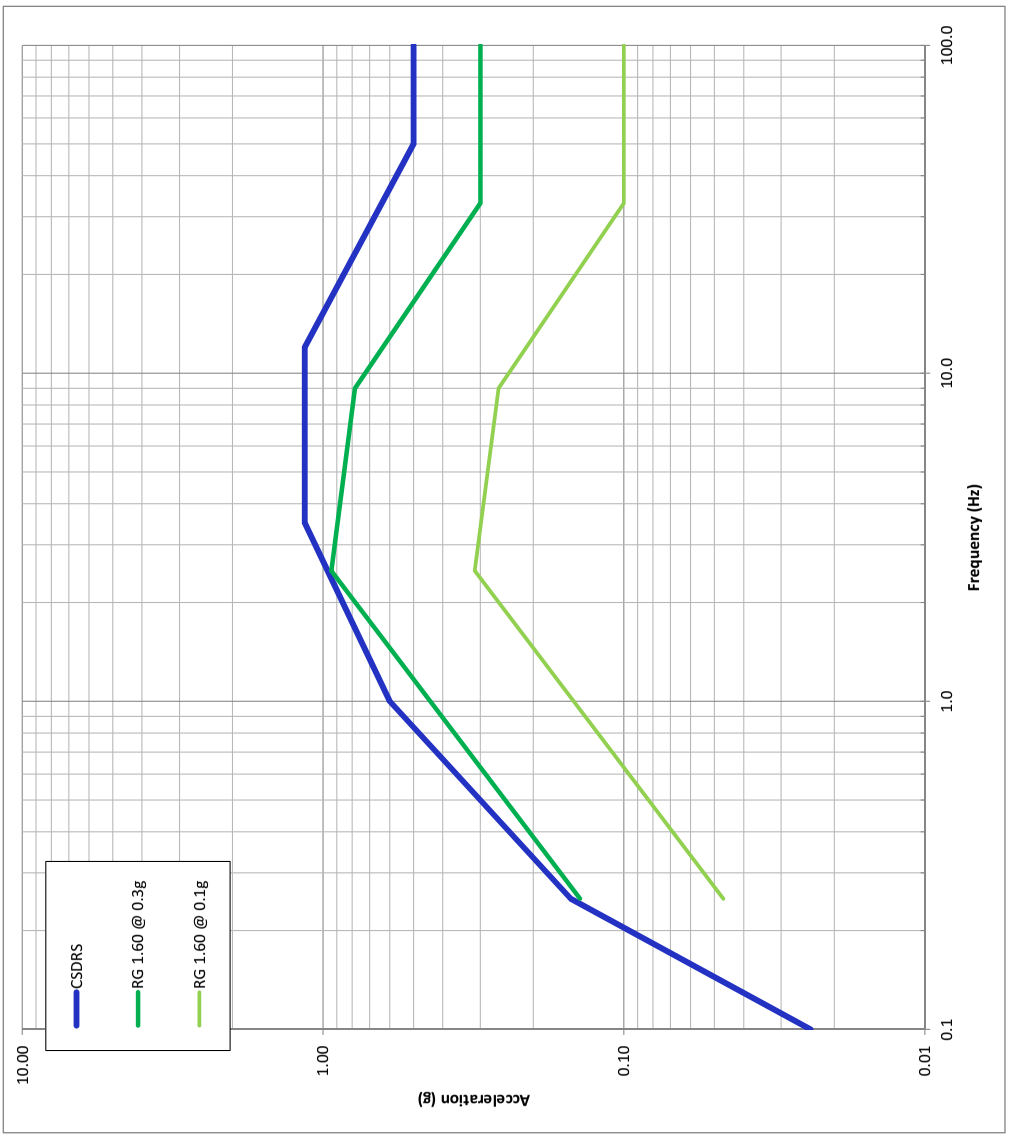


Figure 3.7.1-1: NuScale Horizontal Certified Seismic Design Response Spectra at 5 Percent Damping

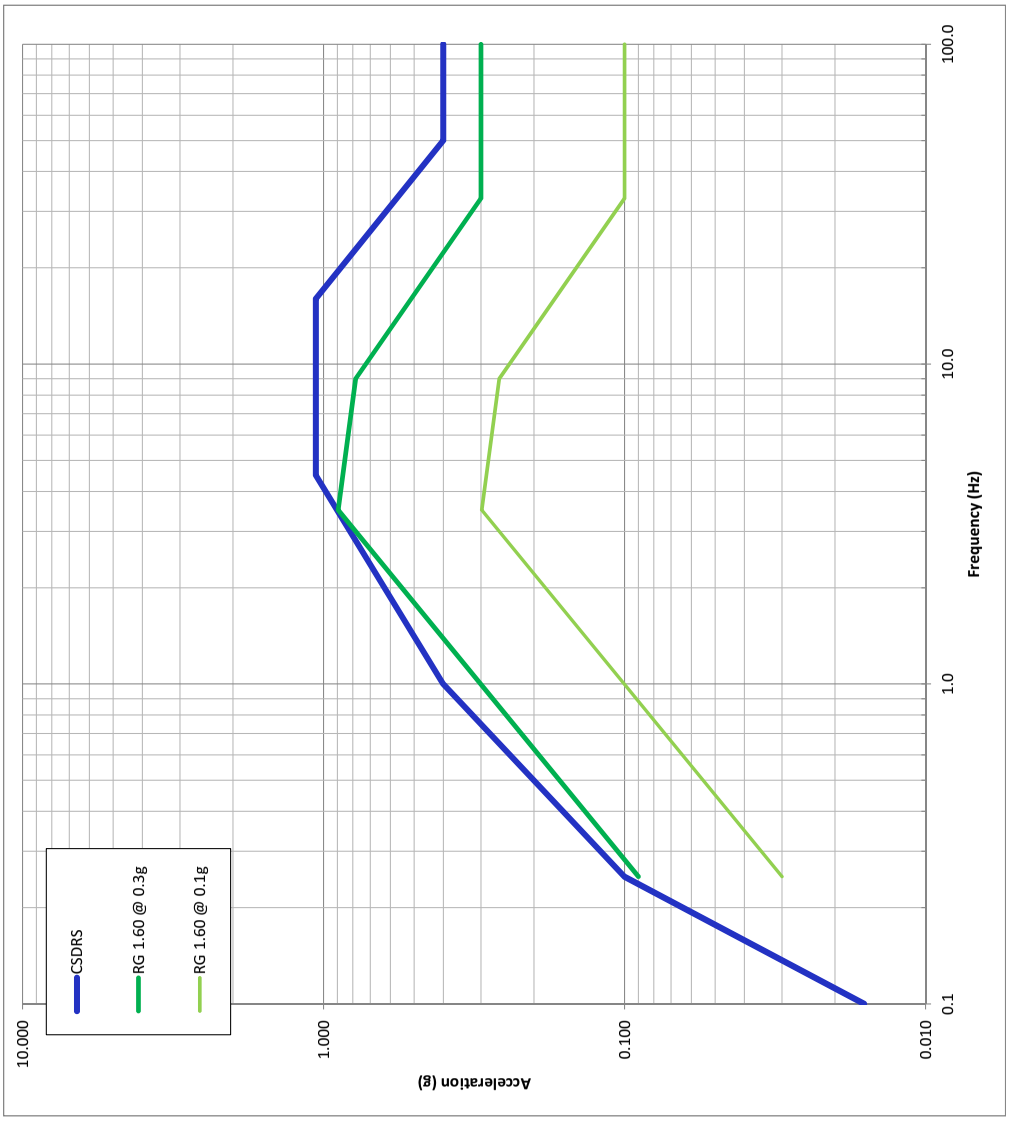
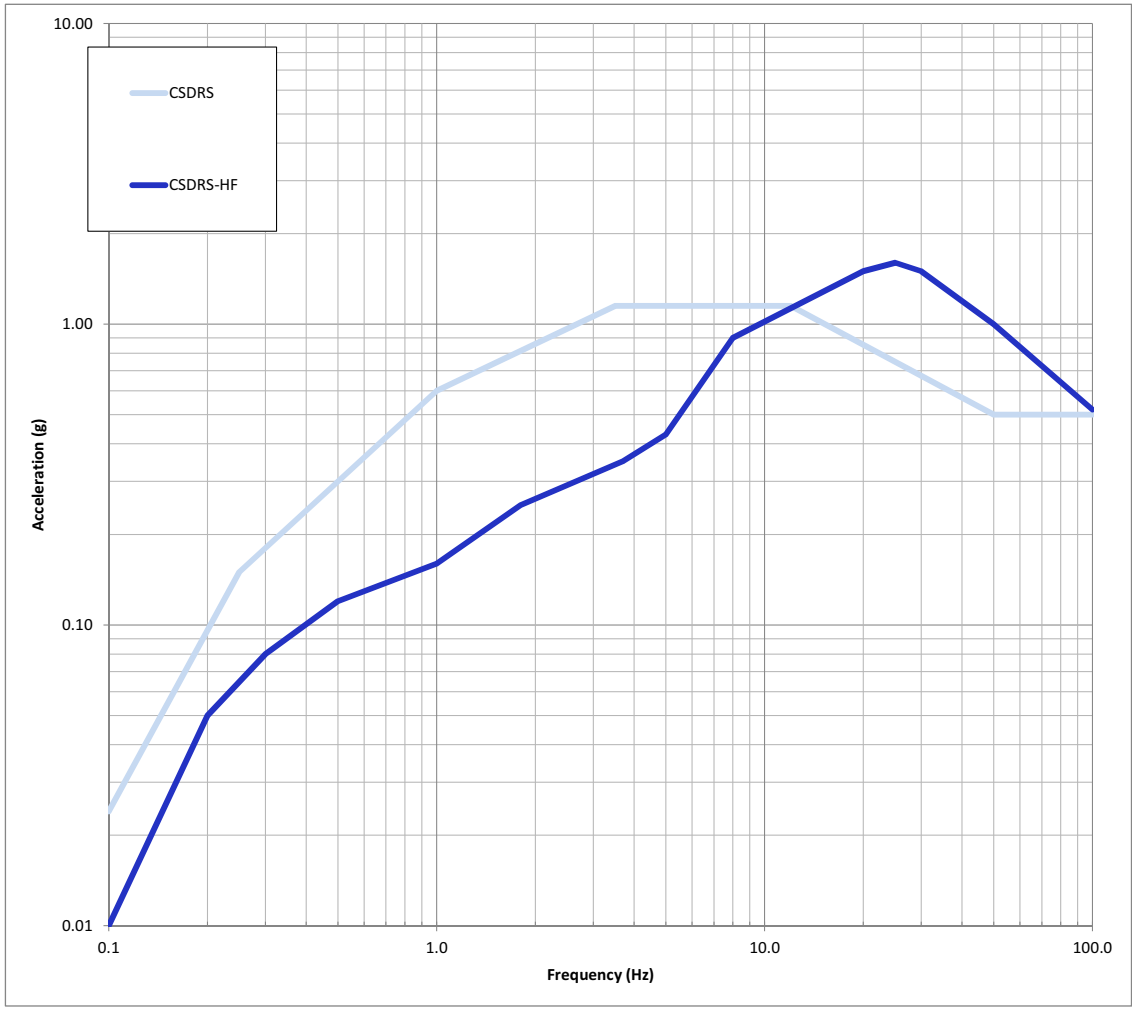


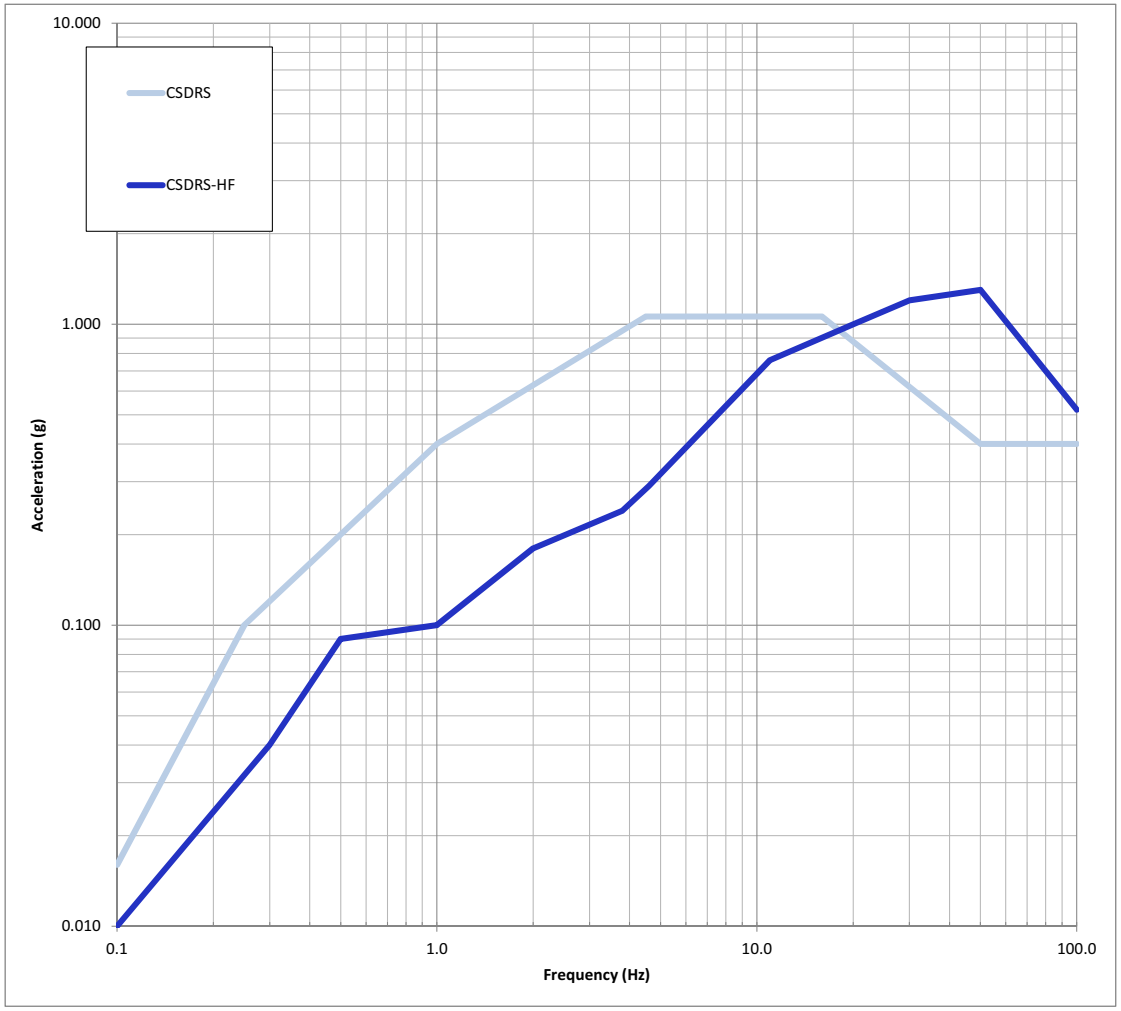
Figure 3.7.1-2: NuScale Vertical Certified Seismic Design Response Spectra at 5 Percent Damping

Figure 3.7.1-3: NuScale Horizontal Certified Seismic Design Response Spectra - High Frequency at 5 Percent Damping



Note: CSDRS-HF is evaluated for the RXB and the CRB only

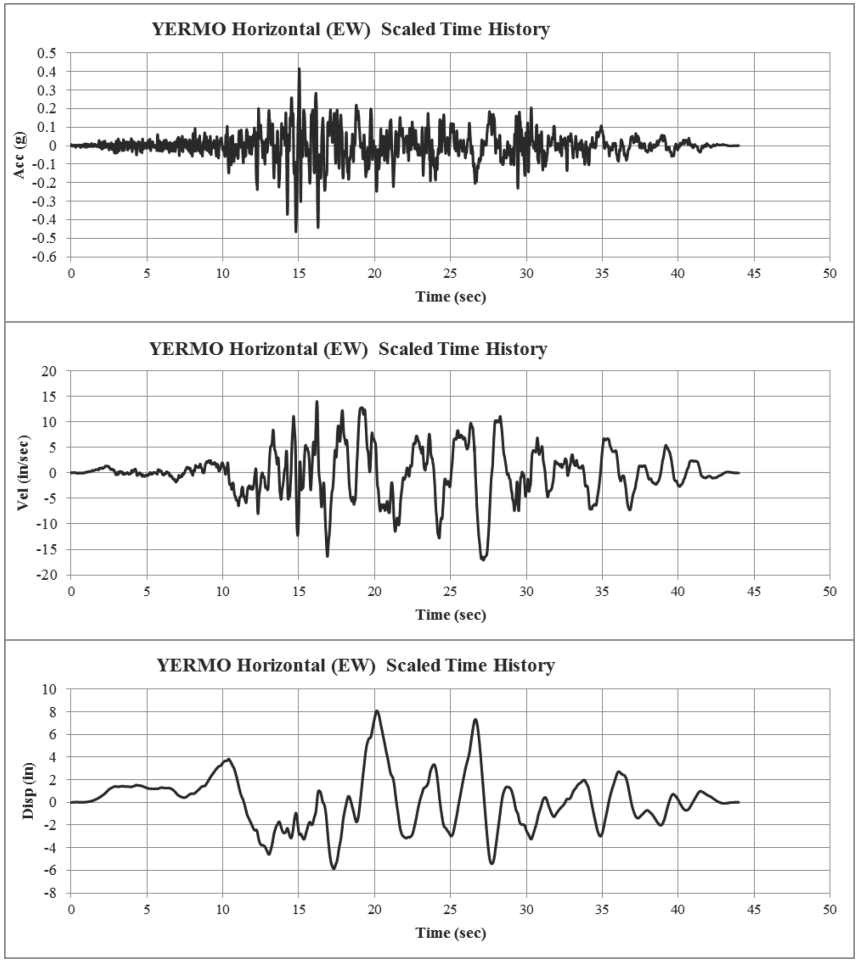
Figure 3.7.1-4: NuScale Vertical Certified Seismic Design Response Spectra - High Frequency at 5 Percent Damping



Note: CSDRS-HF is evaluated for the RXB and the CRB only

Figure 3.7.1-5a: Original Time Histories for Yermo East-West

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

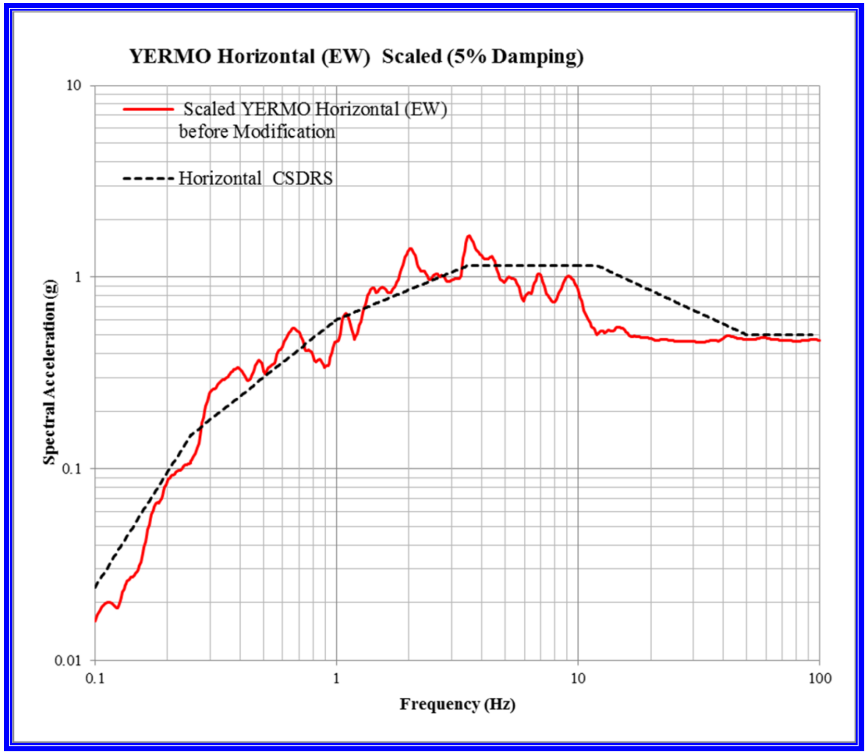
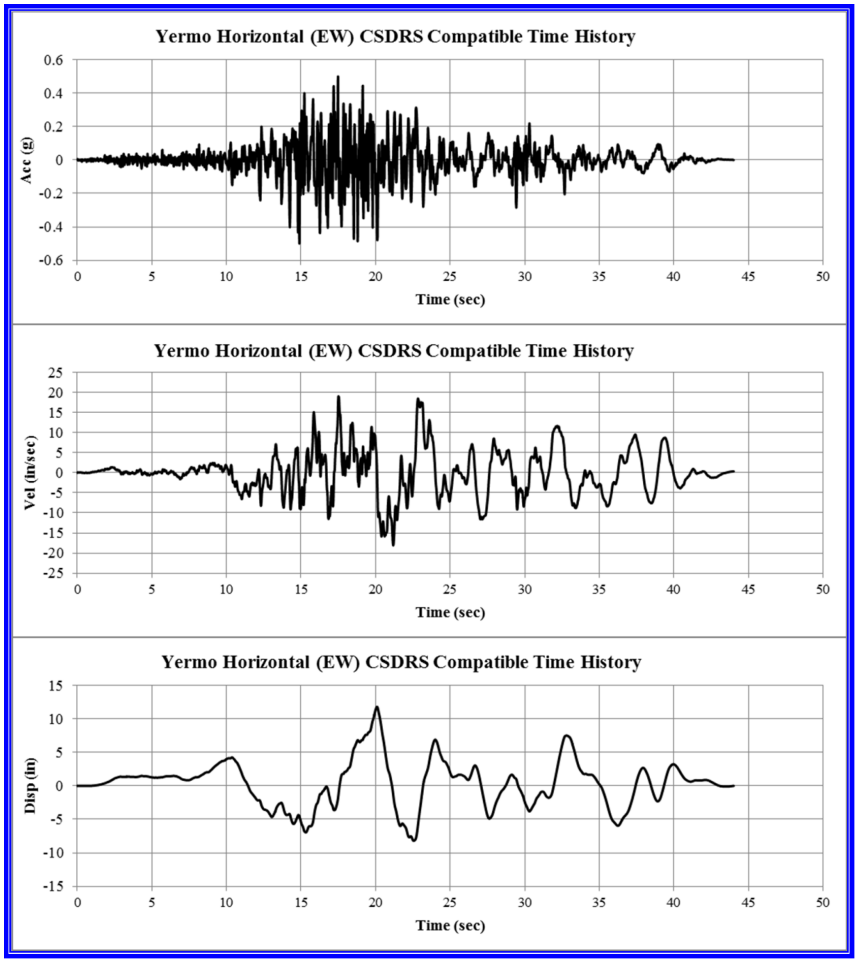
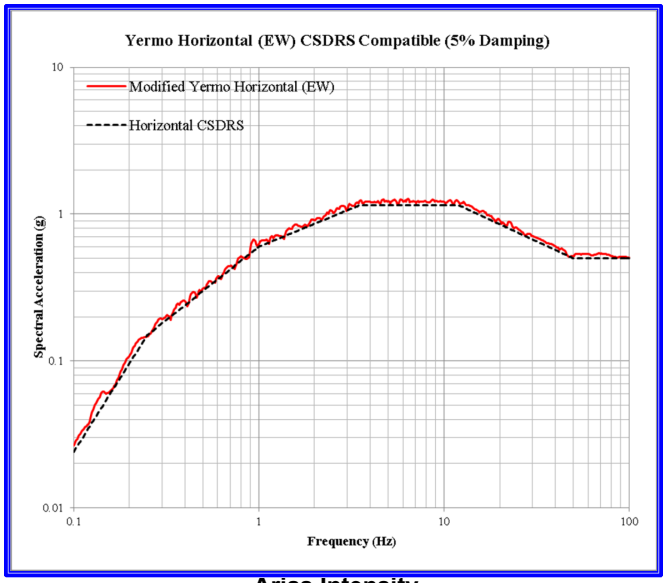


Figure 3.7.1-5b: Certified Seismic Design Response Spectra Compatible Time Histories for Yermo East-West

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

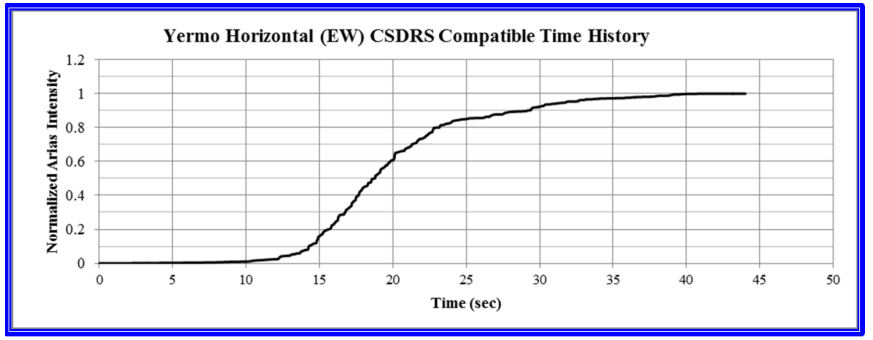
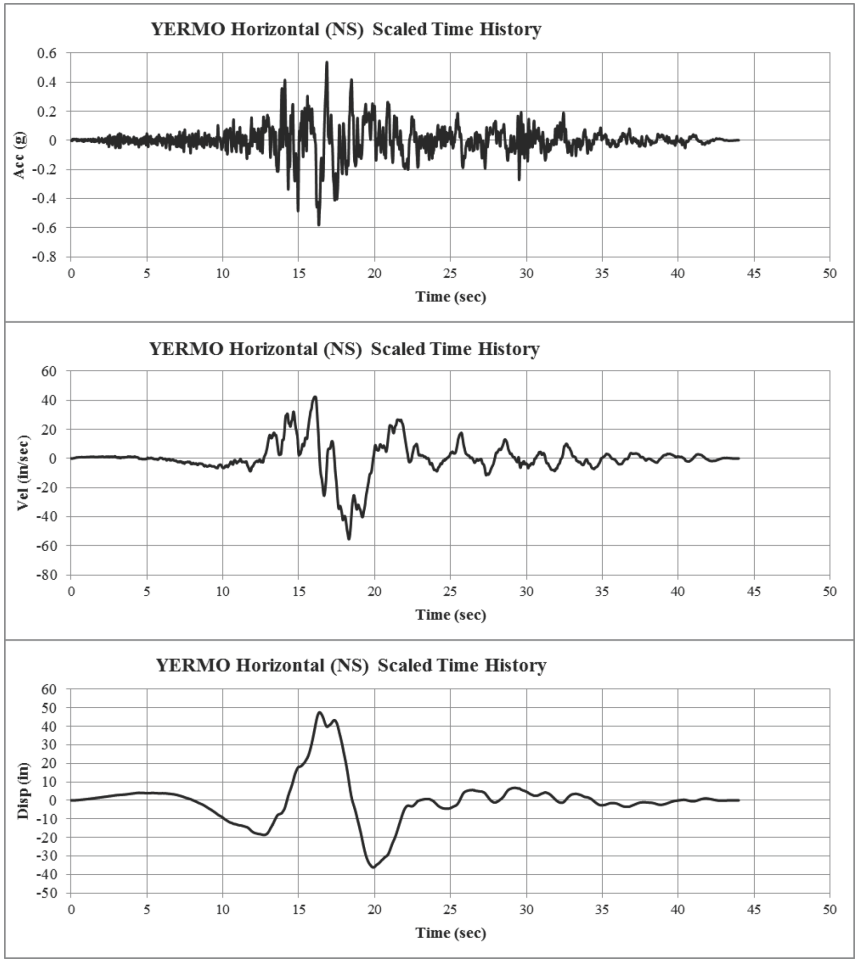


Figure 3.7.1-5c: Original Time Histories for Yermo North-South

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

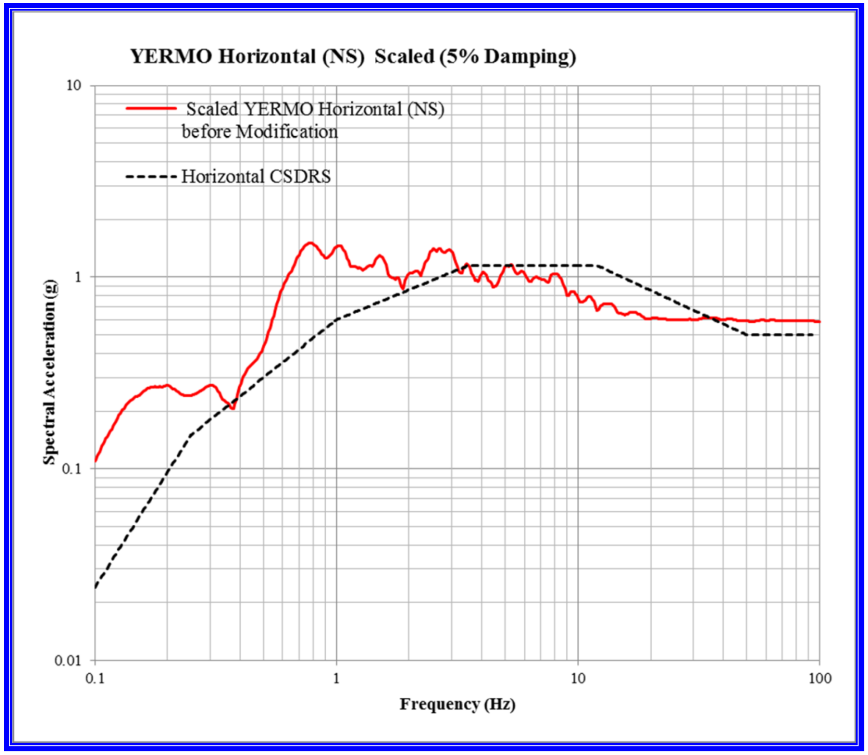
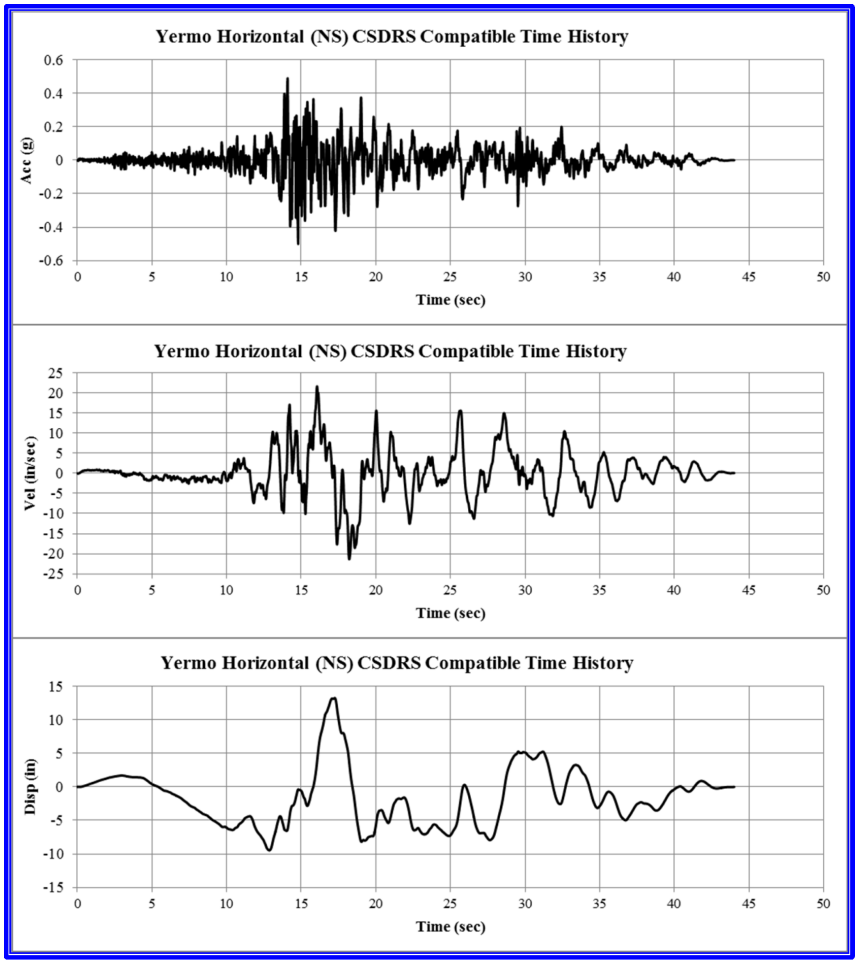
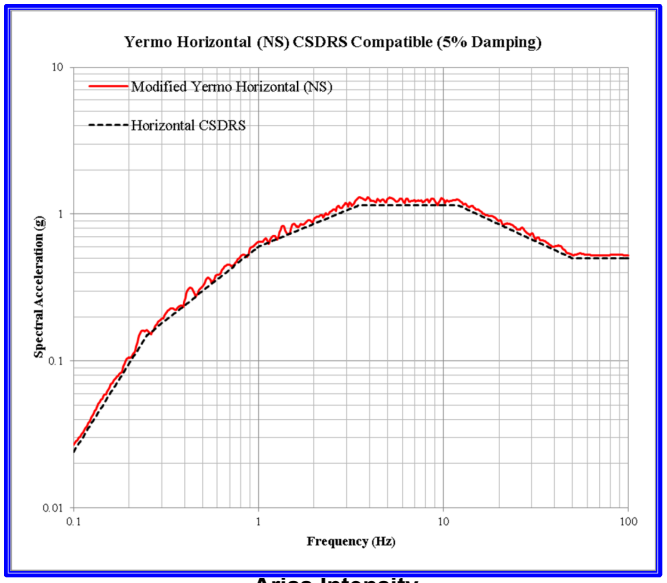


Figure 3.7.1-5d: Certified Seismic Design Response Spectra Compatible Time Histories for Yermo North-South

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

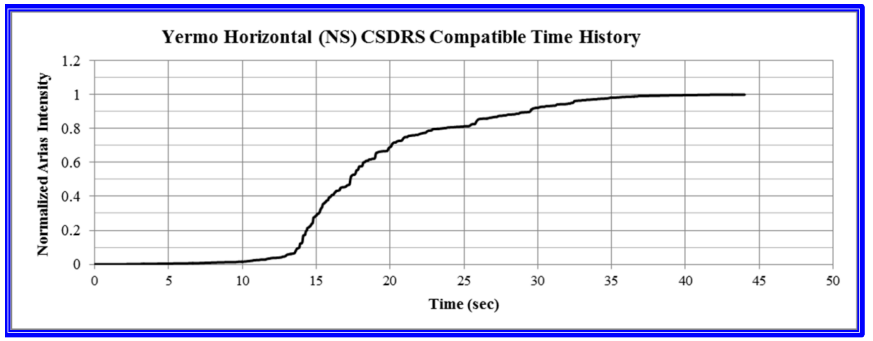
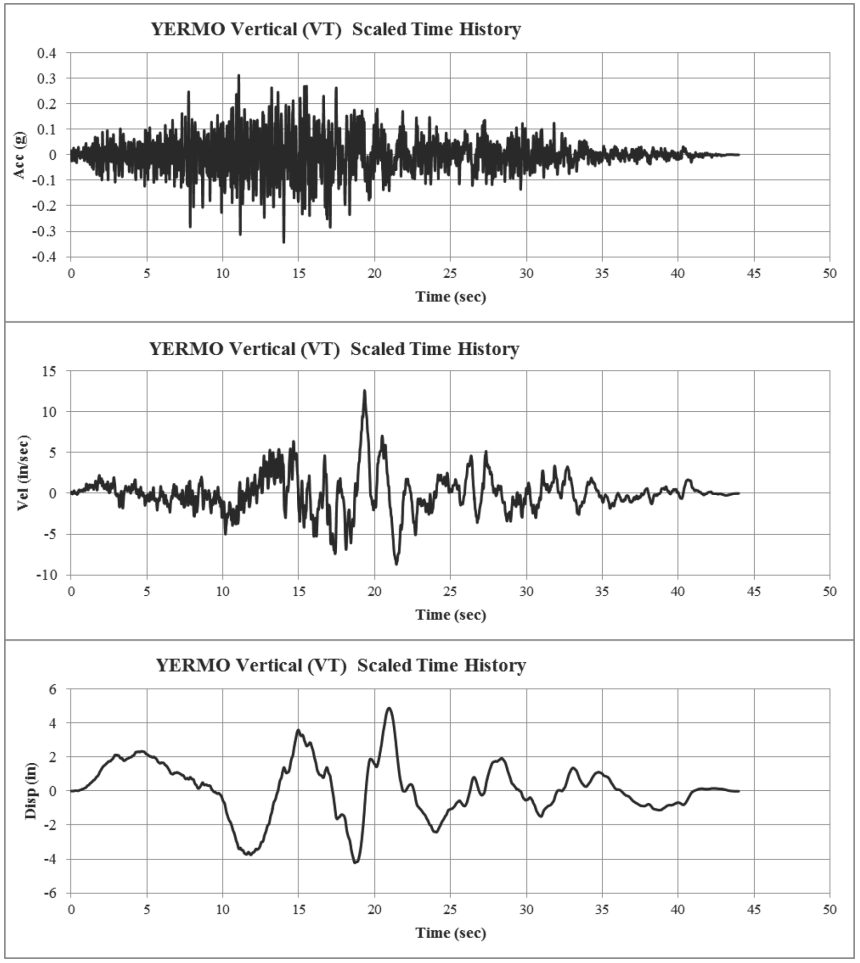


Figure 3.7.1-5e: Original Time Histories for Yermo Vertical

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

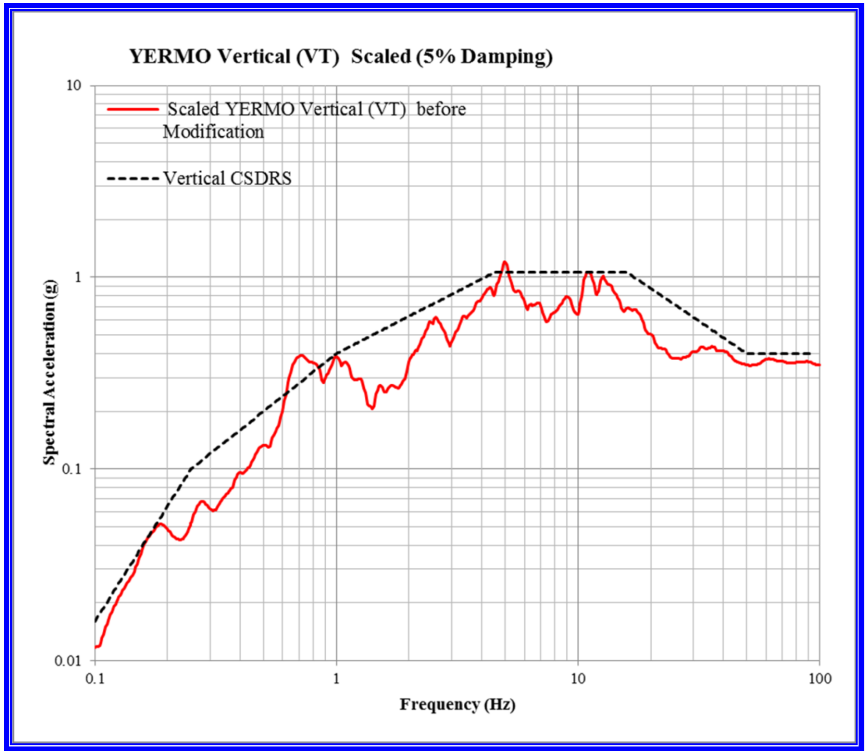
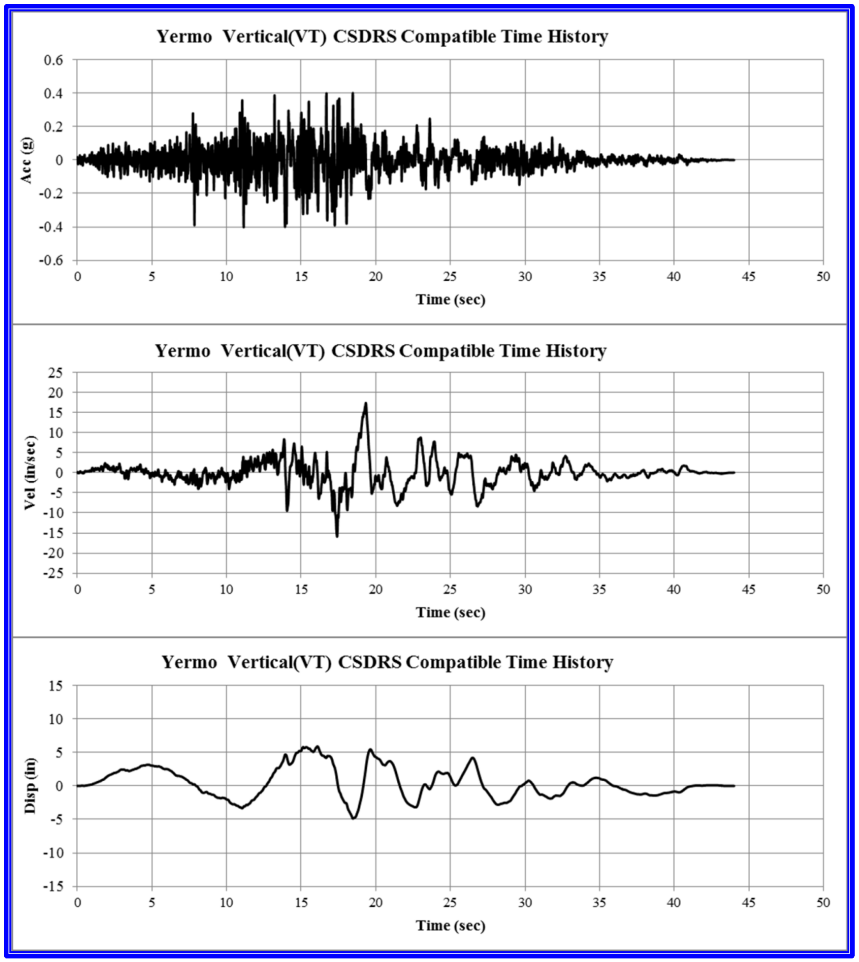
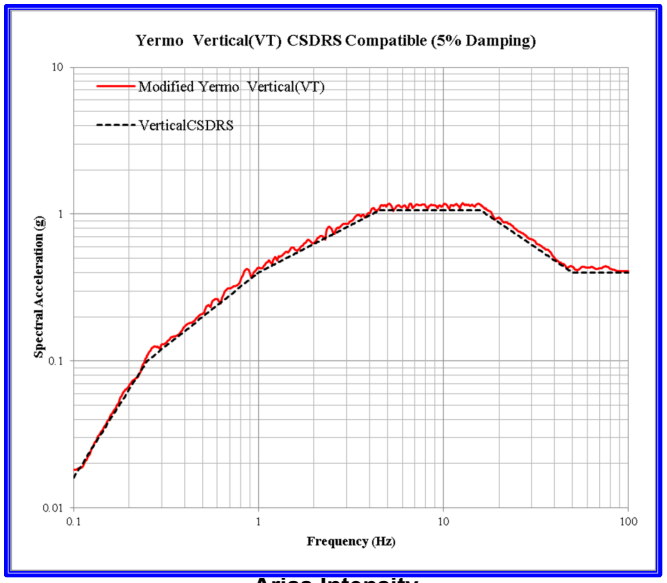


Figure 3.7.1-5f: Certified Seismic Design Response Spectra Compatible Time Histories for Yermo Vertical

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

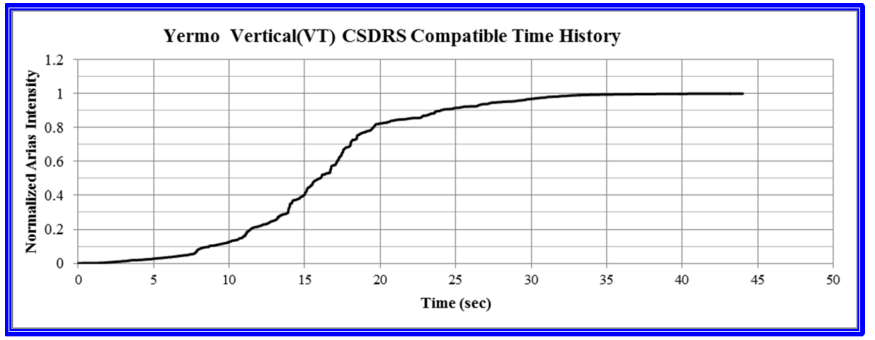
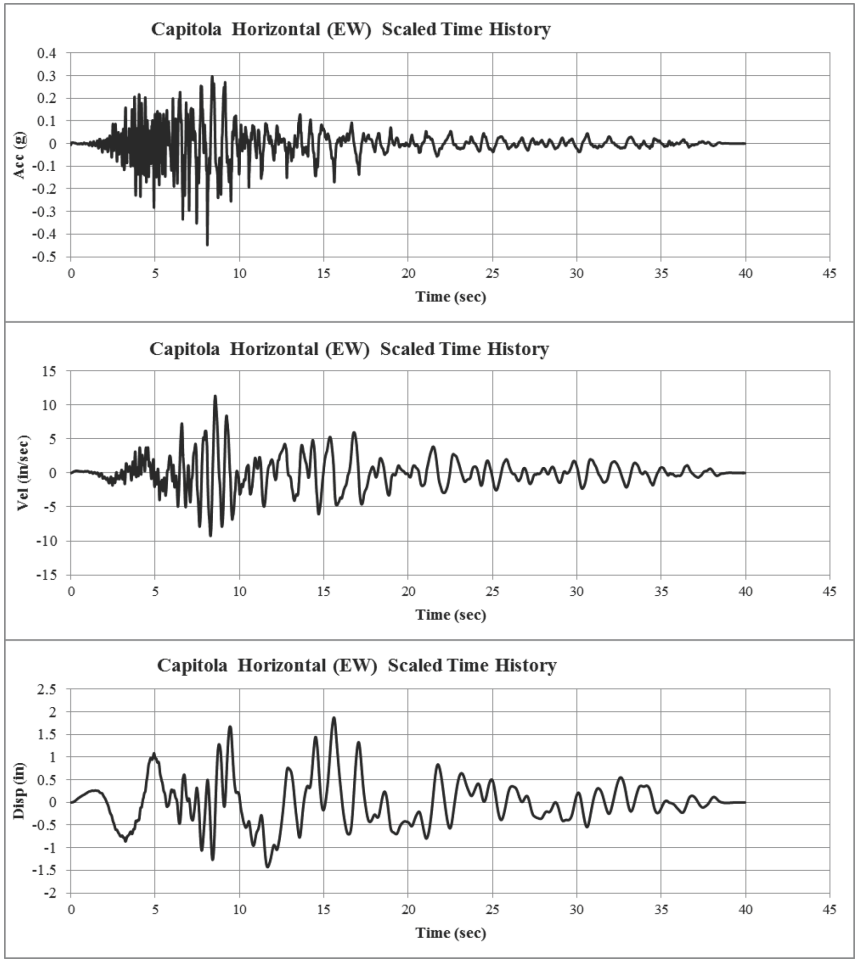


Figure 3.7.1-6a: Original Time Histories for Capitola East-West

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

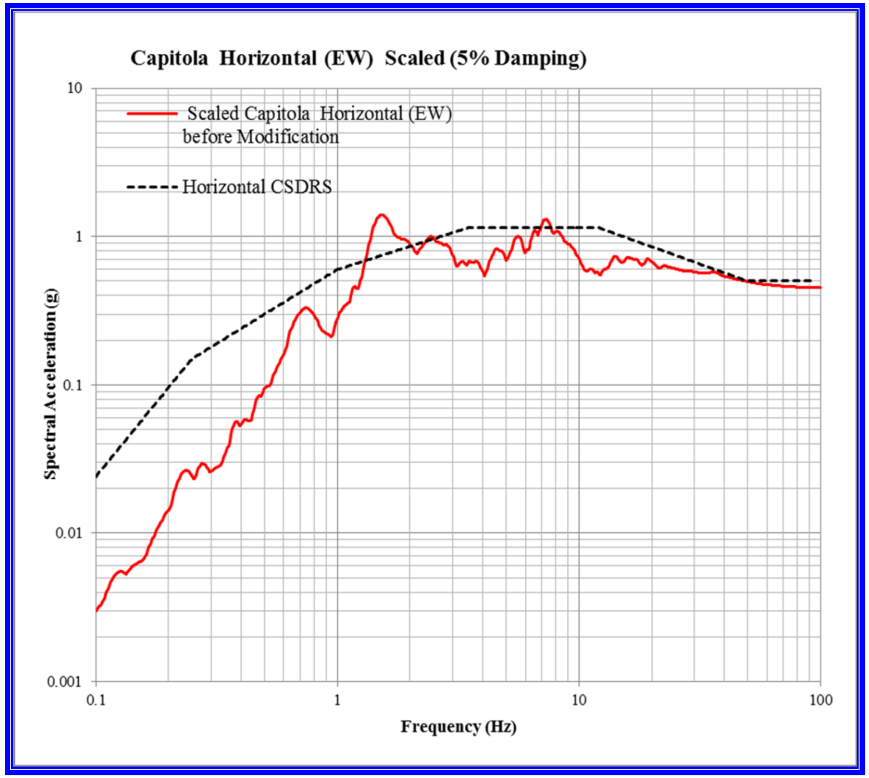
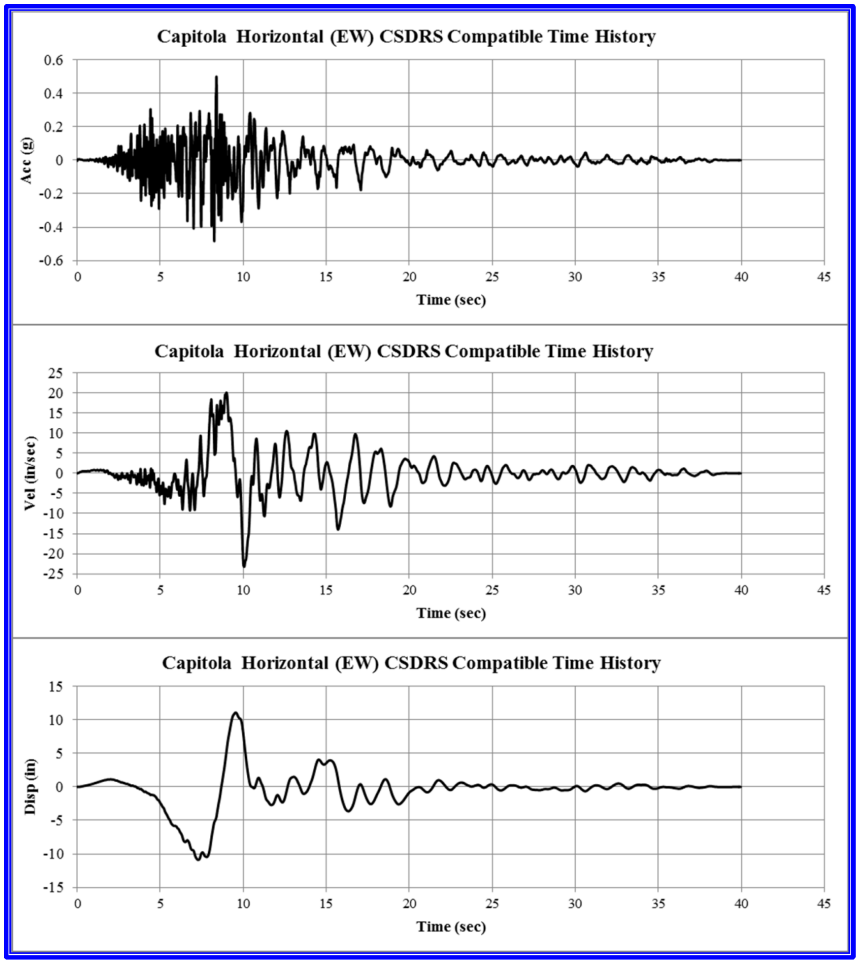
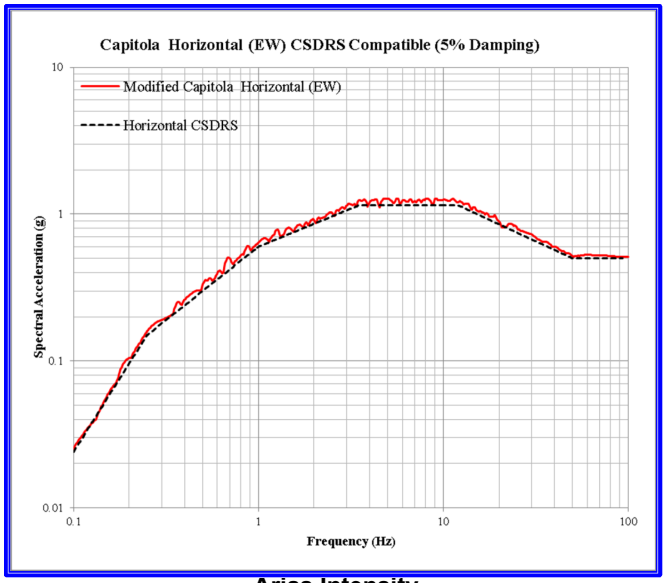


Figure 3.7.1-6b: Certified Seismic Design Response Spectra Compatible Time Histories for Capitola East-West

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

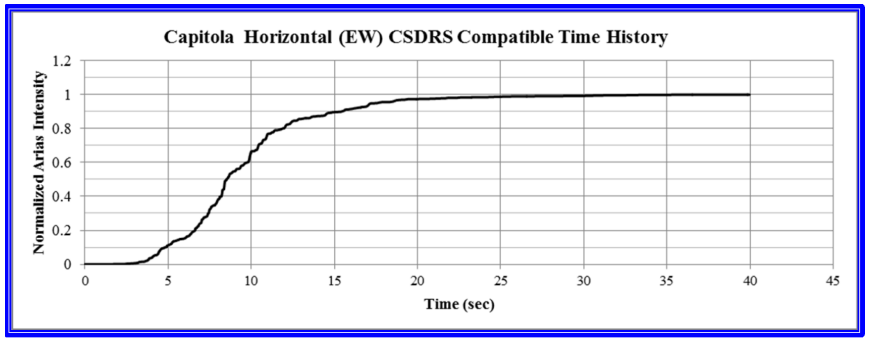
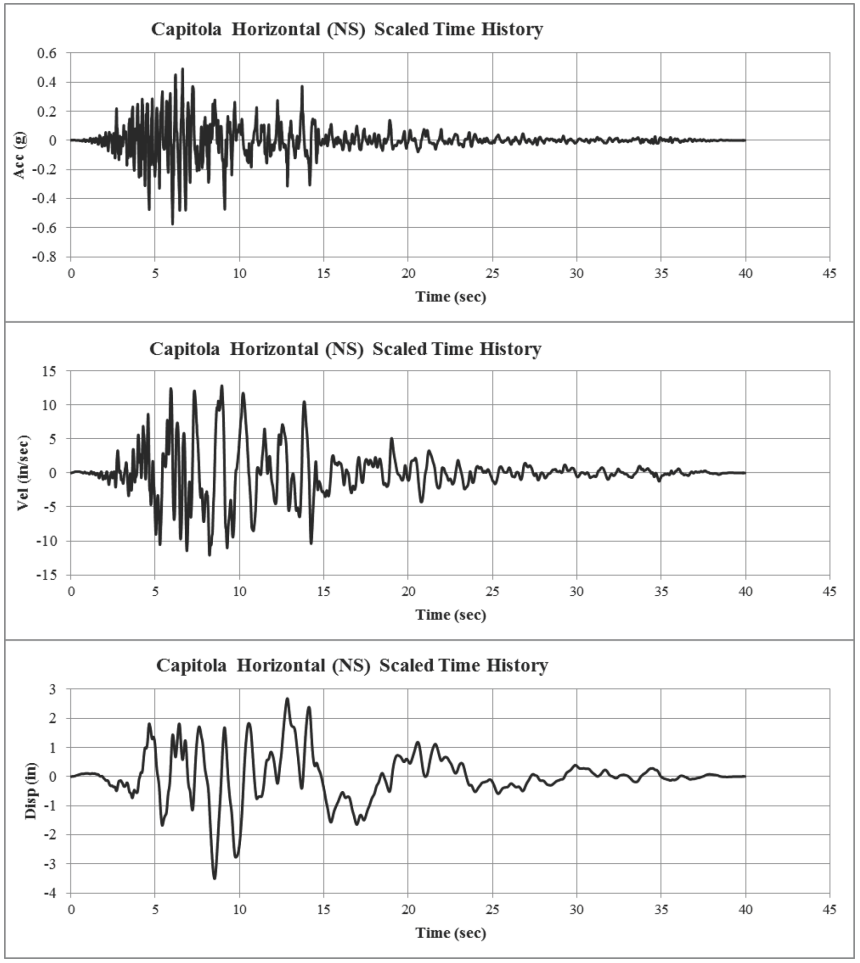


Figure 3.7.1-6c: Original Time Histories for Capitola North-South

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

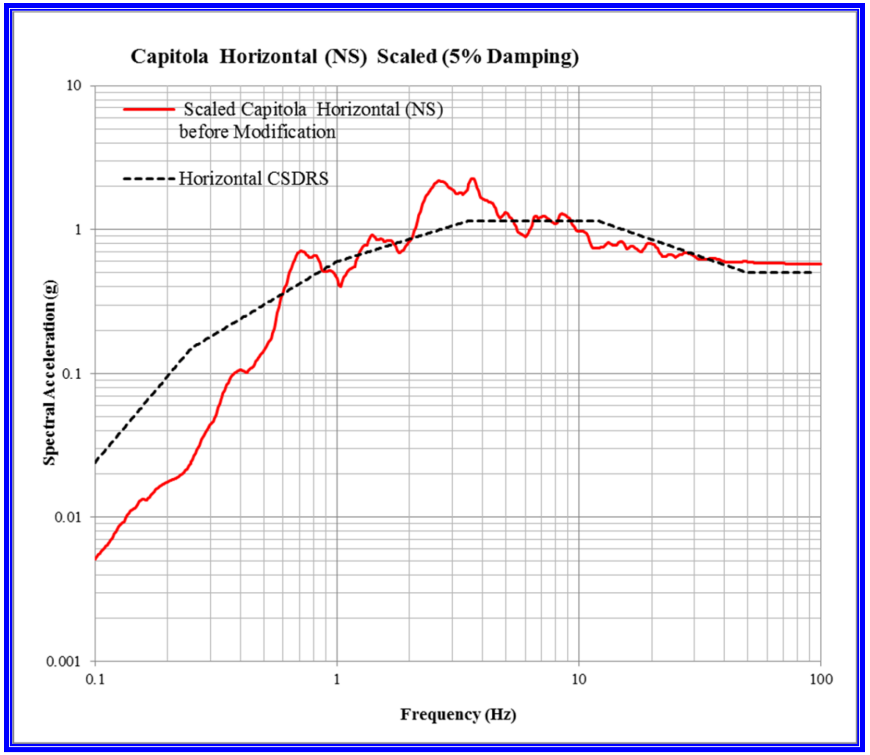
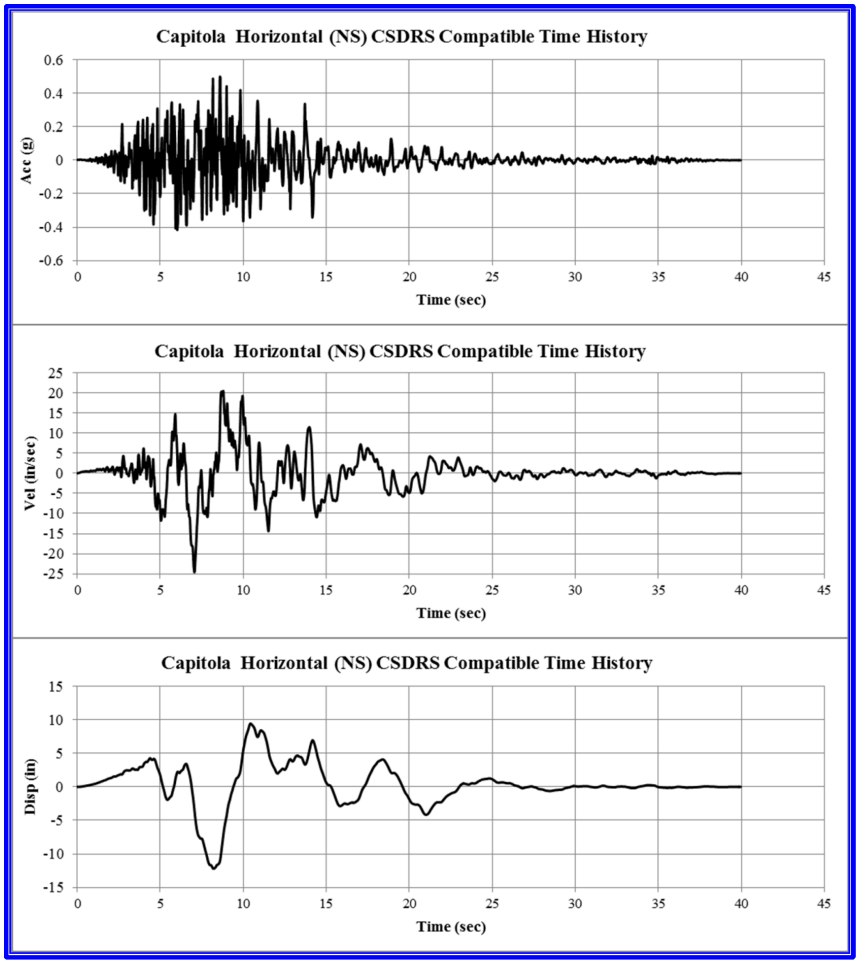
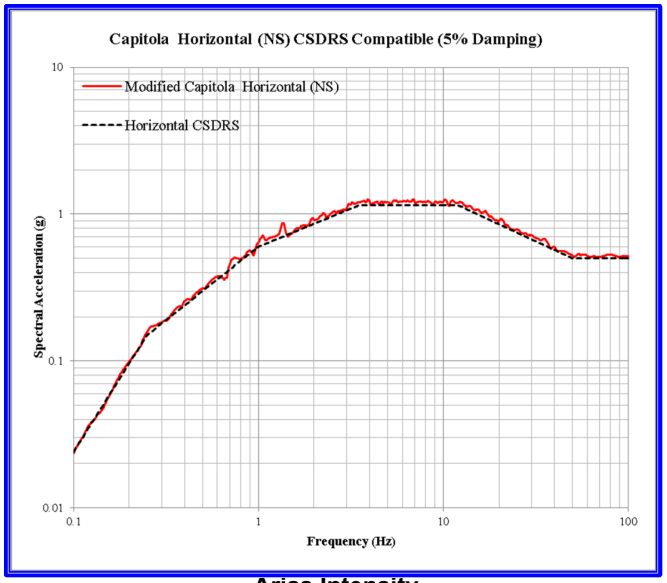


Figure 3.7.1-6d: Certified Seismic Design Response Spectra Compatible Time Histories for Capitola North-South

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

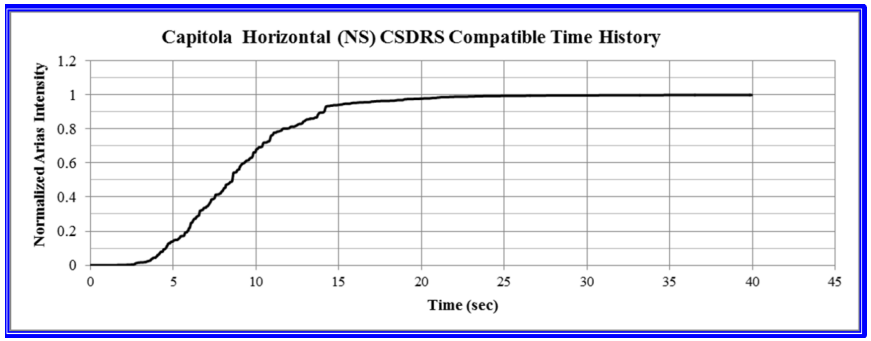
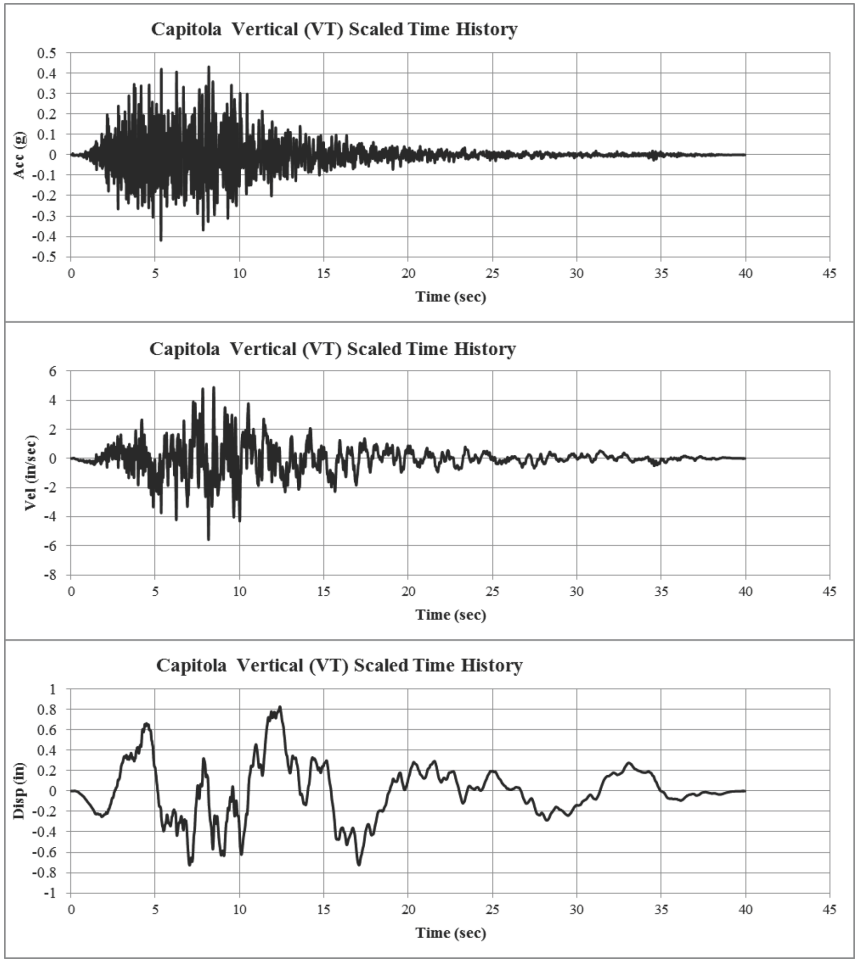


Figure 3.7.1-6e: Original Time Histories for Capitola Vertical

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

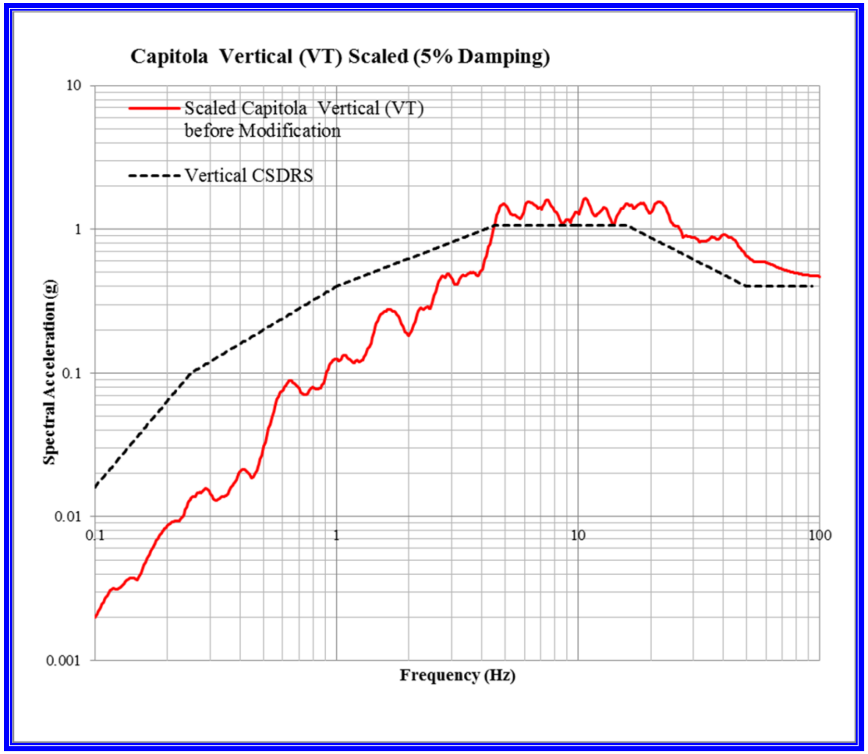
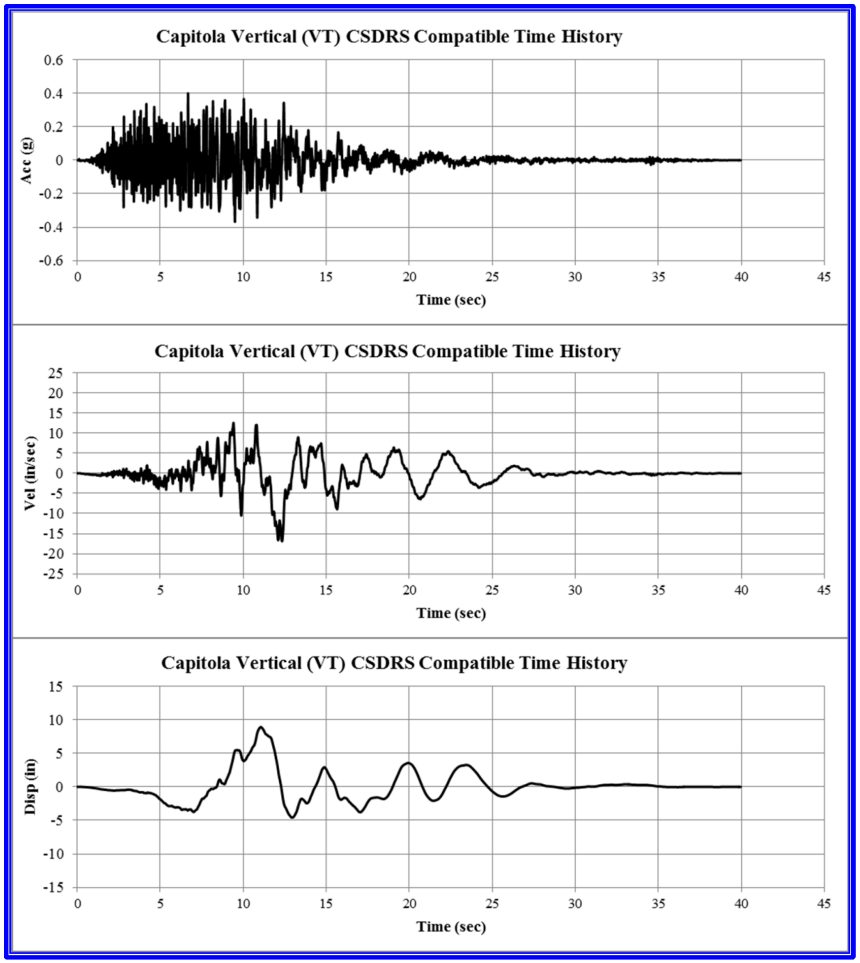
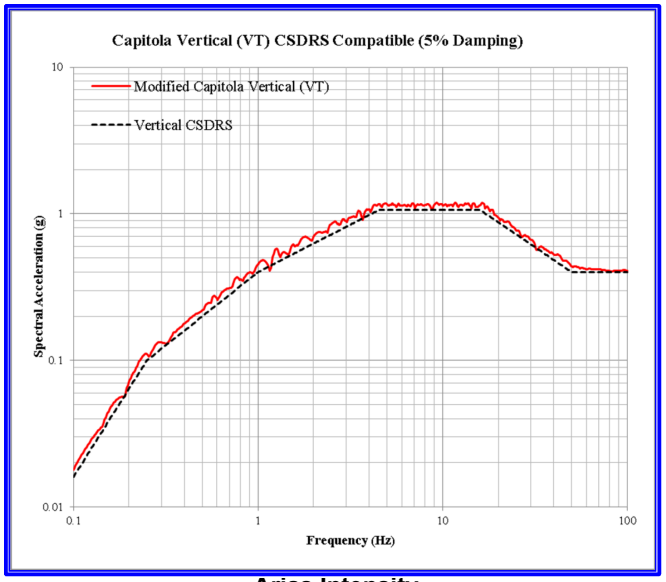


Figure 3.7.1-6f: Certified Seismic Design Response Spectra Compatible Time Histories for Capitola Vertical

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

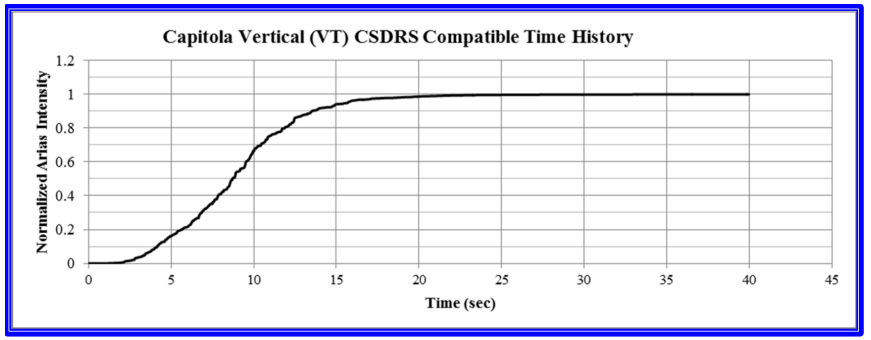
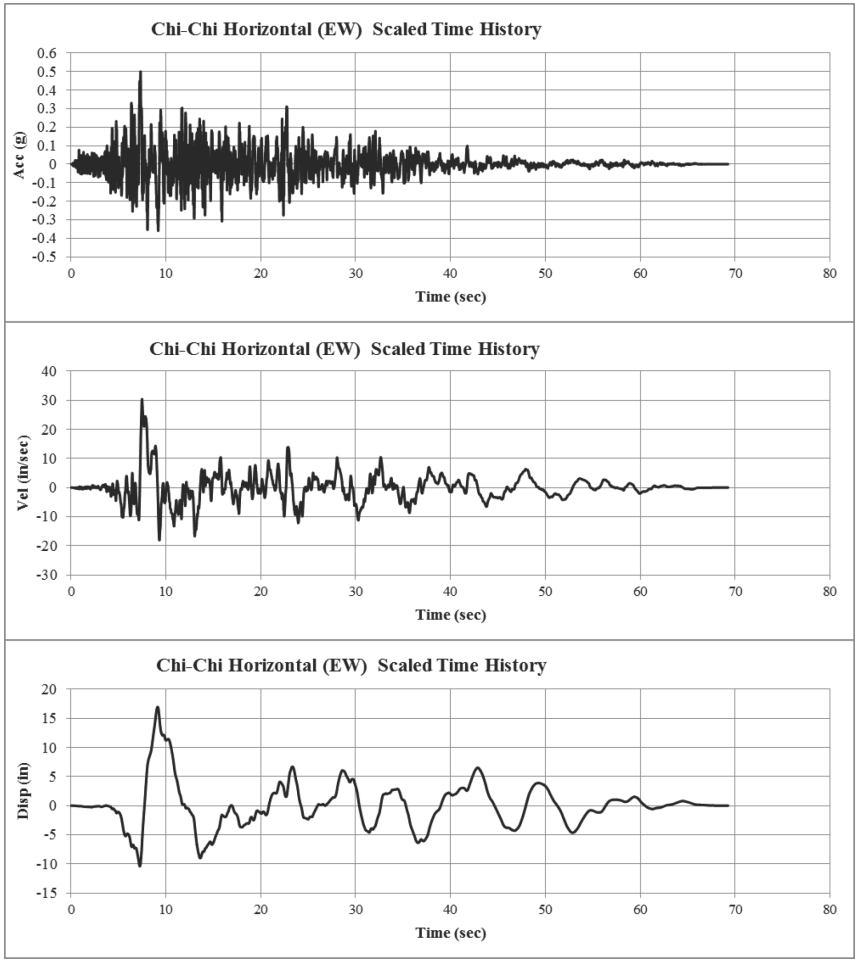


Figure 3.7.1-7a: Original Time Histories for Chi-Chi East-West

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

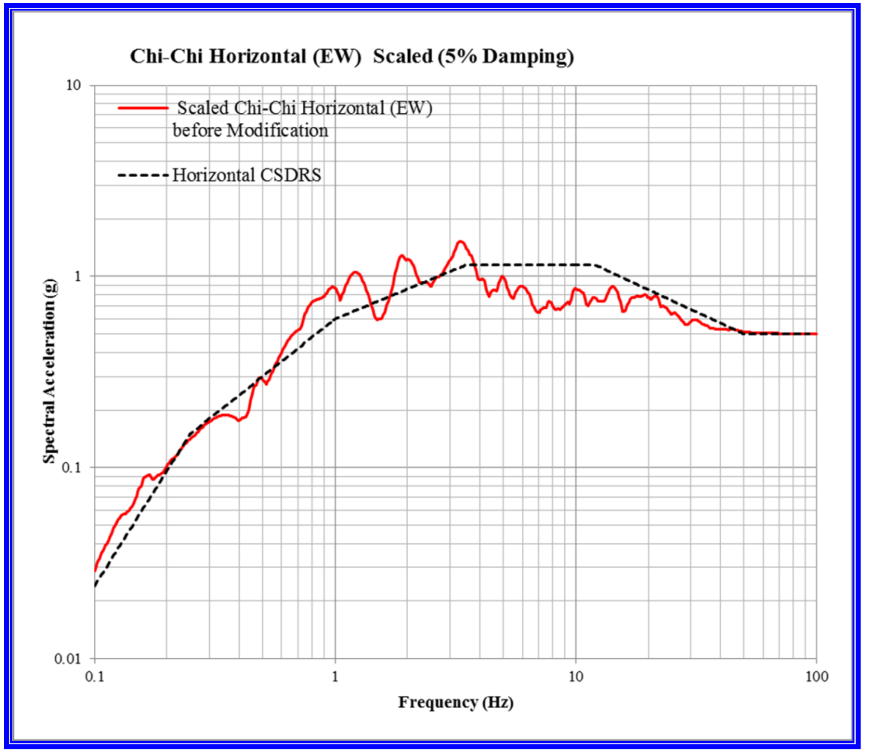
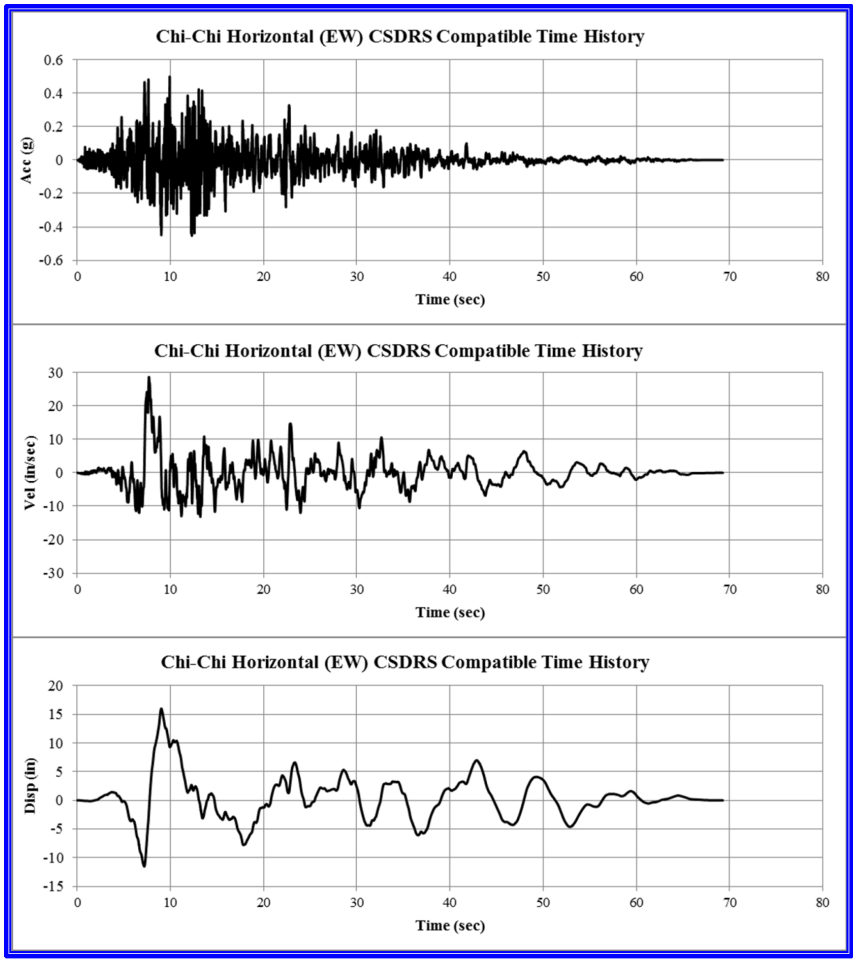
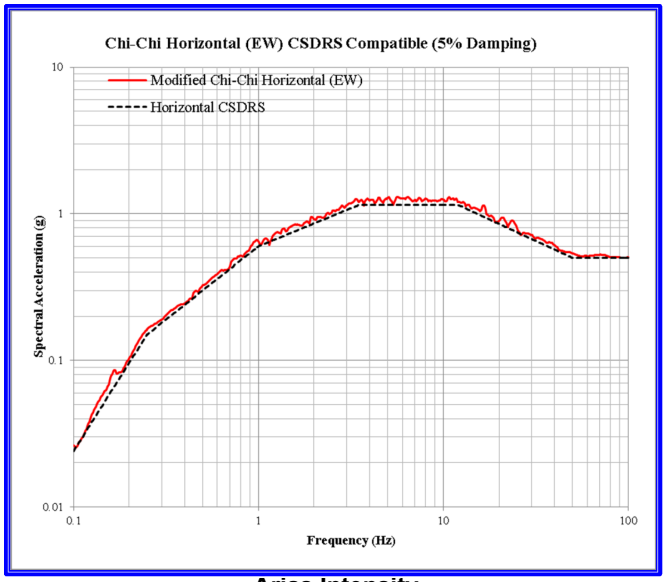


Figure 3.7.1-7b: Certified Seismic Design Response Spectra Compatible Time Histories for Chi-Chi East-West

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

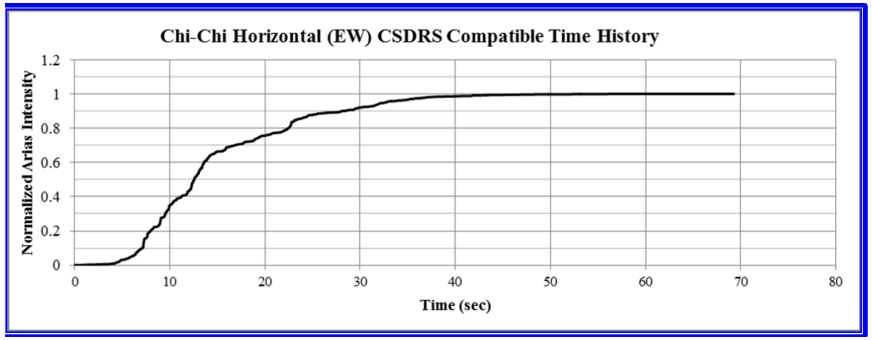
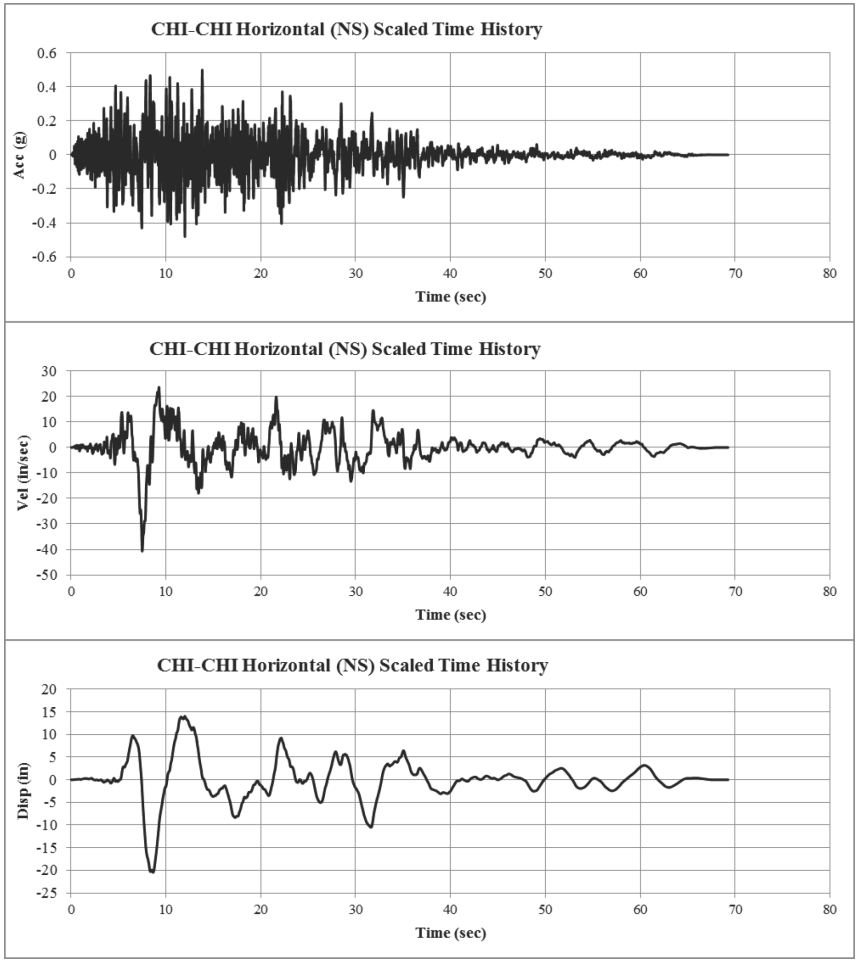


Figure 3.7.1-7c: Original Time Histories for Chi-Chi North-South

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

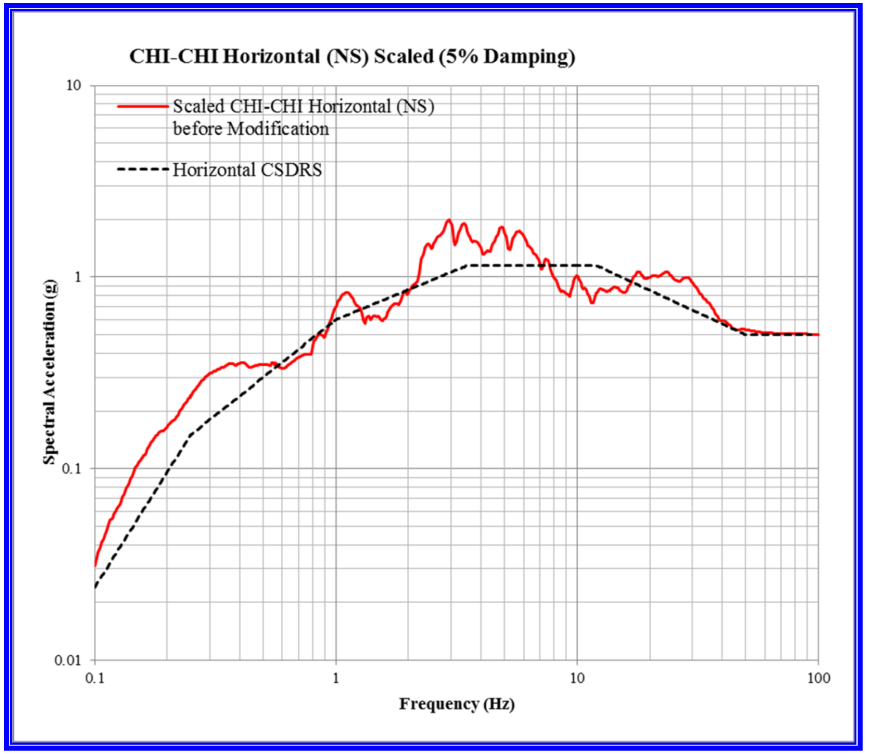
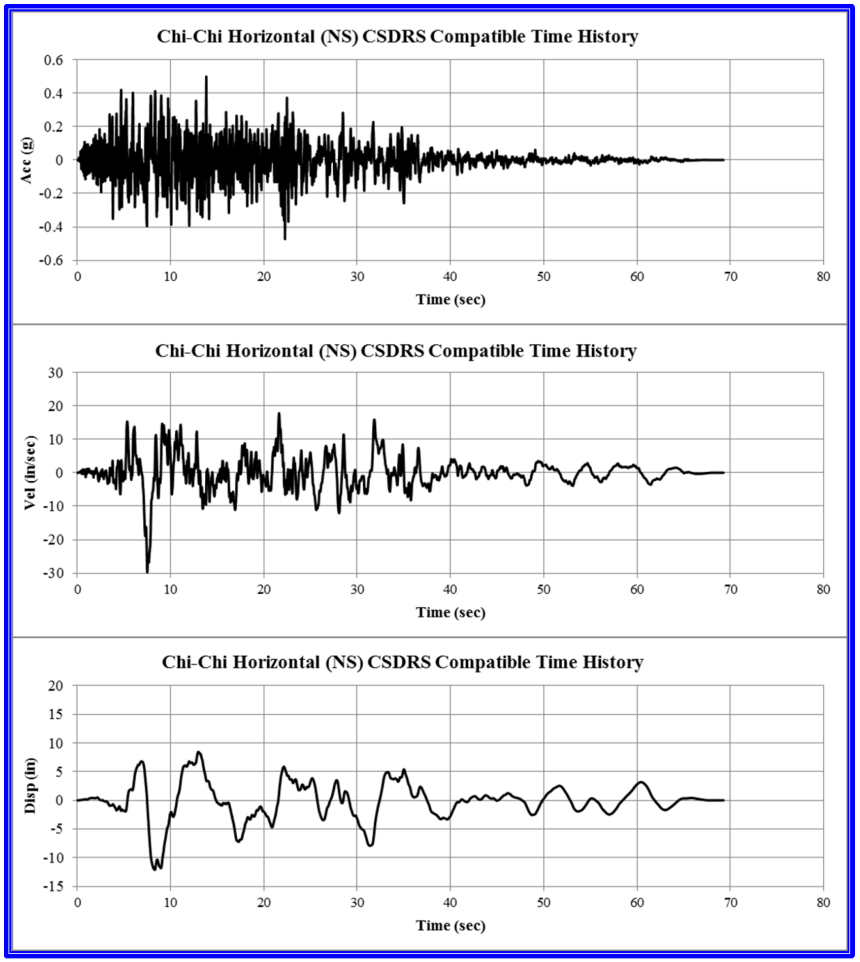
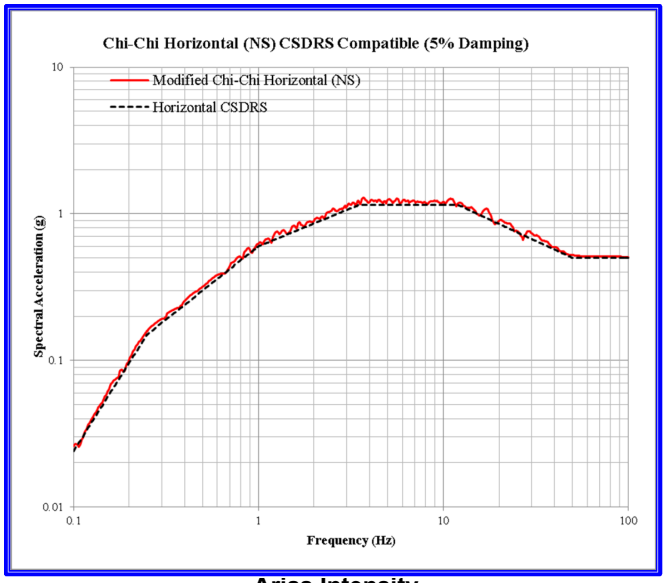


Figure 3.7.1-7d: Certified Seismic Design Response Spectra Compatible Time Histories for Chi-Chi North-South

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

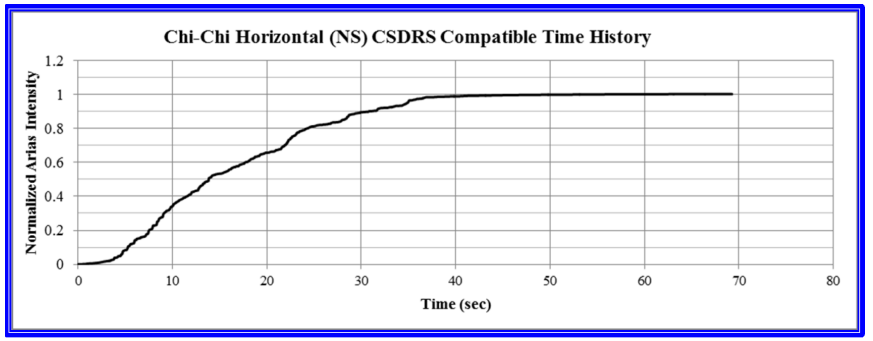
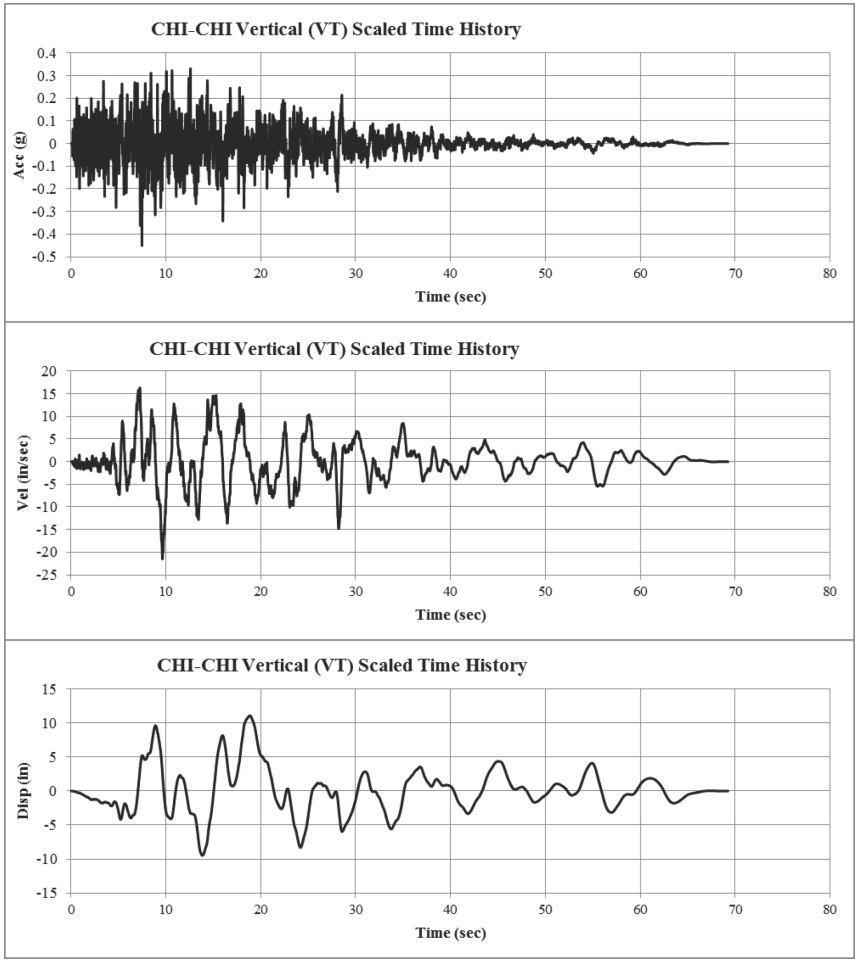


Figure 3.7.1-7e: Original Time Histories for Chi-Chi Vertical

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

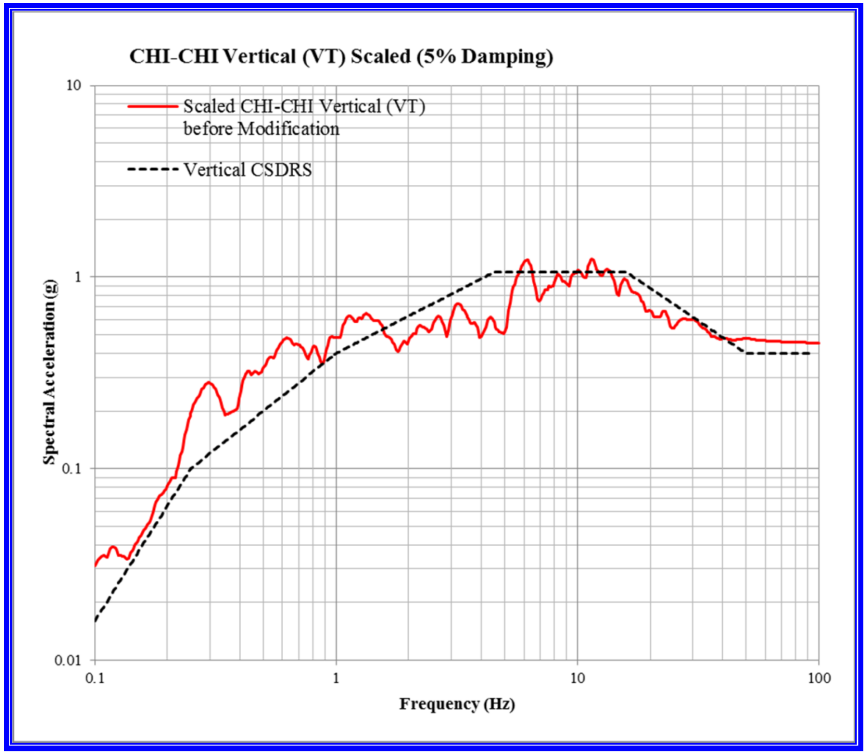
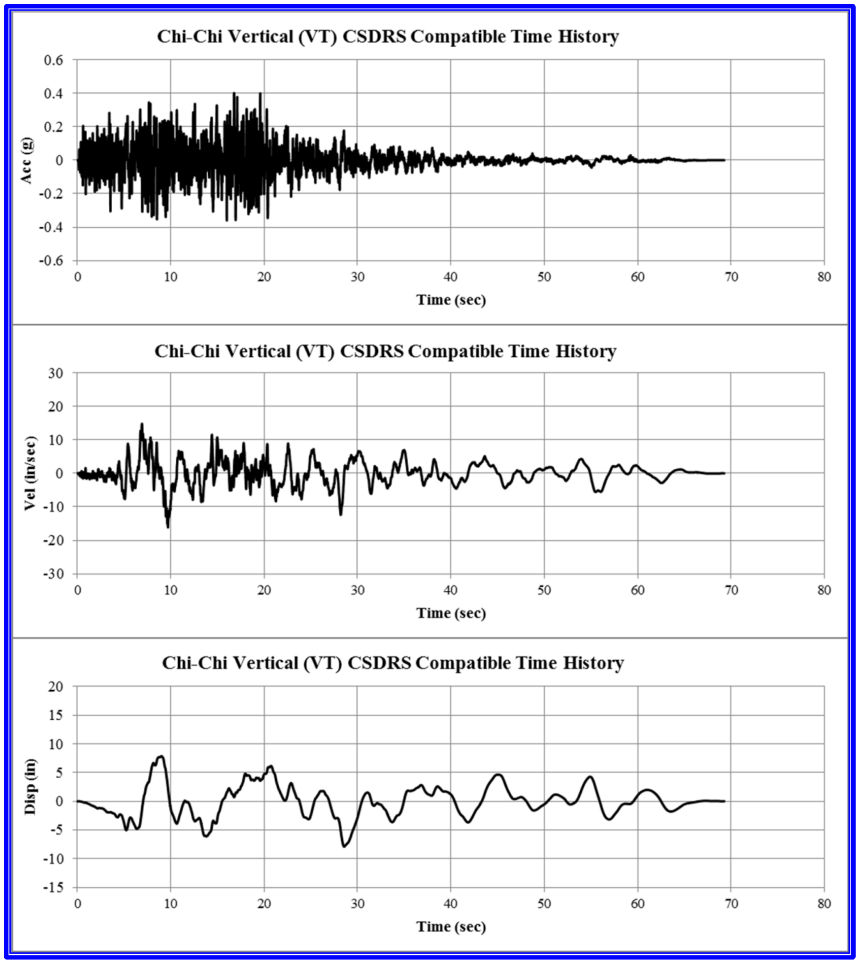
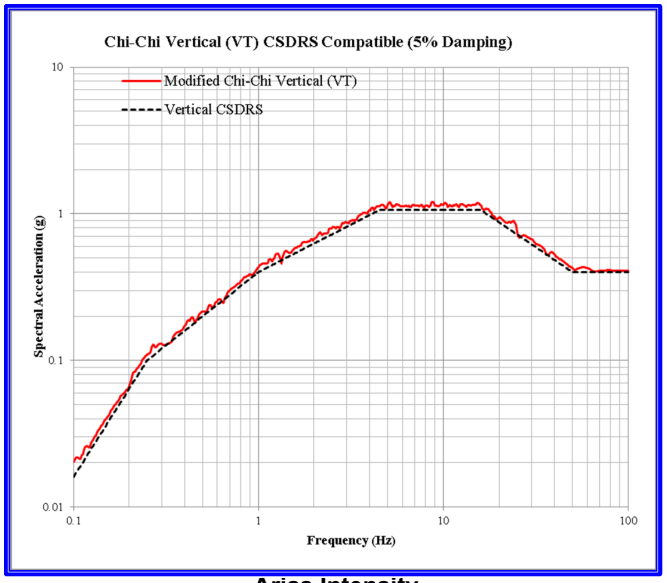


Figure 3.7.1-7f: Certified Seismic Design Response Spectra Compatible Time Histories for Chi-Chi Vertical

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

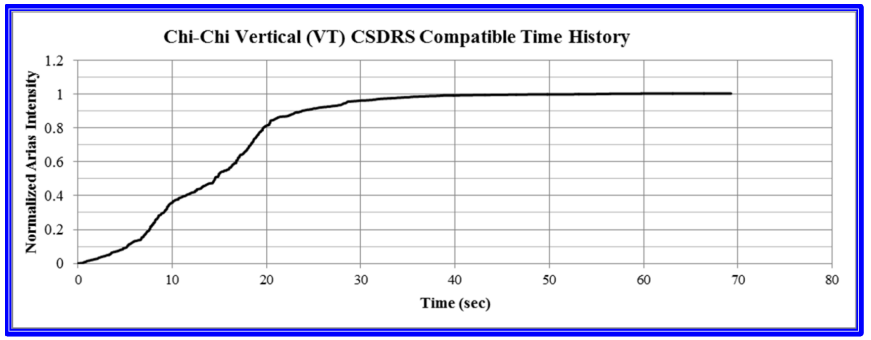
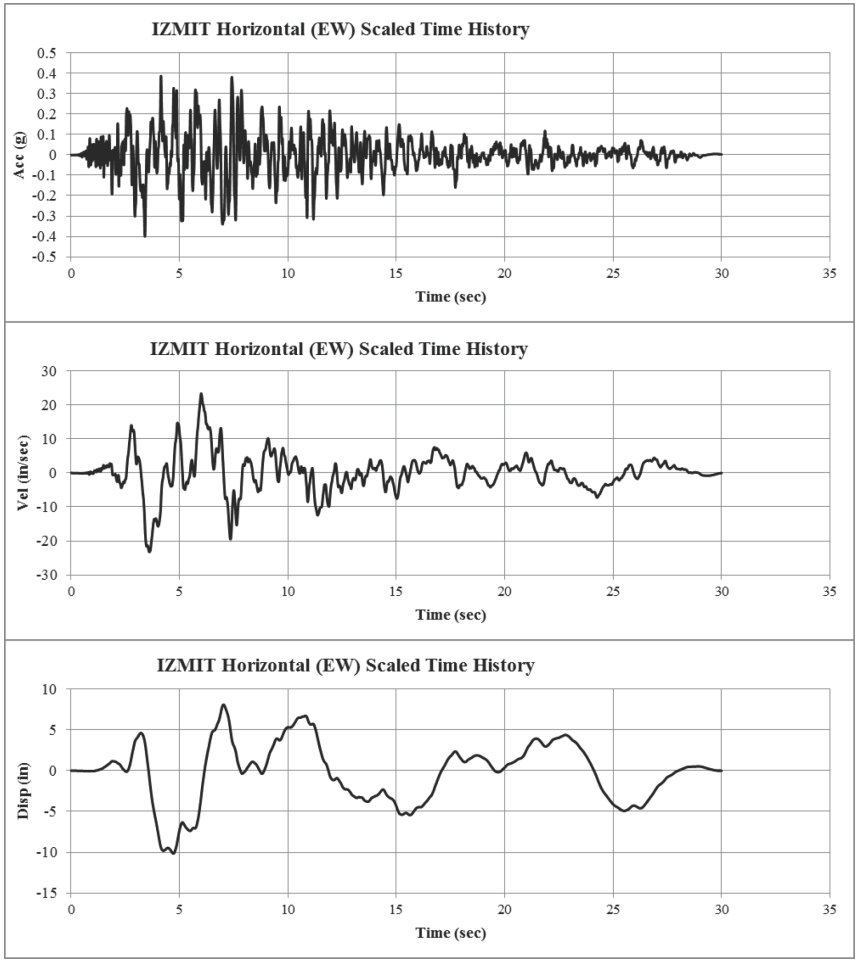


Figure 3.7.1-8a: Original Time Histories for Izmit East-West

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

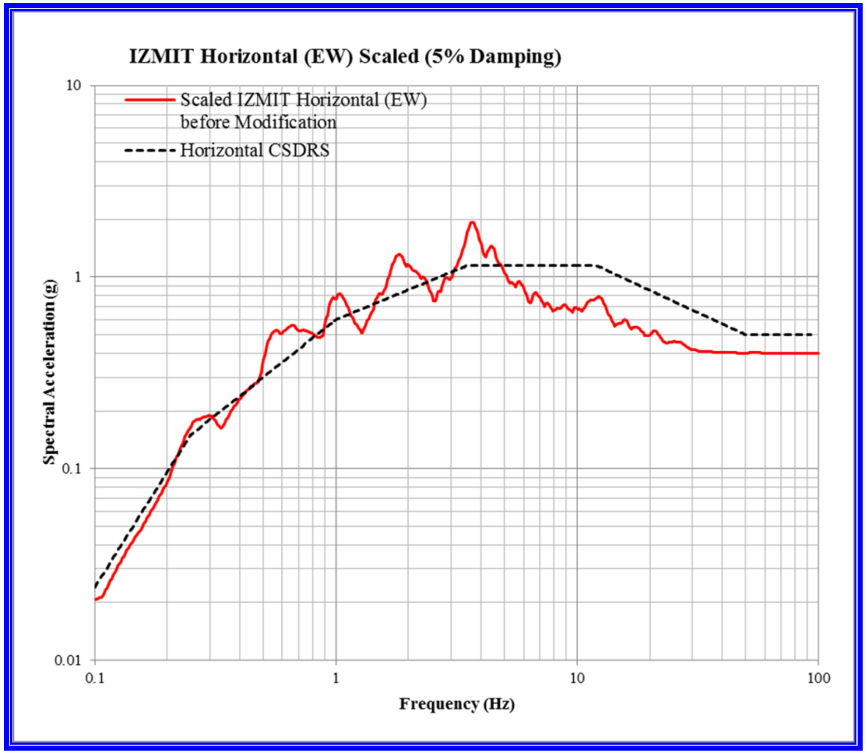
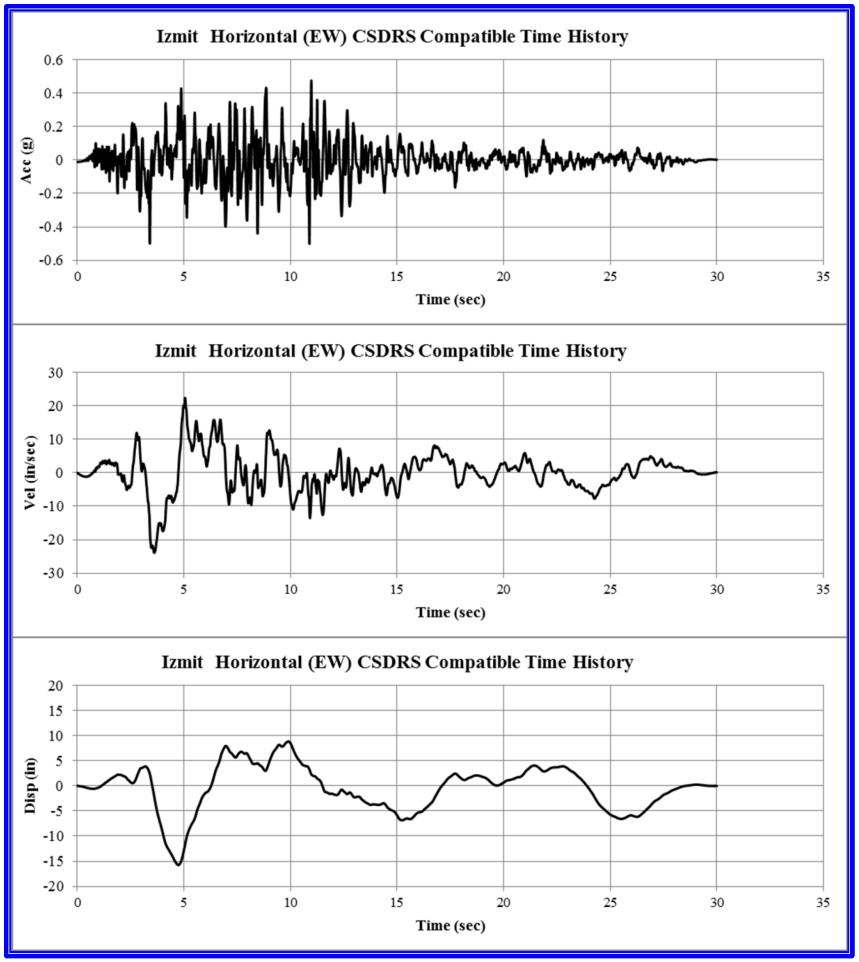
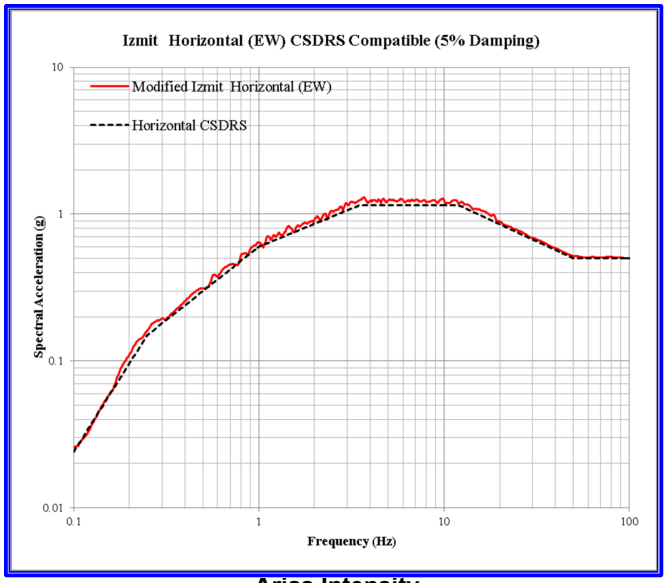


Figure 3.7.1-8b: Certified Seismic Design Response Spectra Compatible Time Histories for Izmit East-West

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

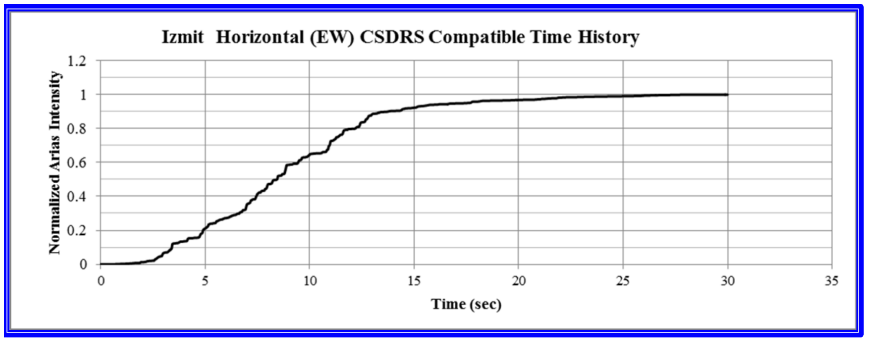
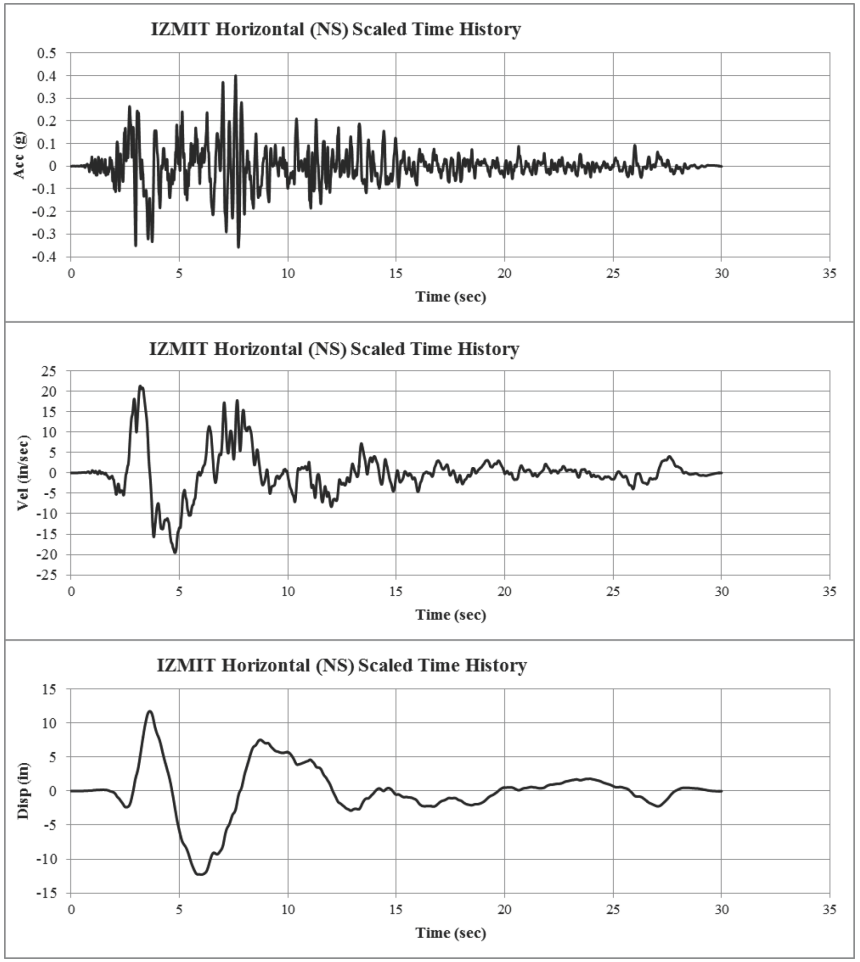


Figure 3.7.1-8c: Original Time Histories for Izmit North-South

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

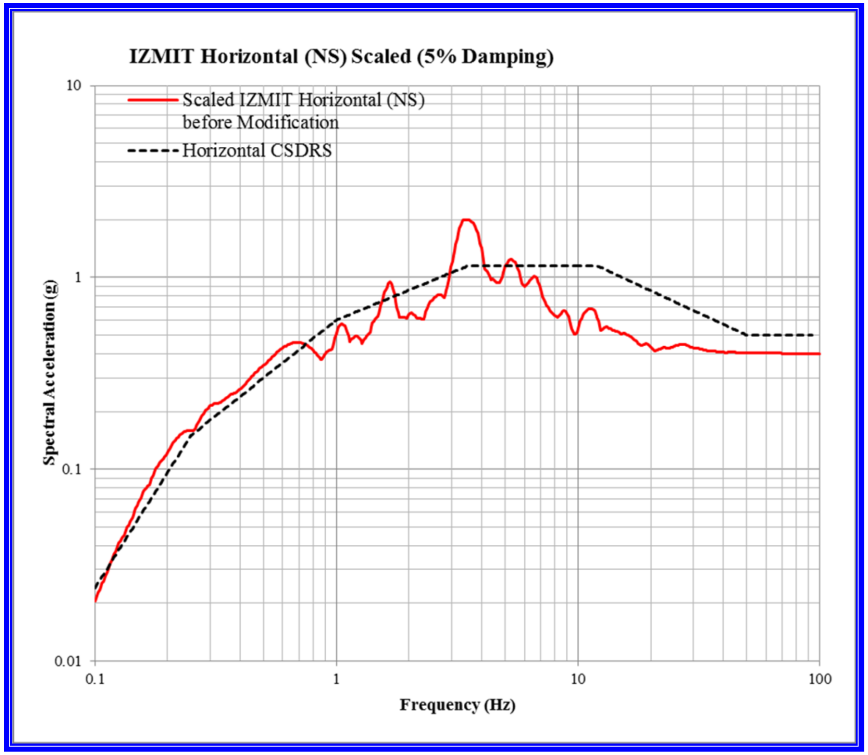
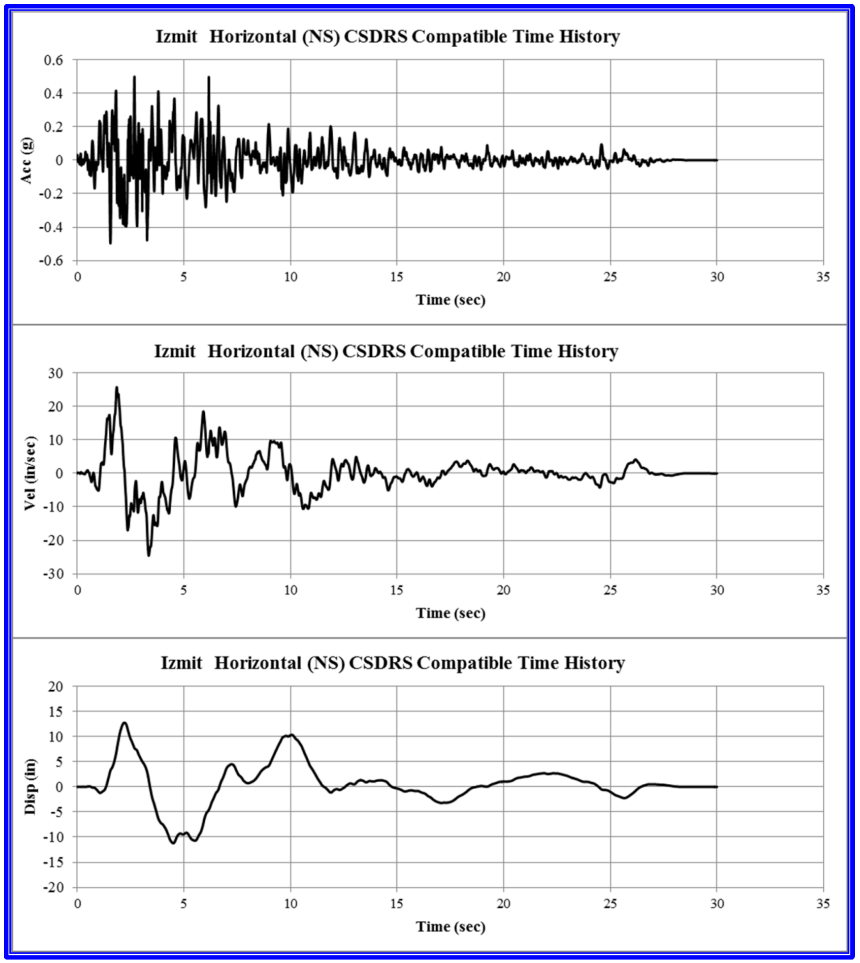
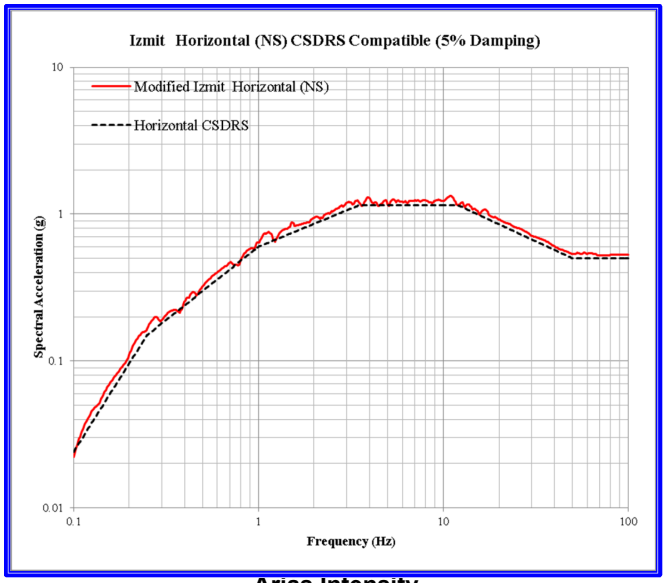


Figure 3.7.1-8d: Certified Seismic Design Response Spectra Compatible Time Histories for Izmit North-South

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

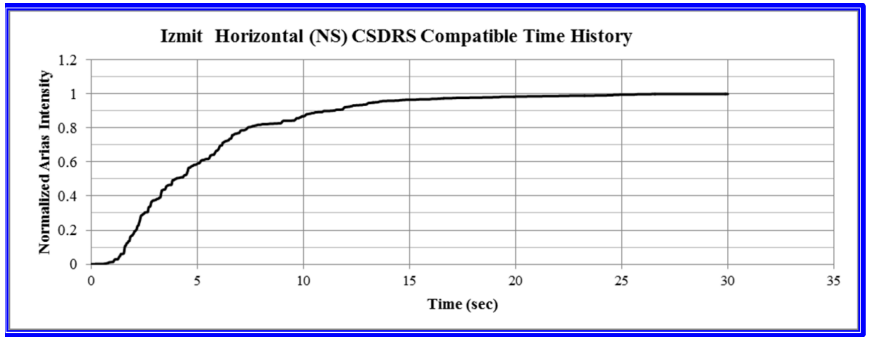
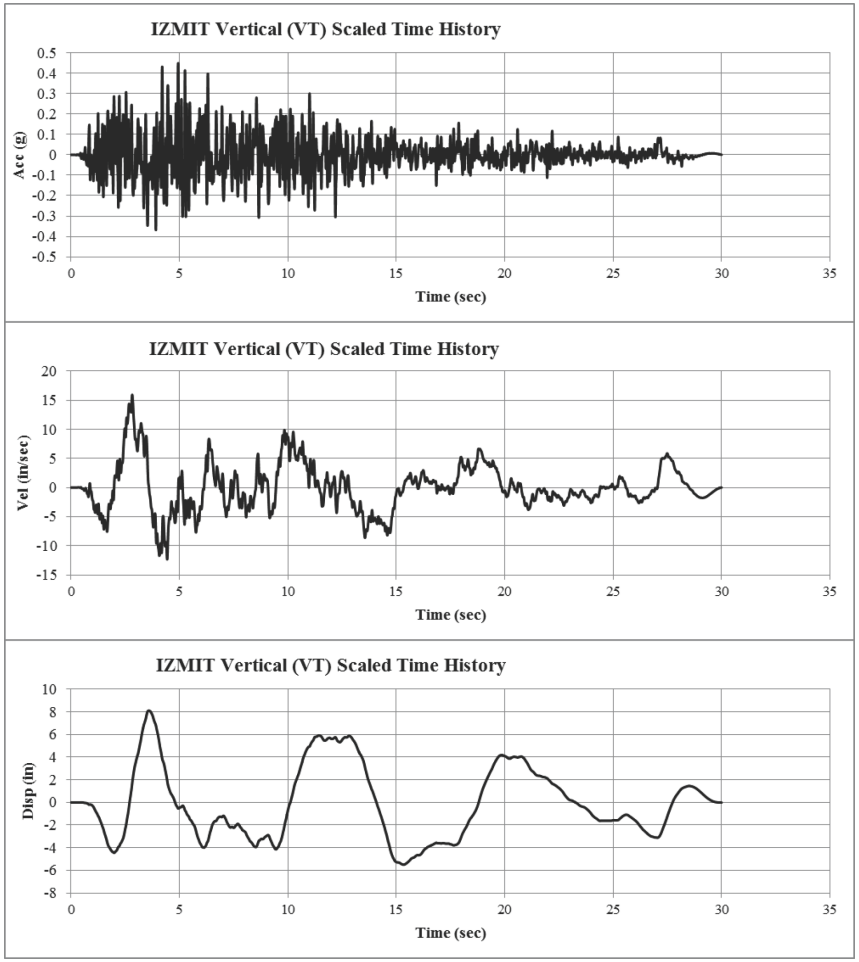


Figure 3.7.1-8e: Original Time Histories for Izmit Vertical

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

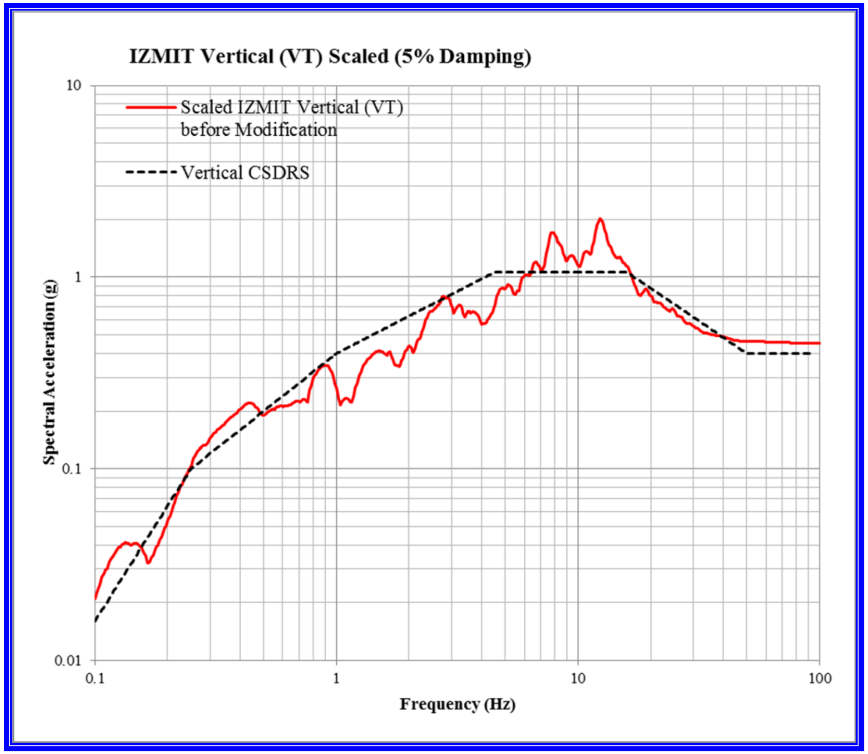
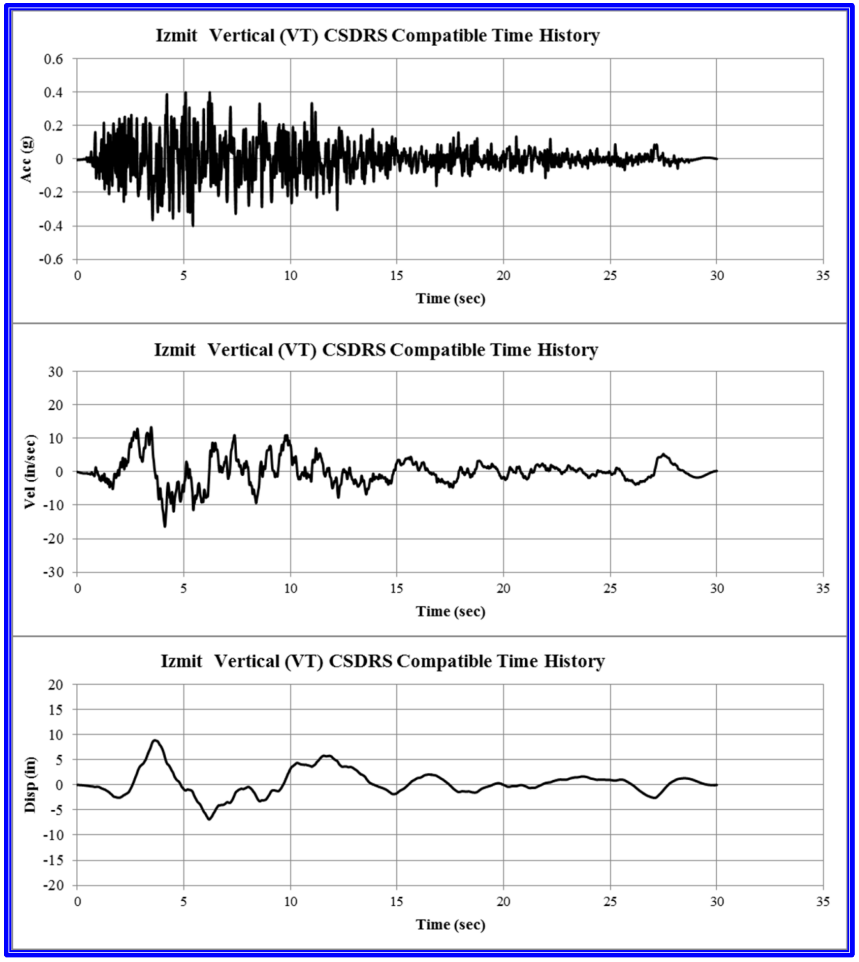
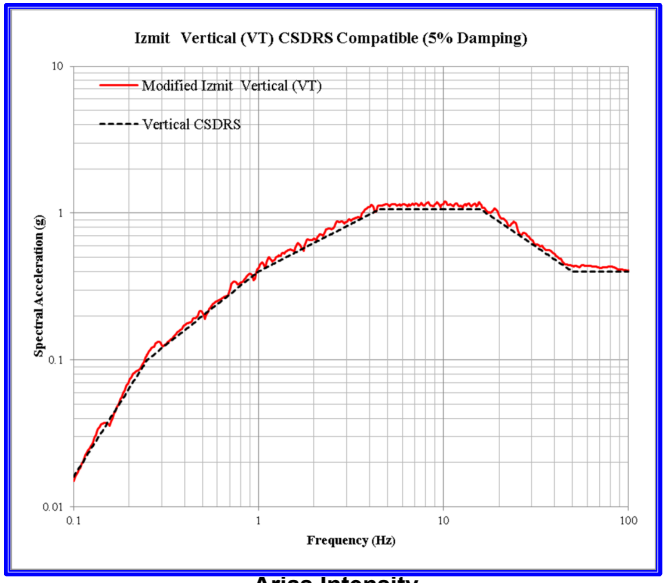


Figure 3.7.1-8f: Certified Seismic Design Response Spectra Compatible Time Histories for Izmit Vertical

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

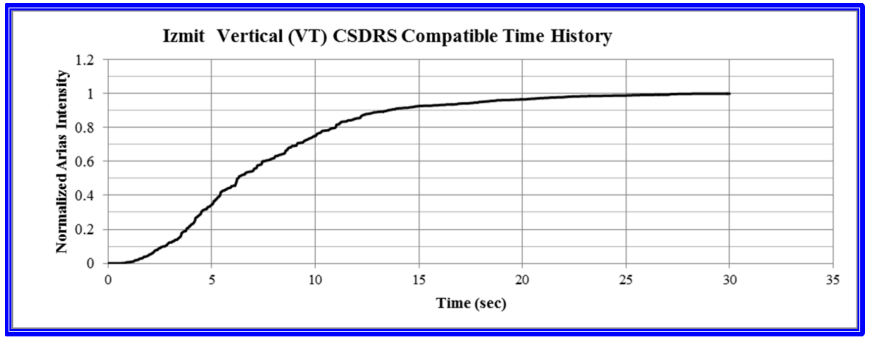
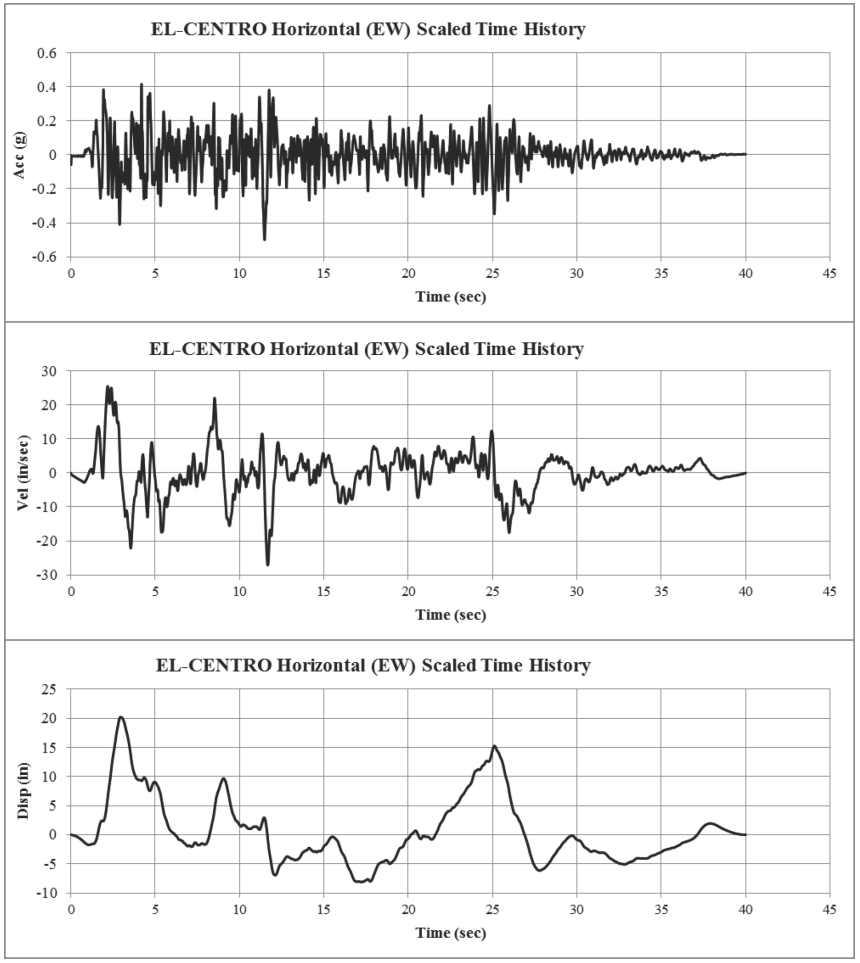


Figure 3.7.1-9a: Original Time Histories for El Centro East-West

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

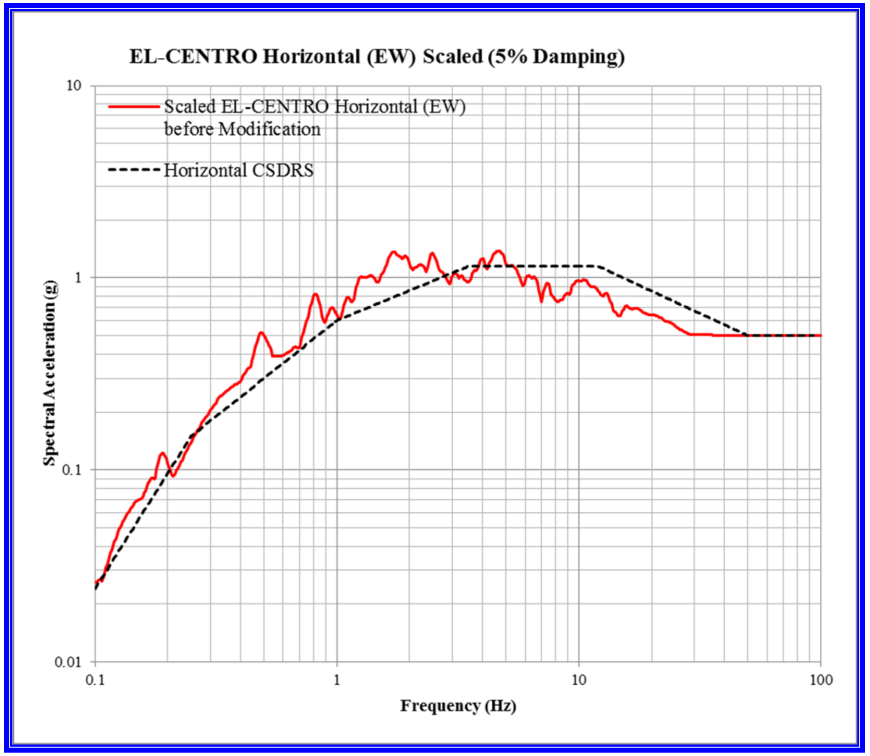
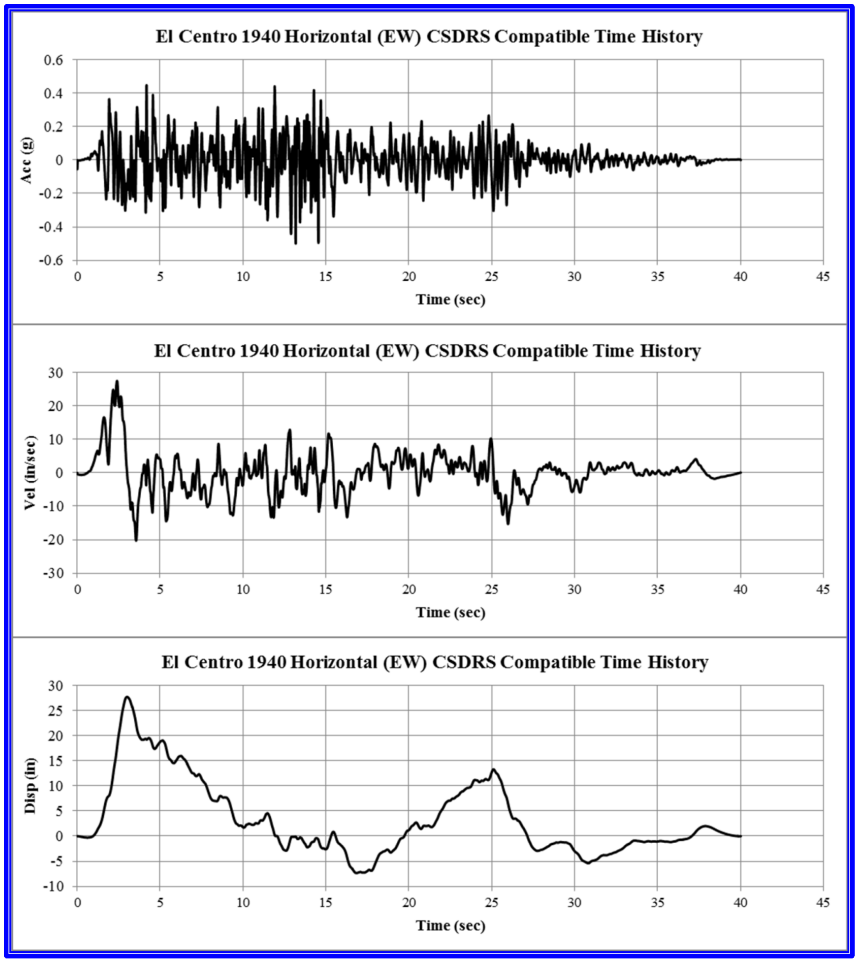
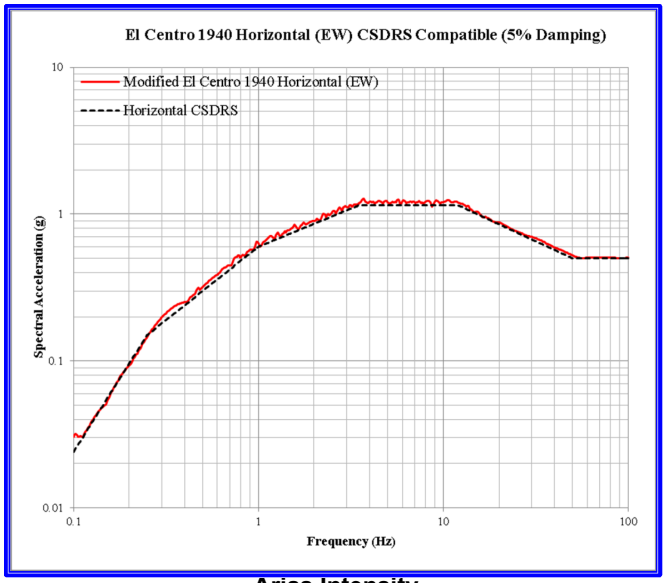


Figure 3.7.1-9b: Certified Seismic Design Response Spectra Compatible Time Histories for El Centro East-West

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

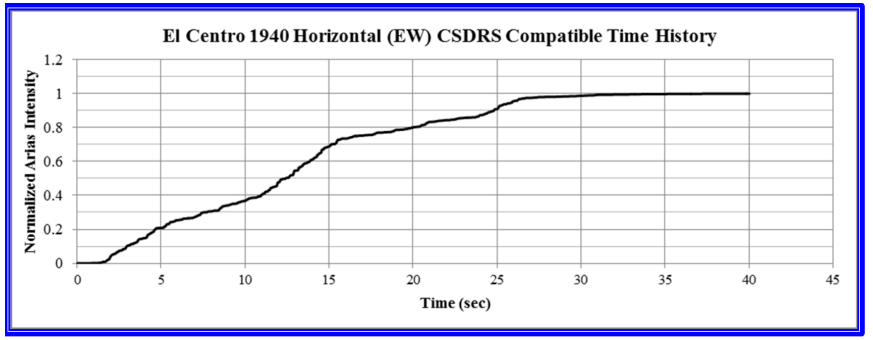
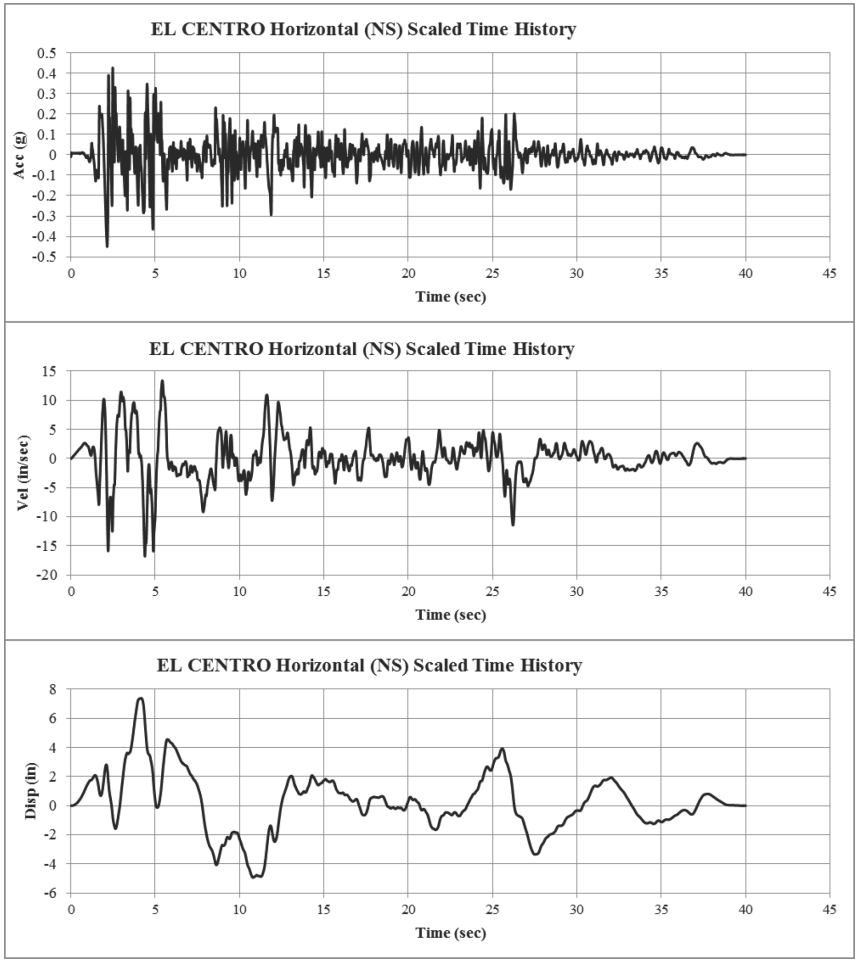


Figure 3.7.1-9c: Original Time Histories for El Centro North South

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

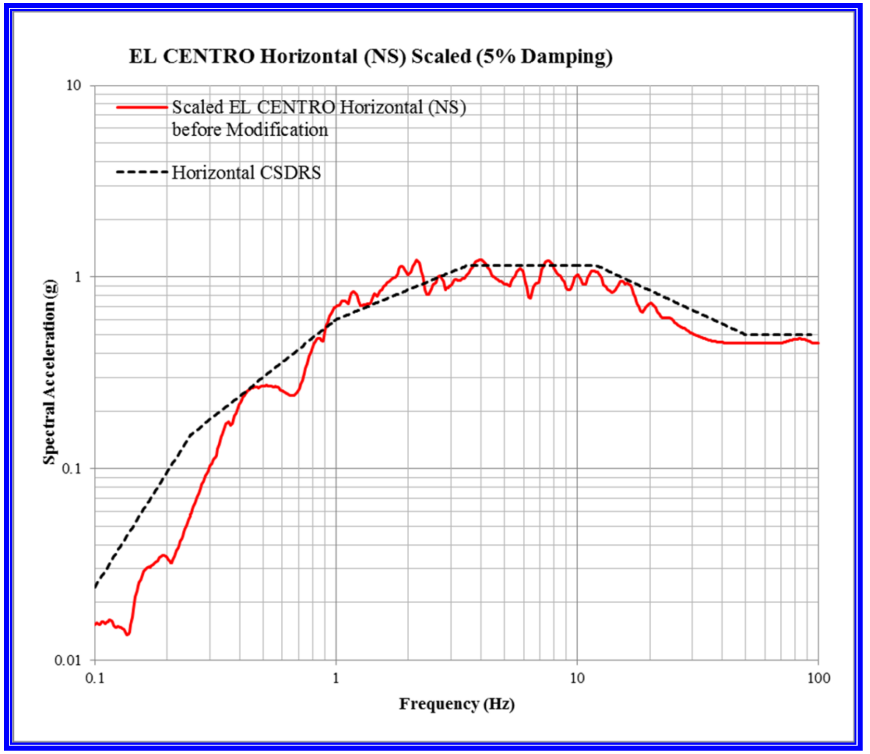
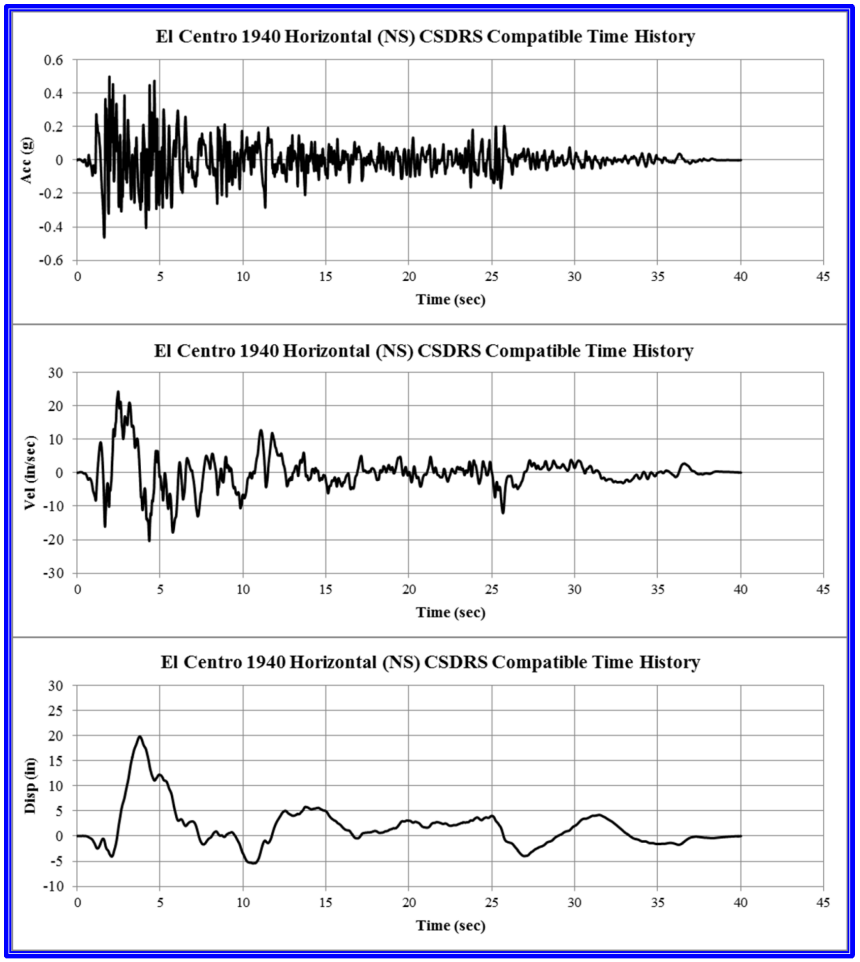
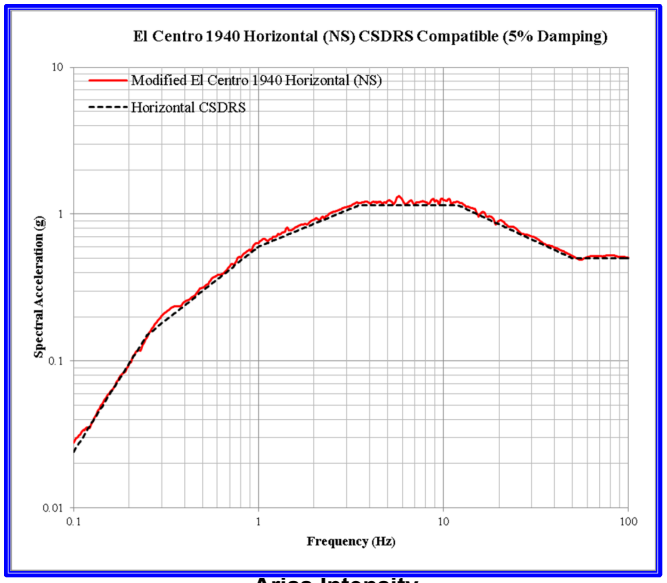


Figure 3.7.1-9d: Certified Seismic Design Response Spectra Compatible Time Histories for El Centro North-South

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

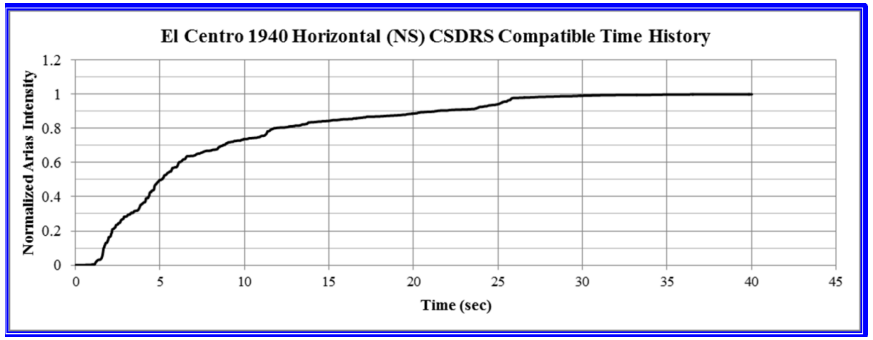
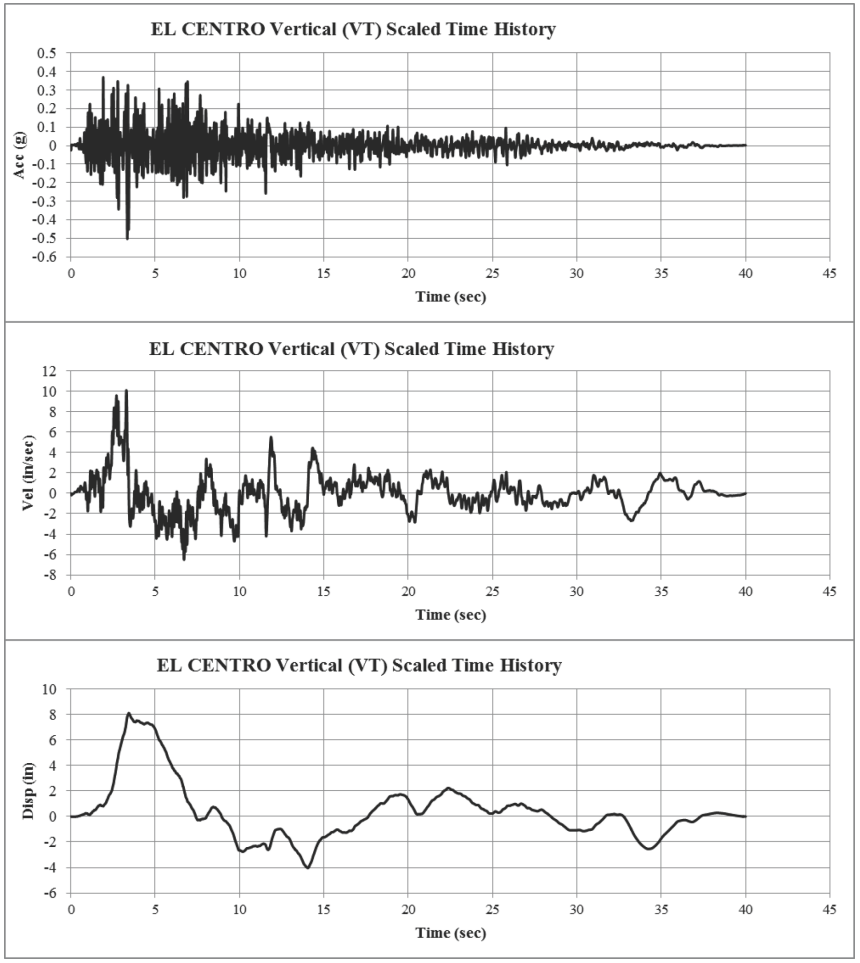


Figure 3.7.1-9e: Original Time Histories for El Centro Vertical

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

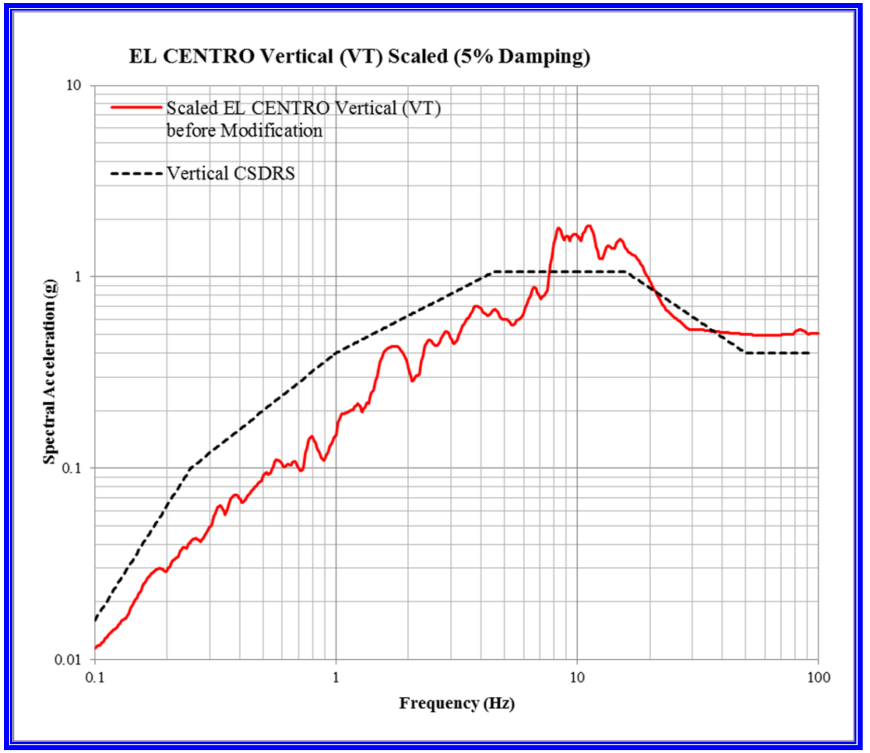
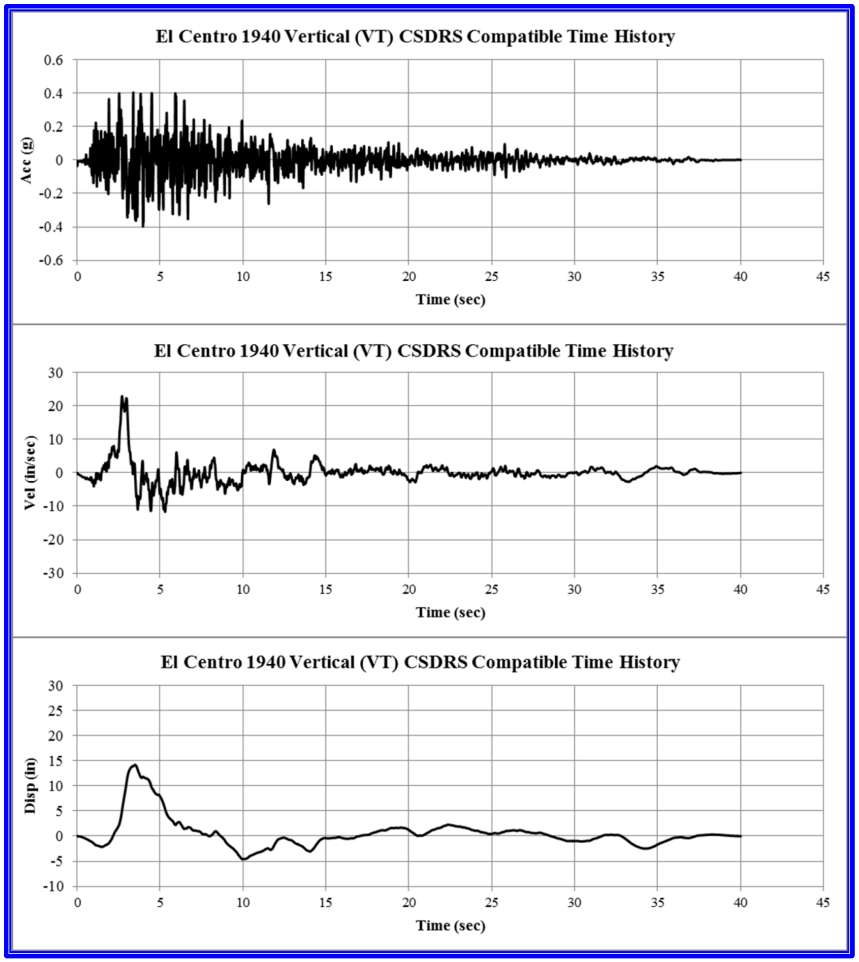
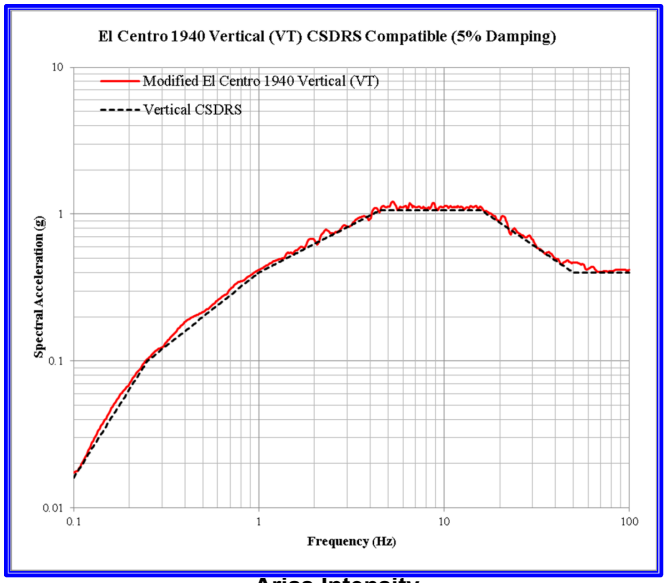


Figure 3.7.1-9f: Certified Seismic Design Response Spectra Compatible Time Histories for El Centro Vertical

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

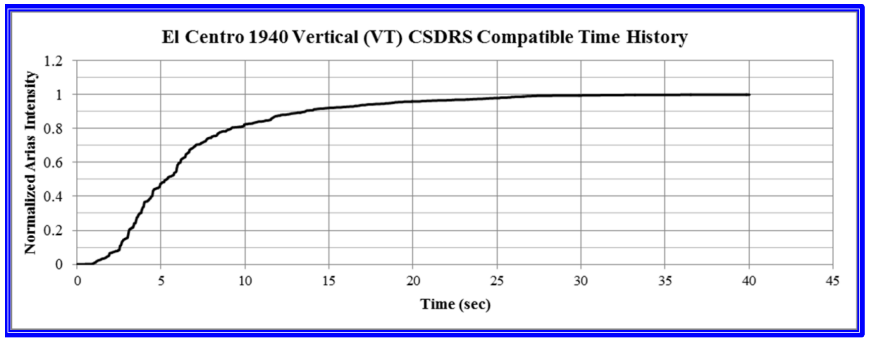
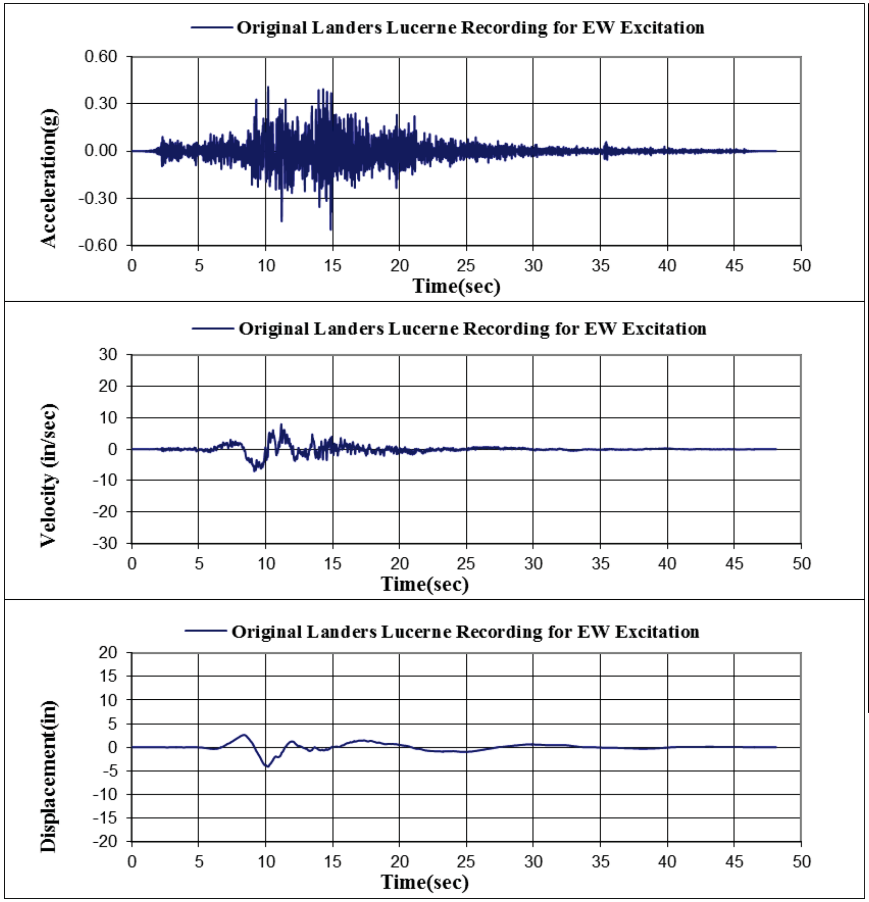


Figure 3.7.1-10a: Original Time Histories for Lucerne East-West

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

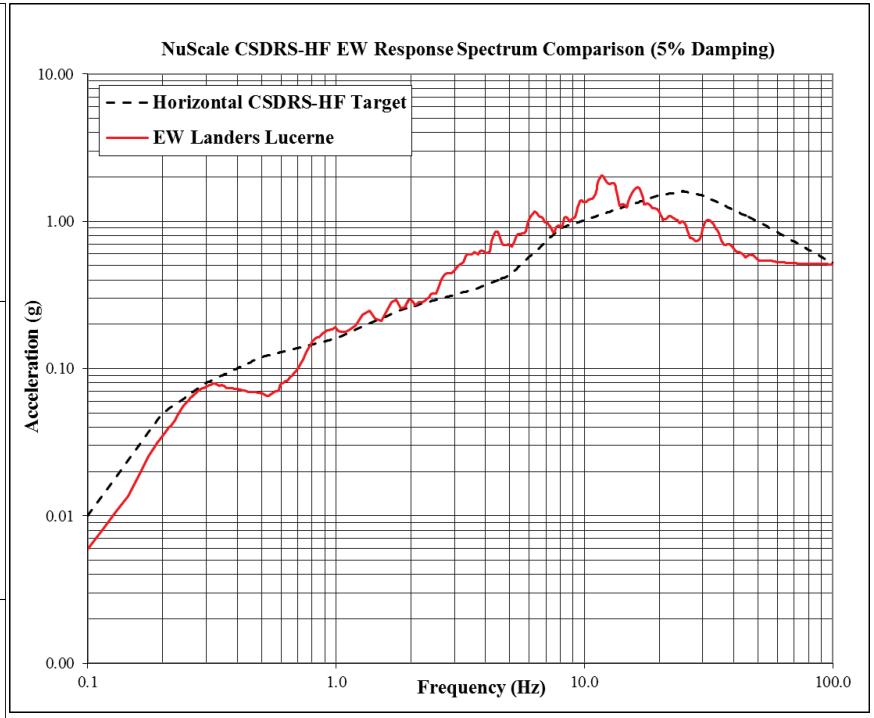
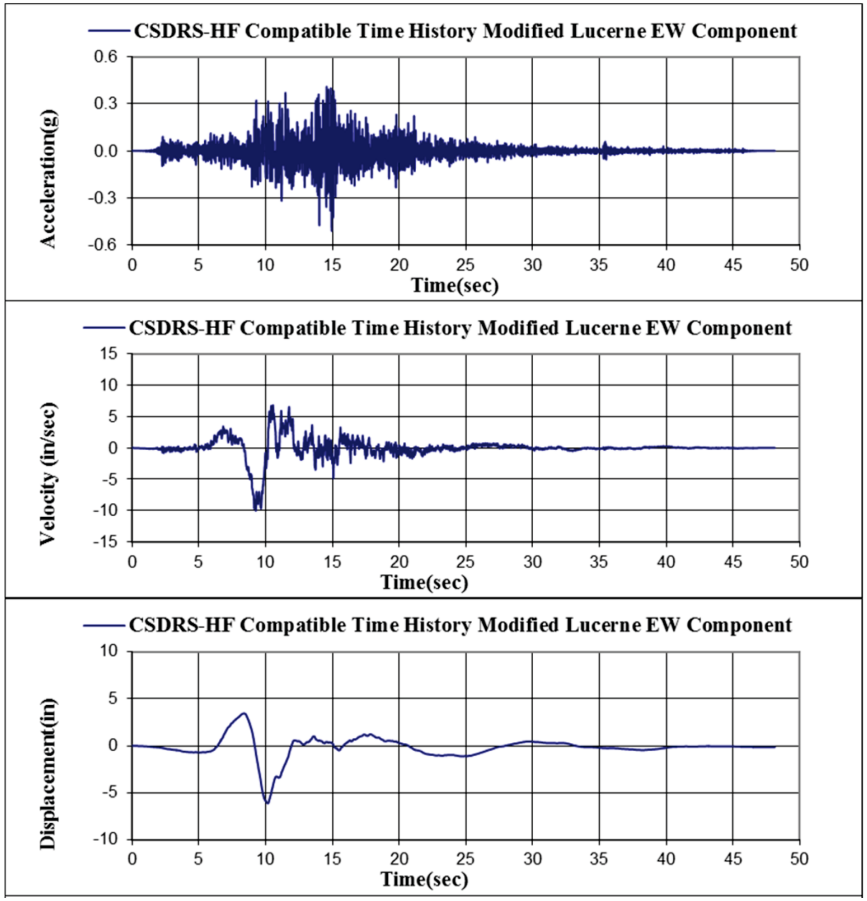
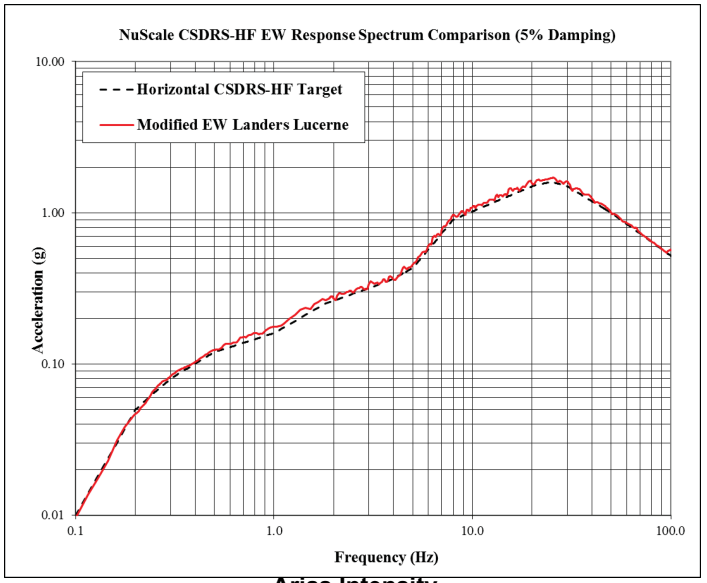


Figure 3.7.1-10b: Certified Seismic Design Response Spectra - High Frequency Compatible Time Histories for Lucerne East-West

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

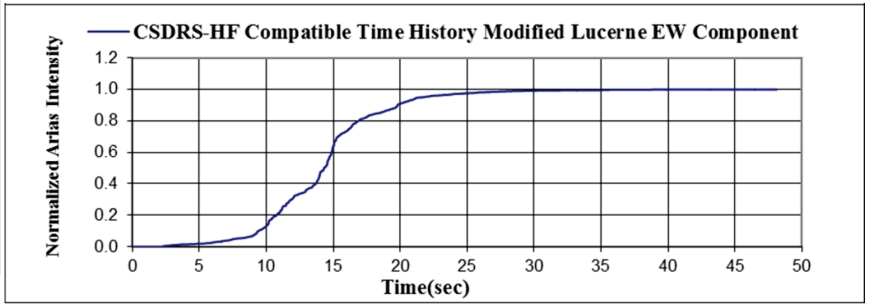
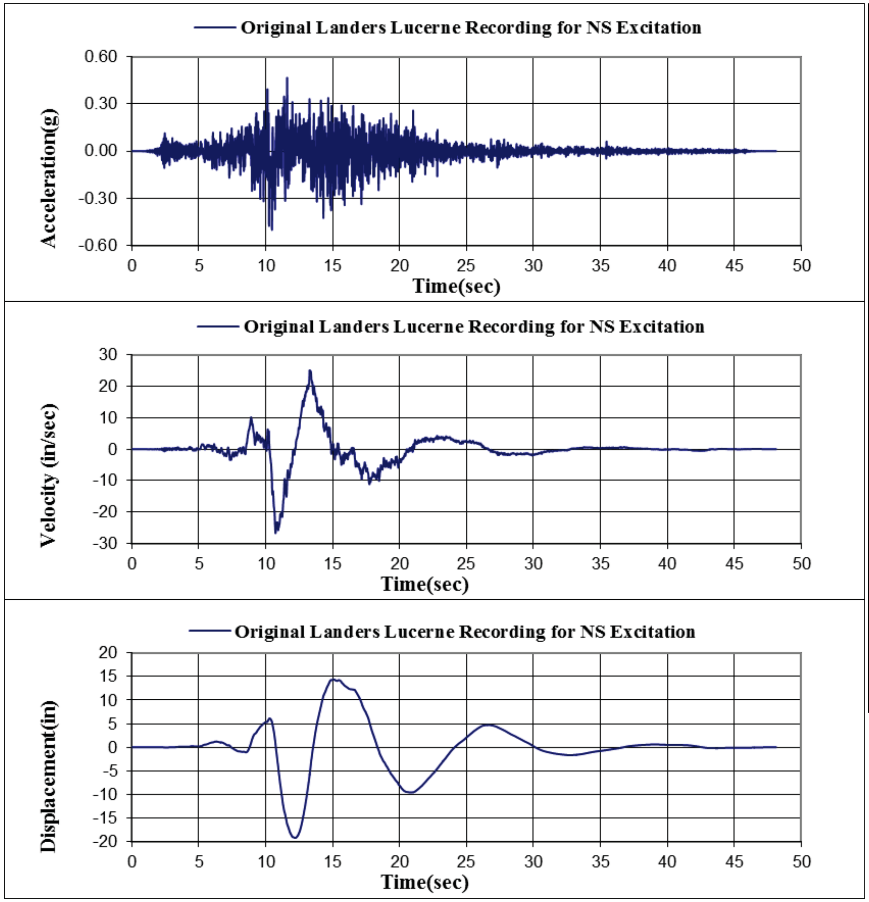


Figure 3.7.1-10c: Original Time Histories for Lucerne North-South

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

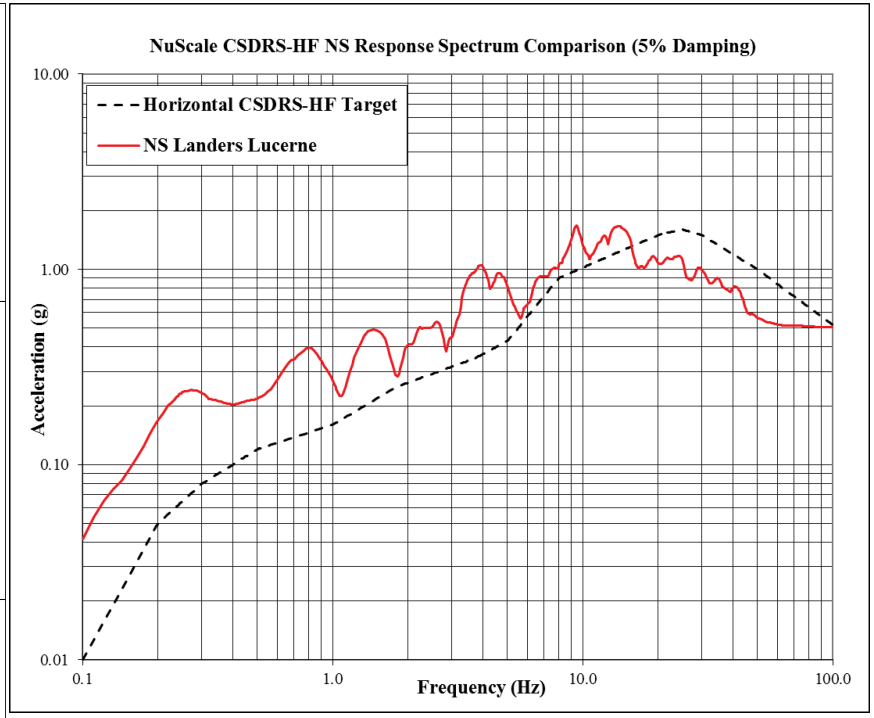
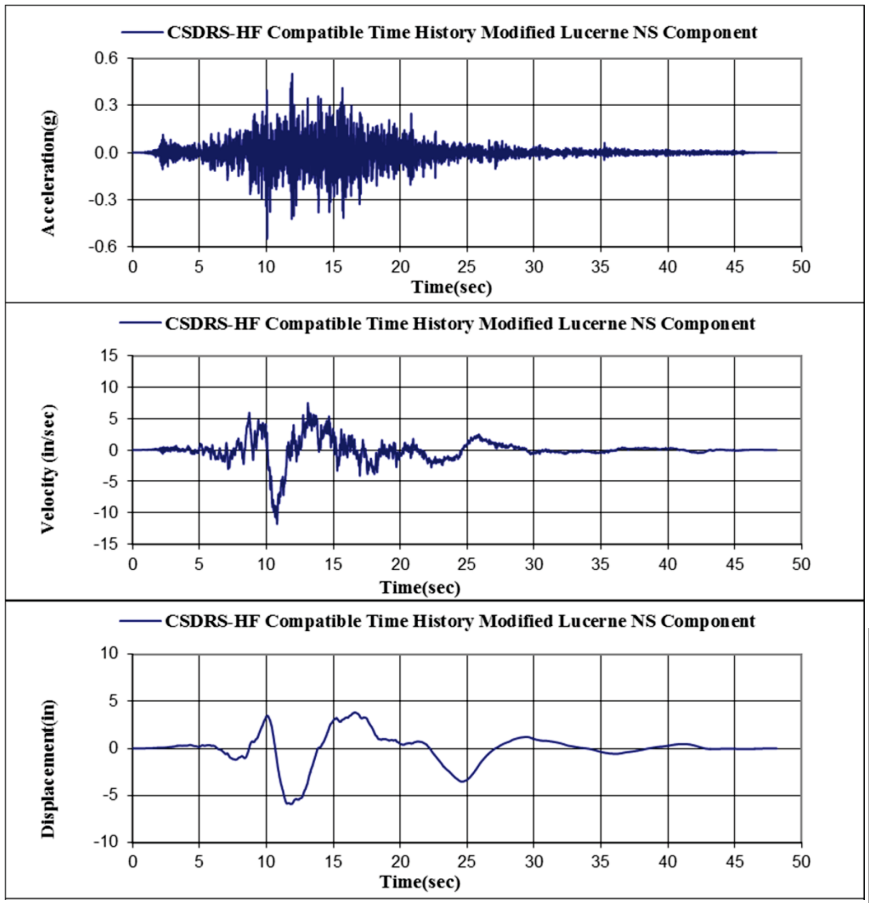
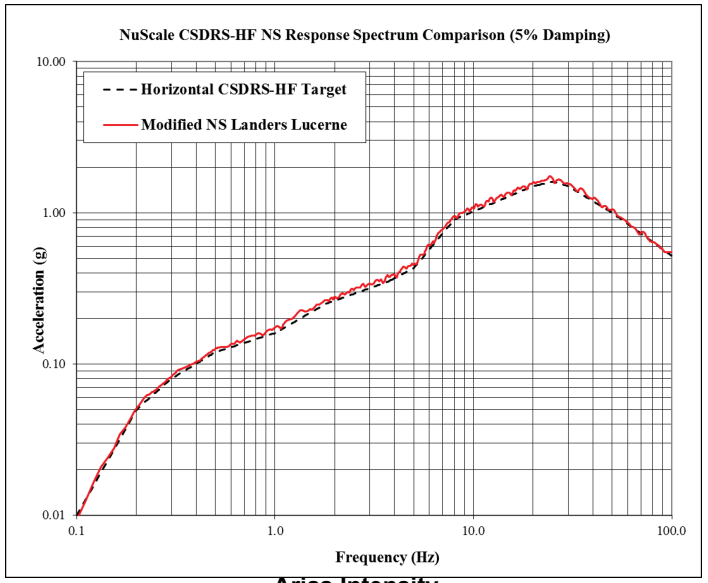


Figure 3.7.1-10d: Certified Seismic Design Response Spectra - High Frequency Compatible Time Histories for Lucerne North-South

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

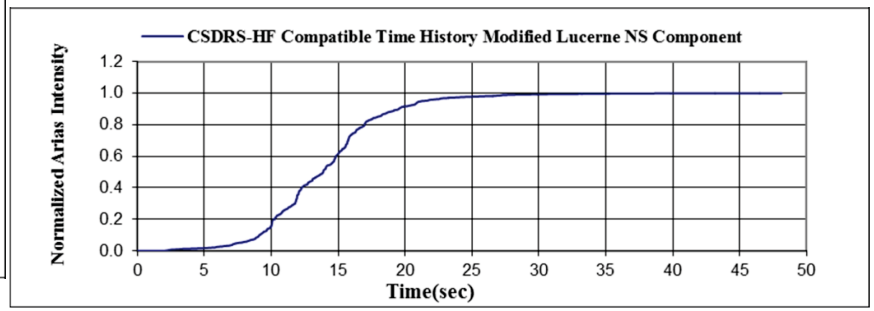
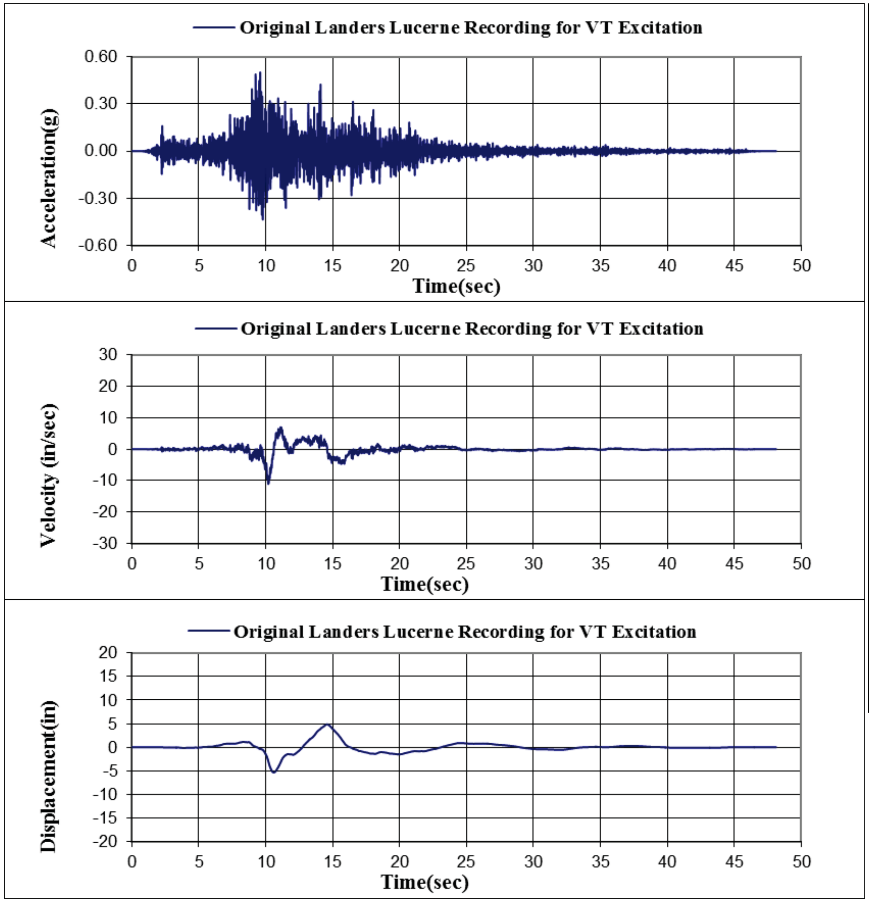


Figure 3.7.1-10e: Original Time Histories for Lucerne Vertical

Acceleration, Velocity, and Displacement Time Histories



Response Spectrum Scaled to CSDRS

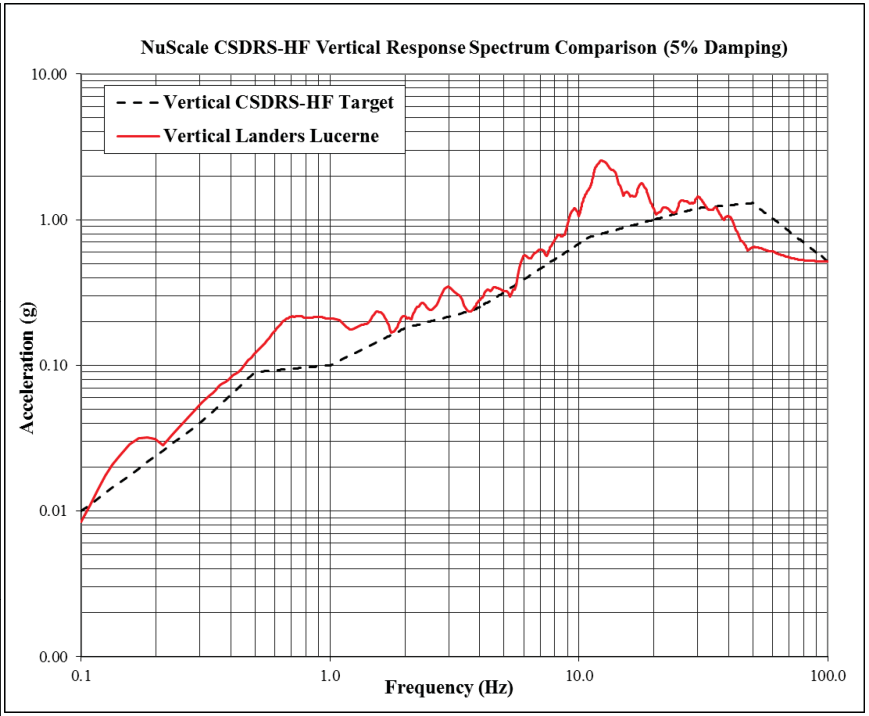
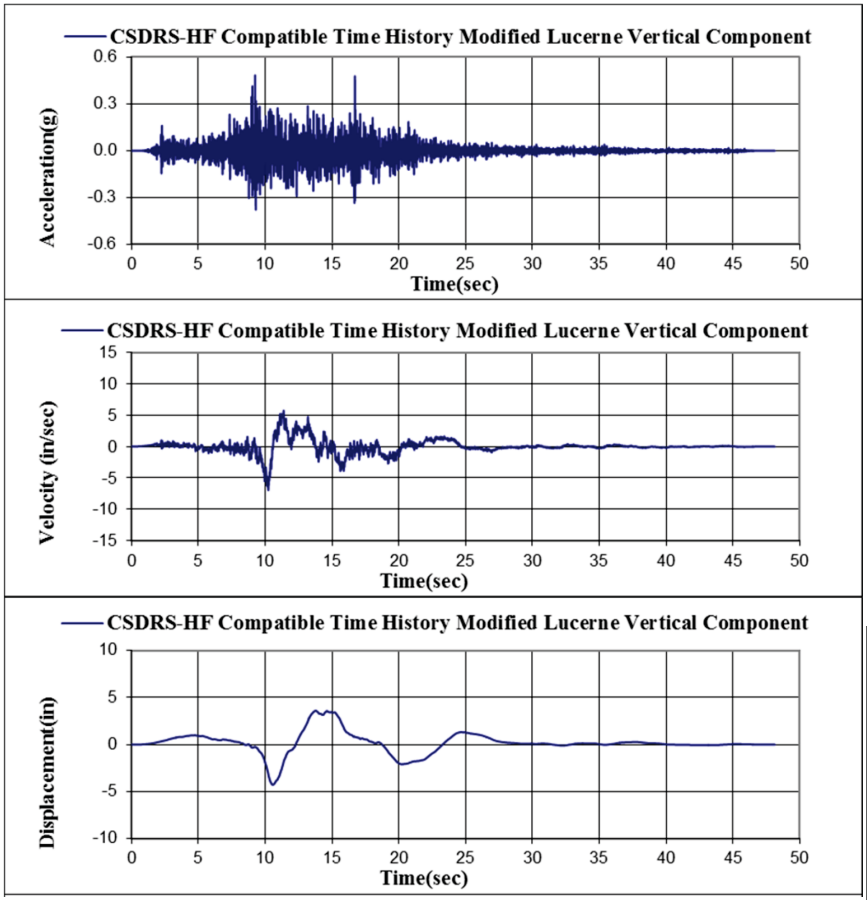
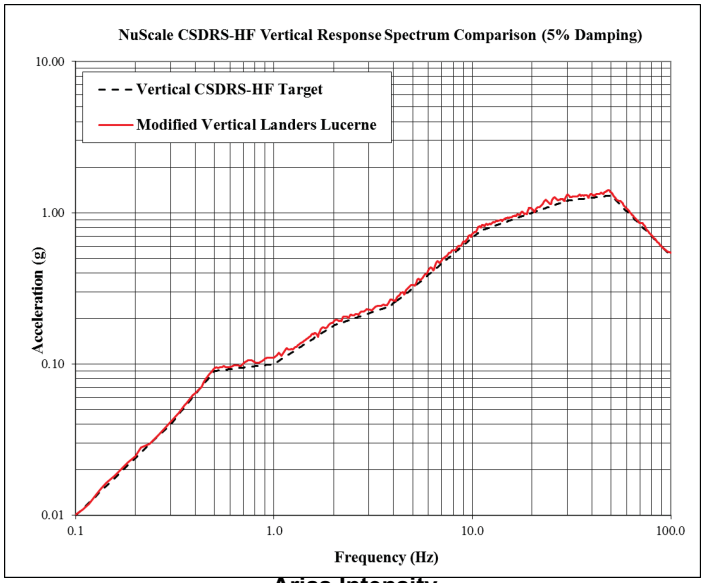


Figure 3.7.1-10f: Certified Seismic Design Response Spectra - High Frequency Compatible Time Histories for Lucerne Vertical

Acceleration, Velocity, and Displacement Time Histories



Modified Response Spectrum Compared to CSDRS



Arias Intensity

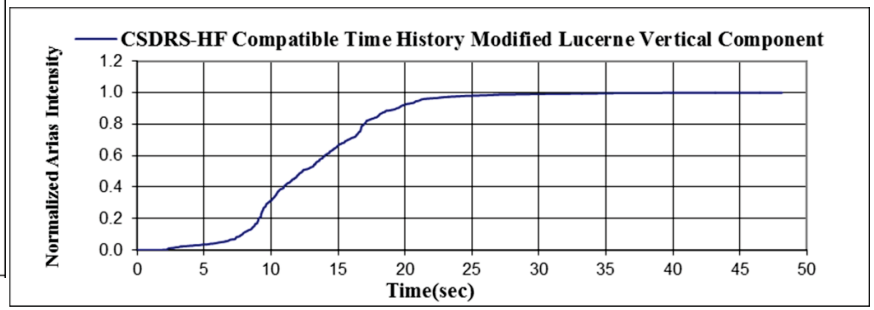
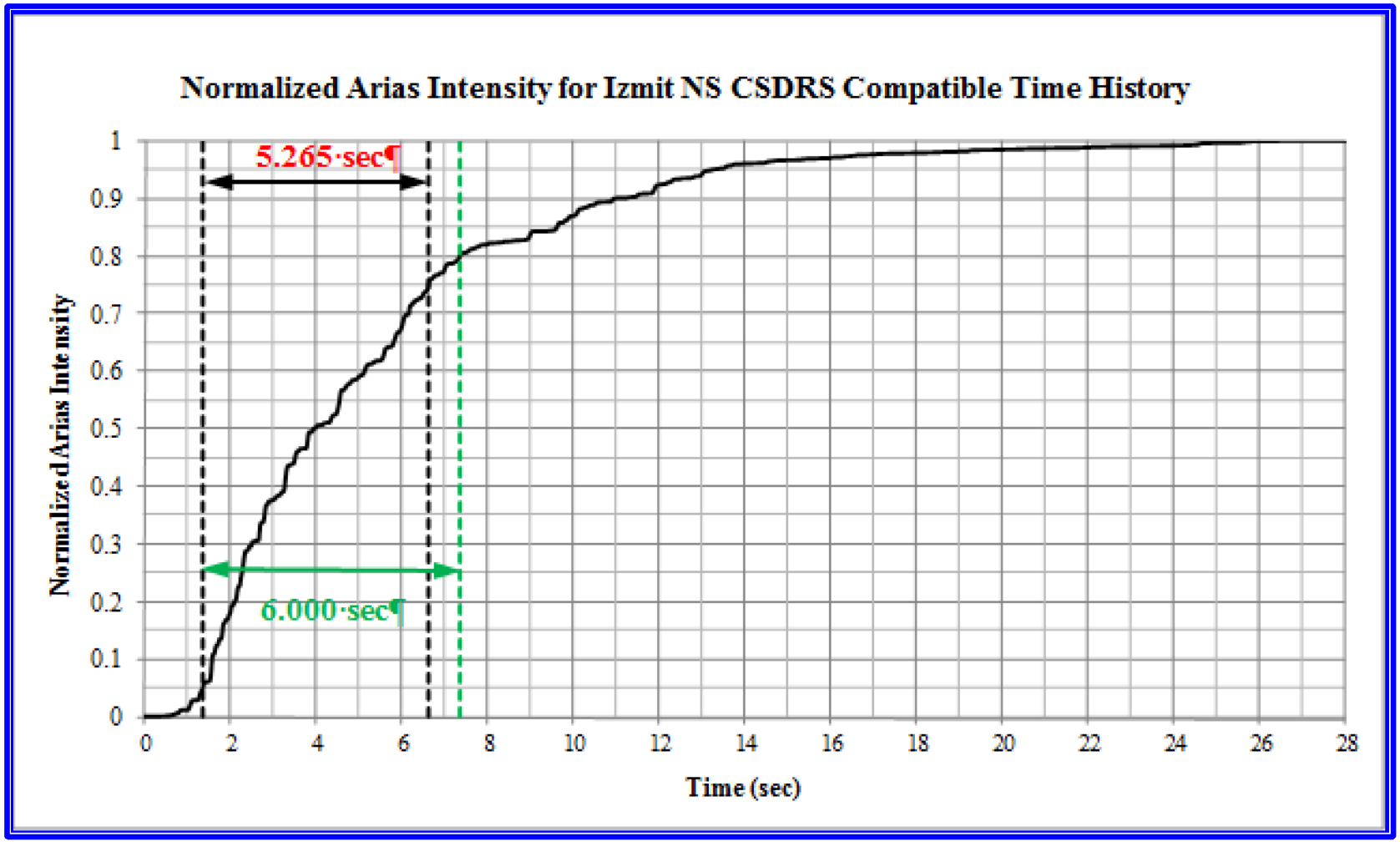


Figure 3.7.1-11: Normalized Arias Intensity Curve of North-South Component of Izmit Time History



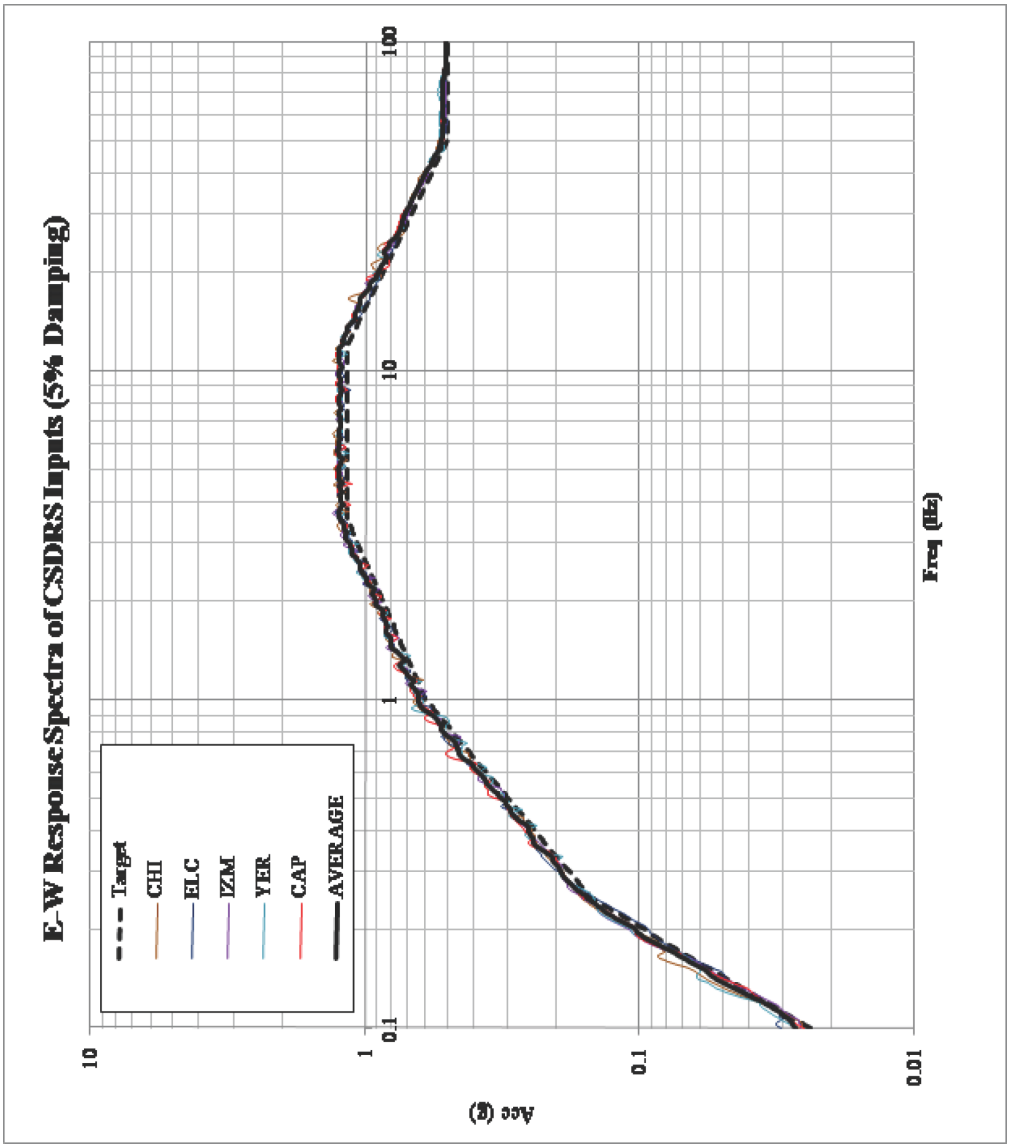


Figure 3.7-12: Average Response Spectra East-West

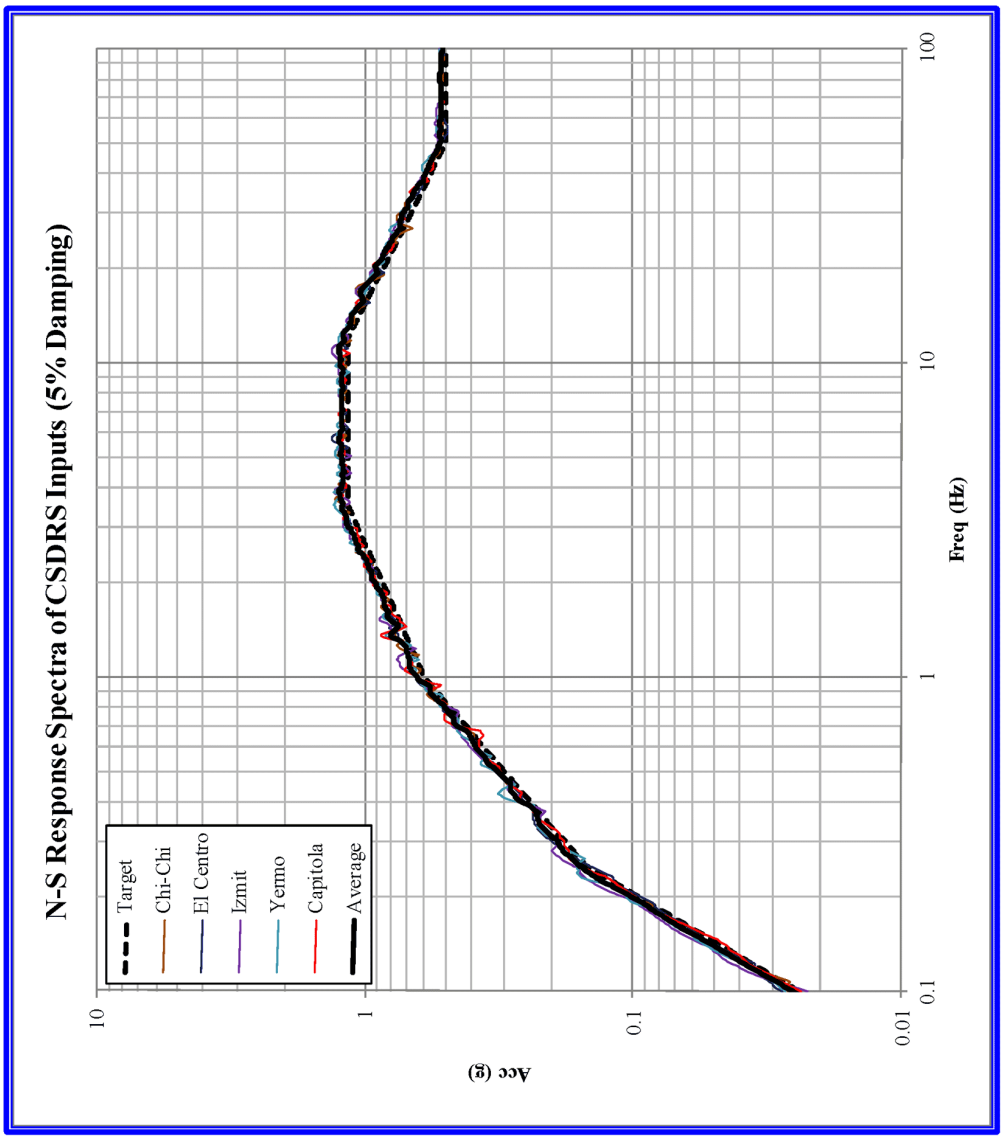


Figure 3.7.1-13: Average Response Spectra North-South

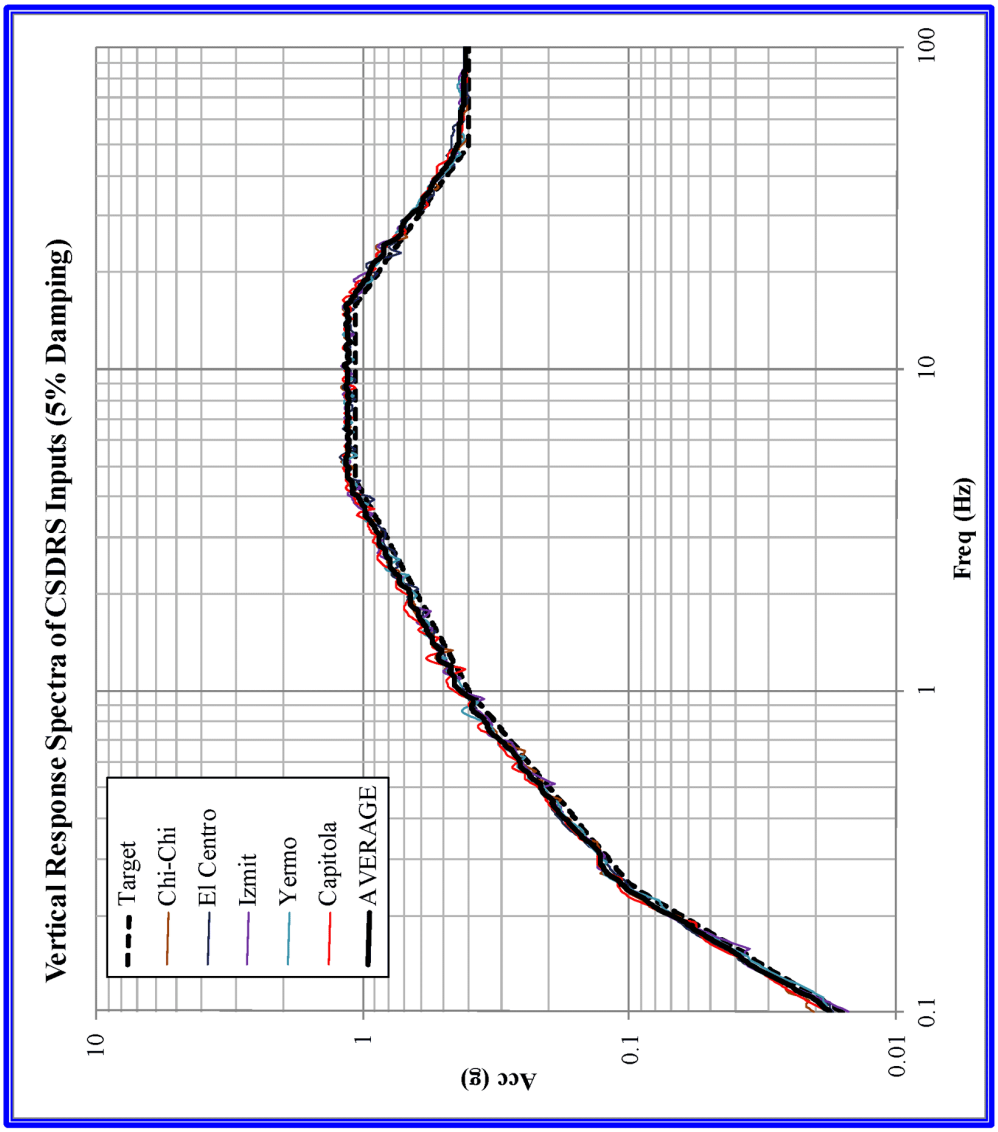
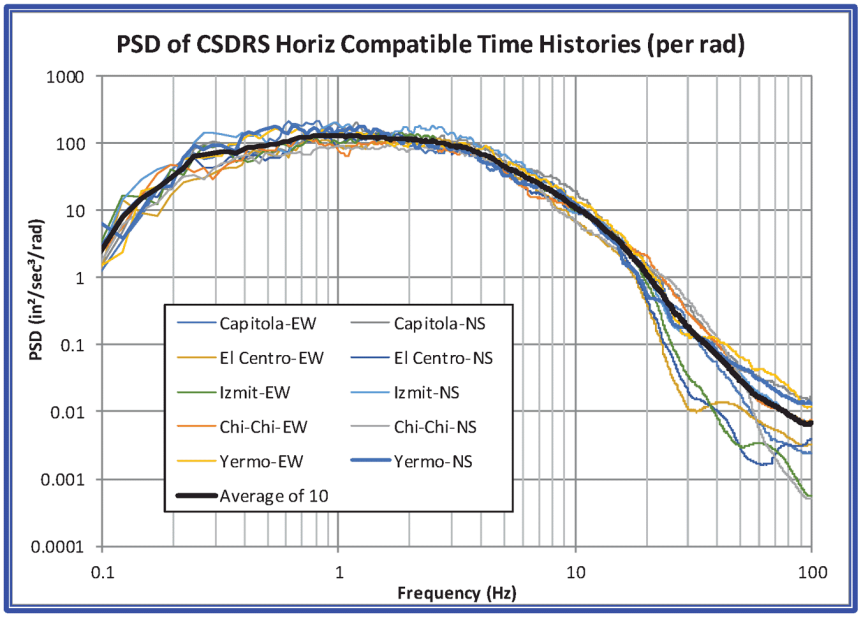
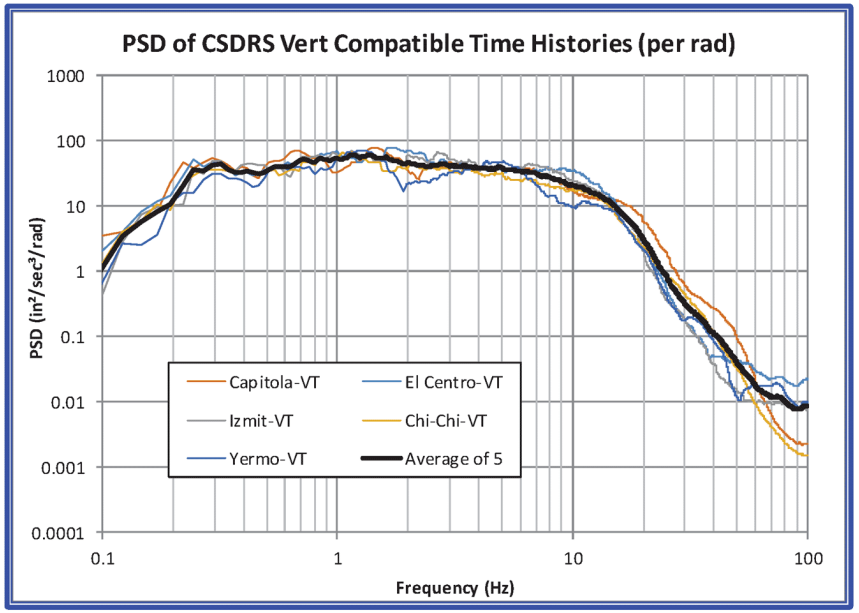
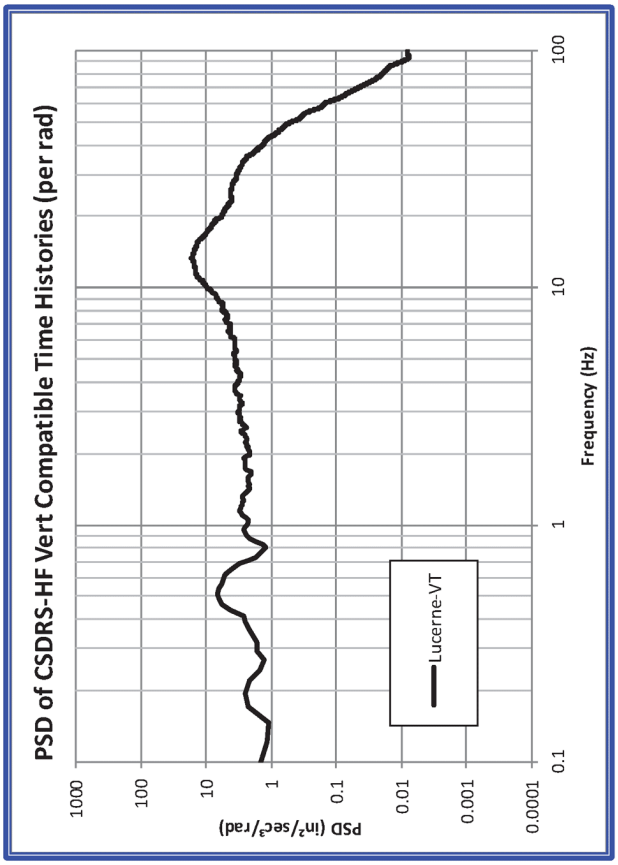


Figure 3.7.1-14: Average Response Spectra Vertical

Figure 3.7.1-15: Power Spectral Density Curves Certified Seismic Design Response Spectra Compatible Time Histories







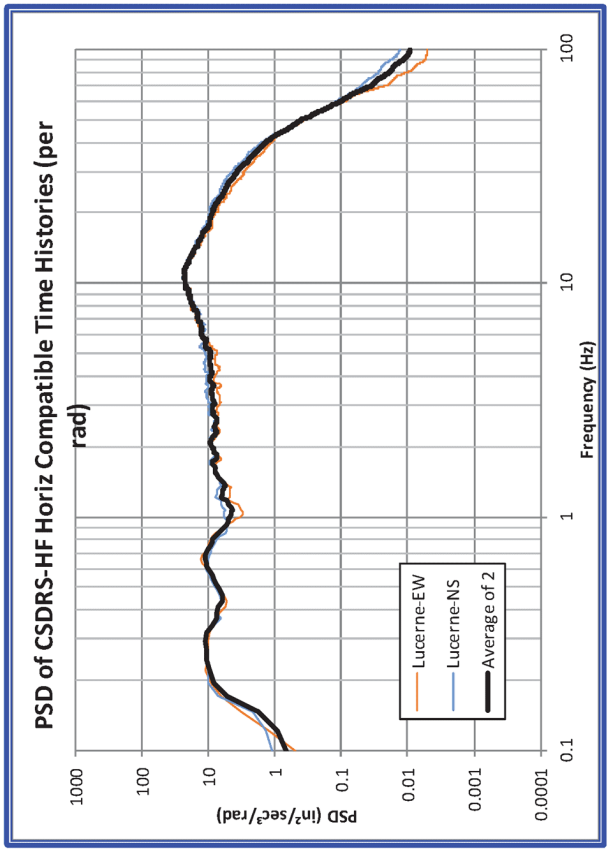


Figure 3.7.1-16: Power Spectral Density Curves Certified Seismic Design Response Spectra - High Frequency Compatible Time Histories

Figure 3.7.1-17: Soil Shear Modulus Degradation Curves

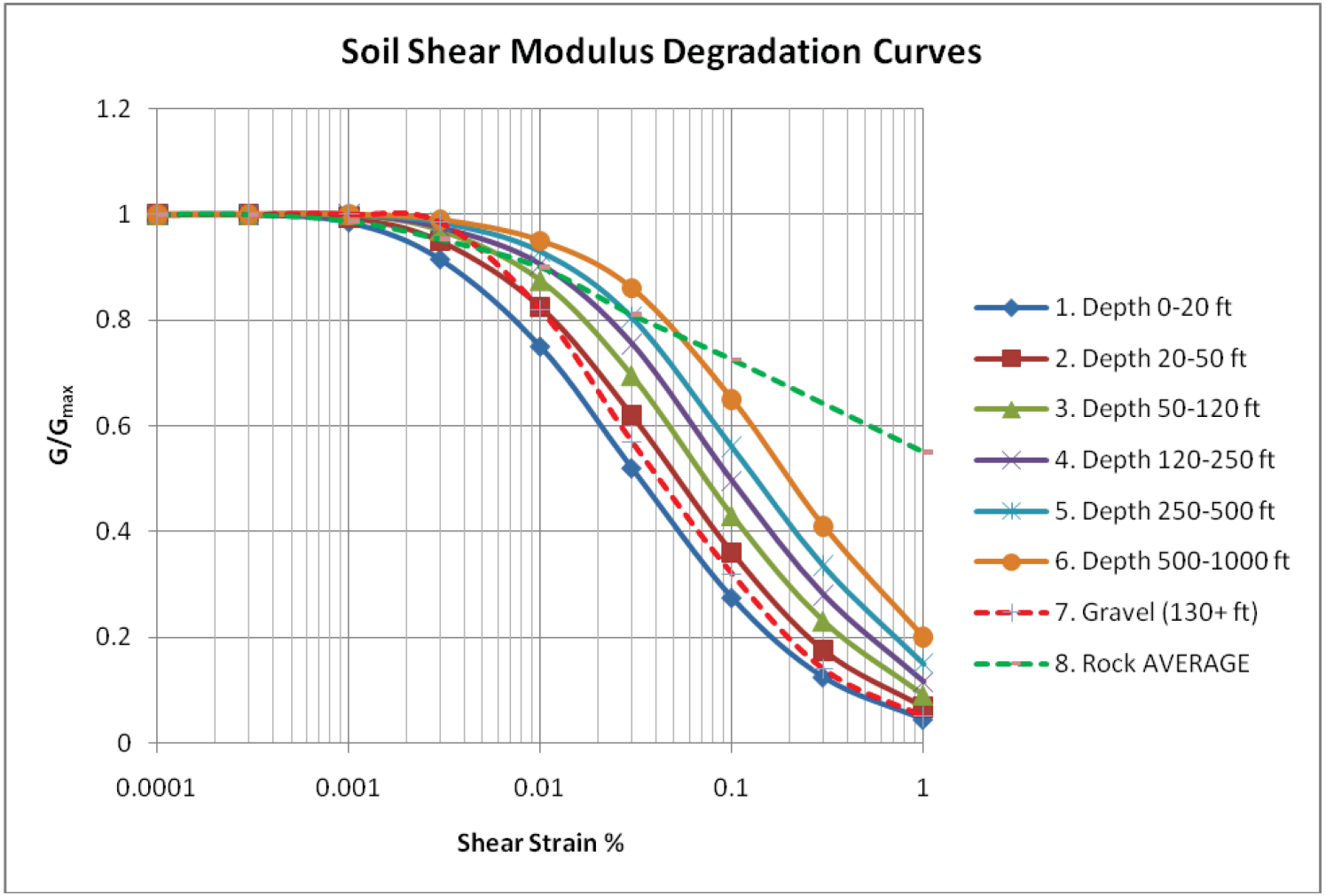


Figure 3.7.1-18: Strain Dependent Soil Damping Curves

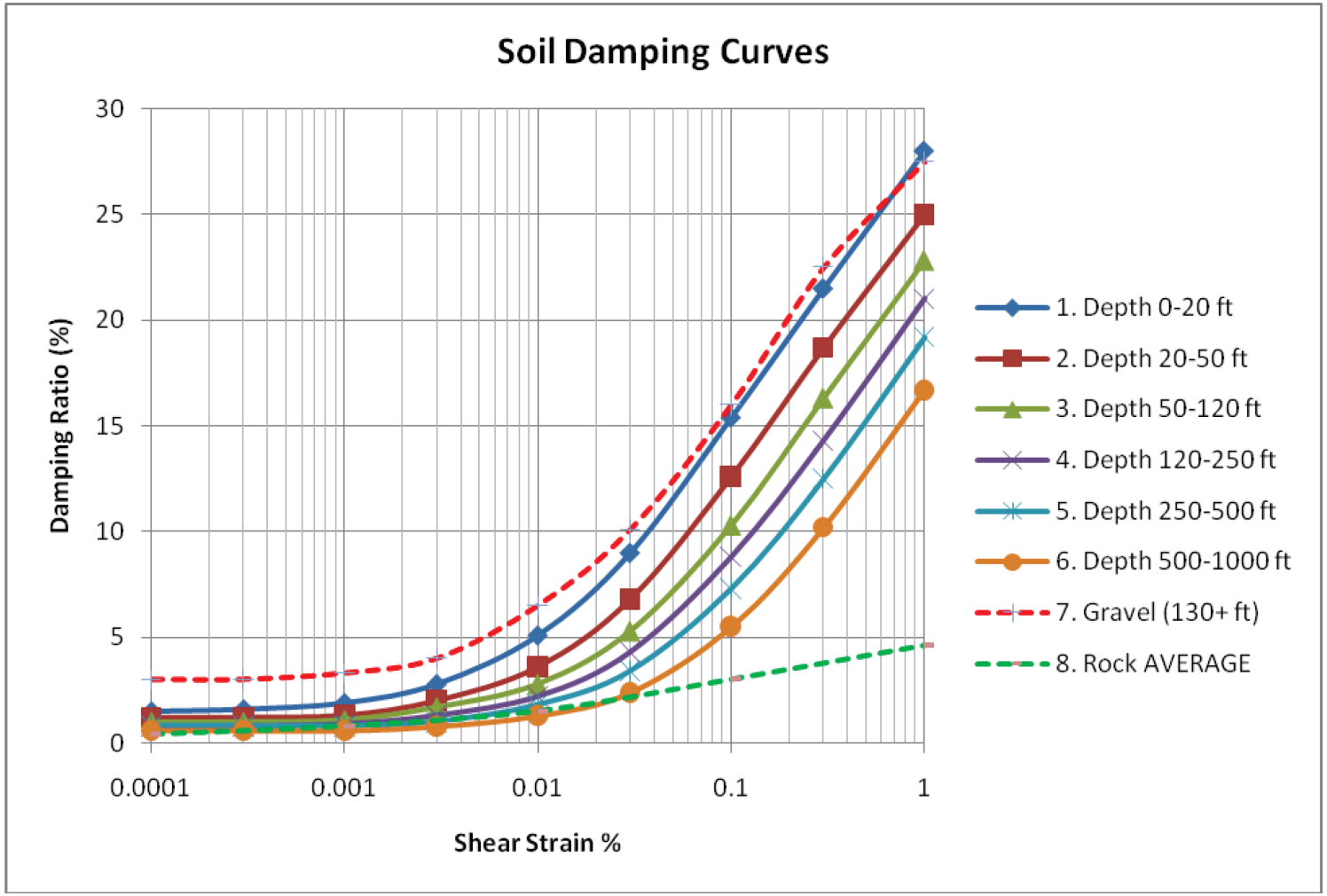


Figure 3.7.1-19: Shear Wave Velocities for All Soil Types

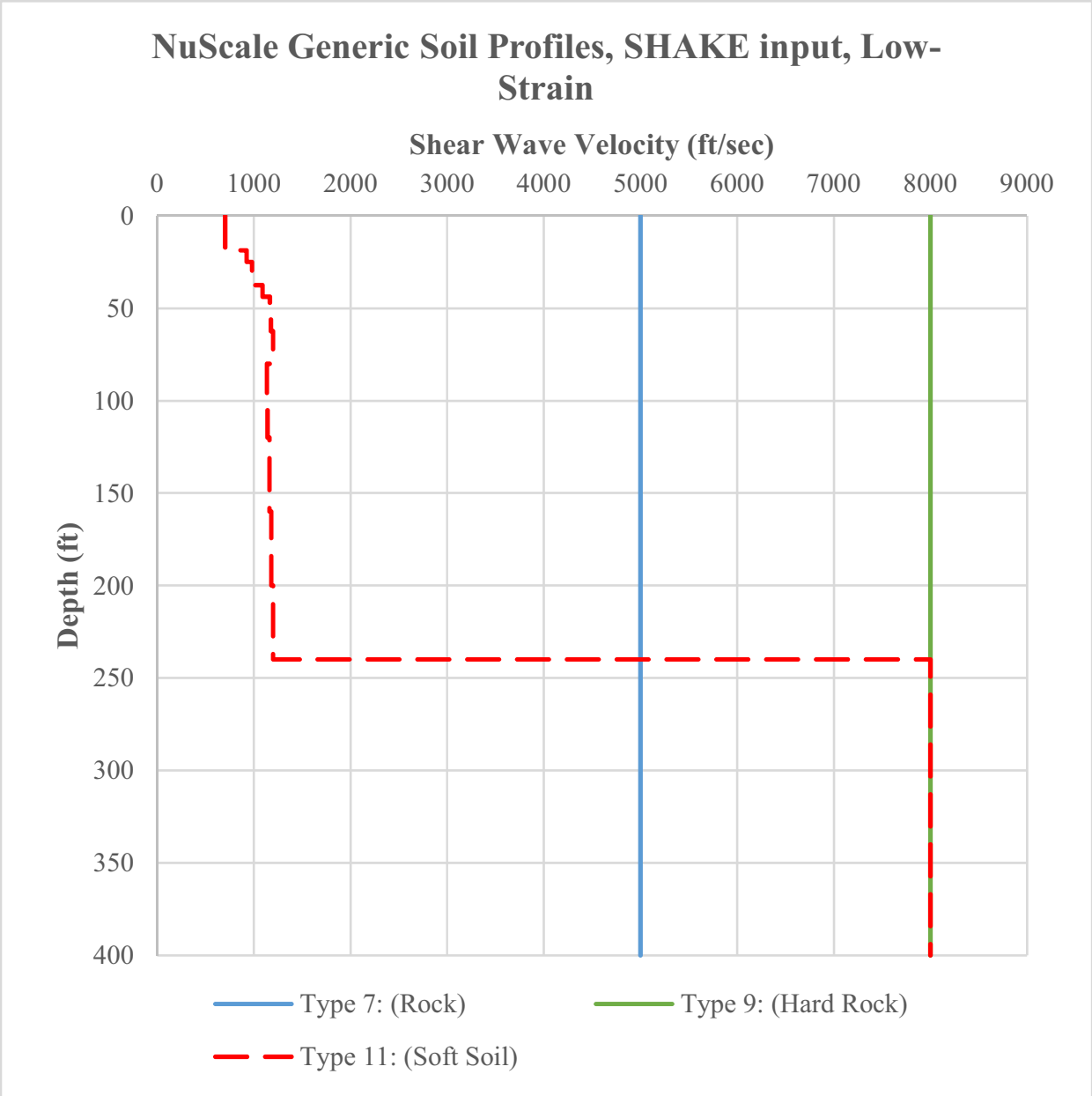


Figure 3.7.1-20: Density for All Soil Types

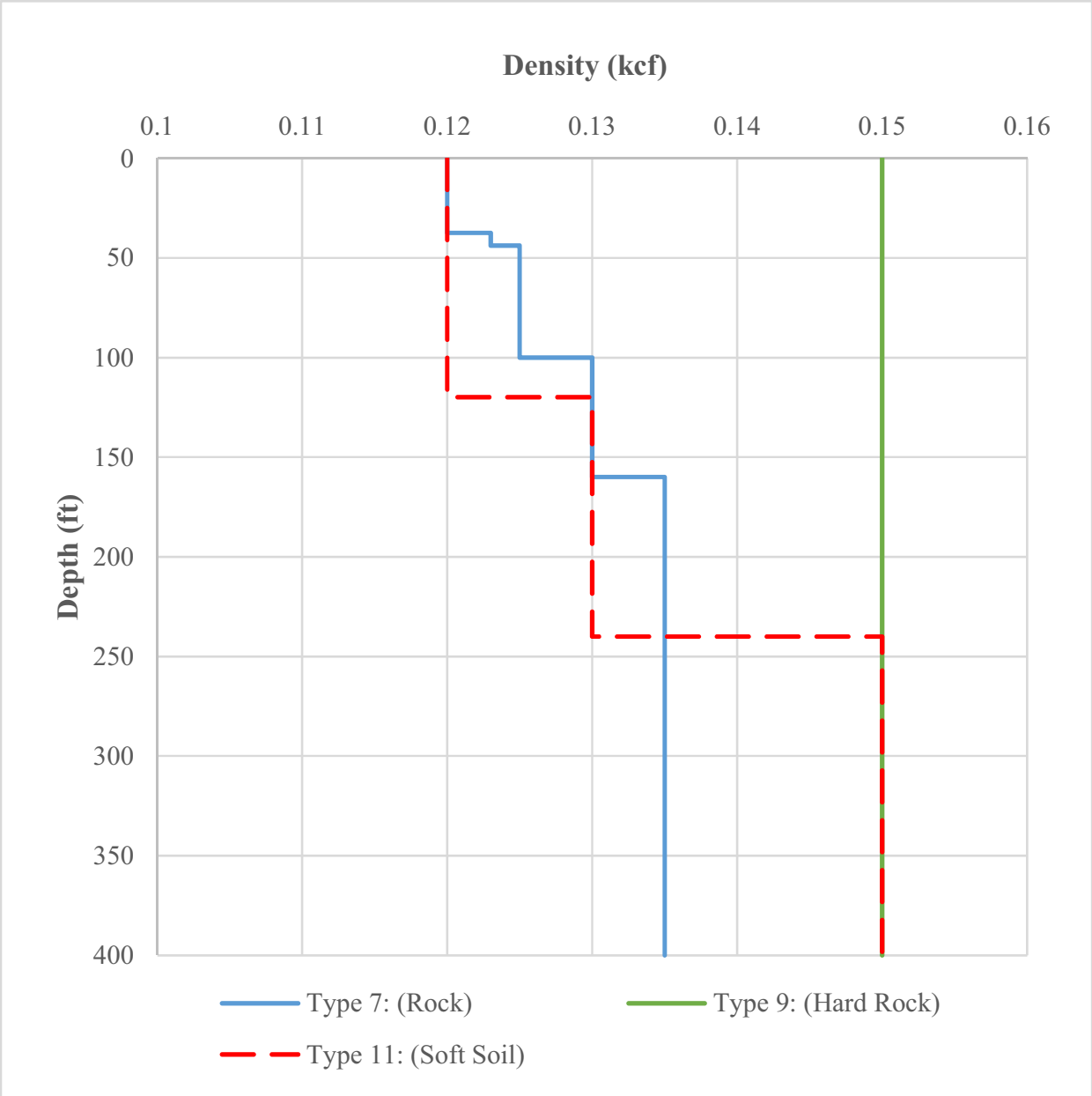


Figure 3.7.1-21: Average Strain Compatible V_s Profiles for Certified Seismic Design Response Spectra Compatible Inputs

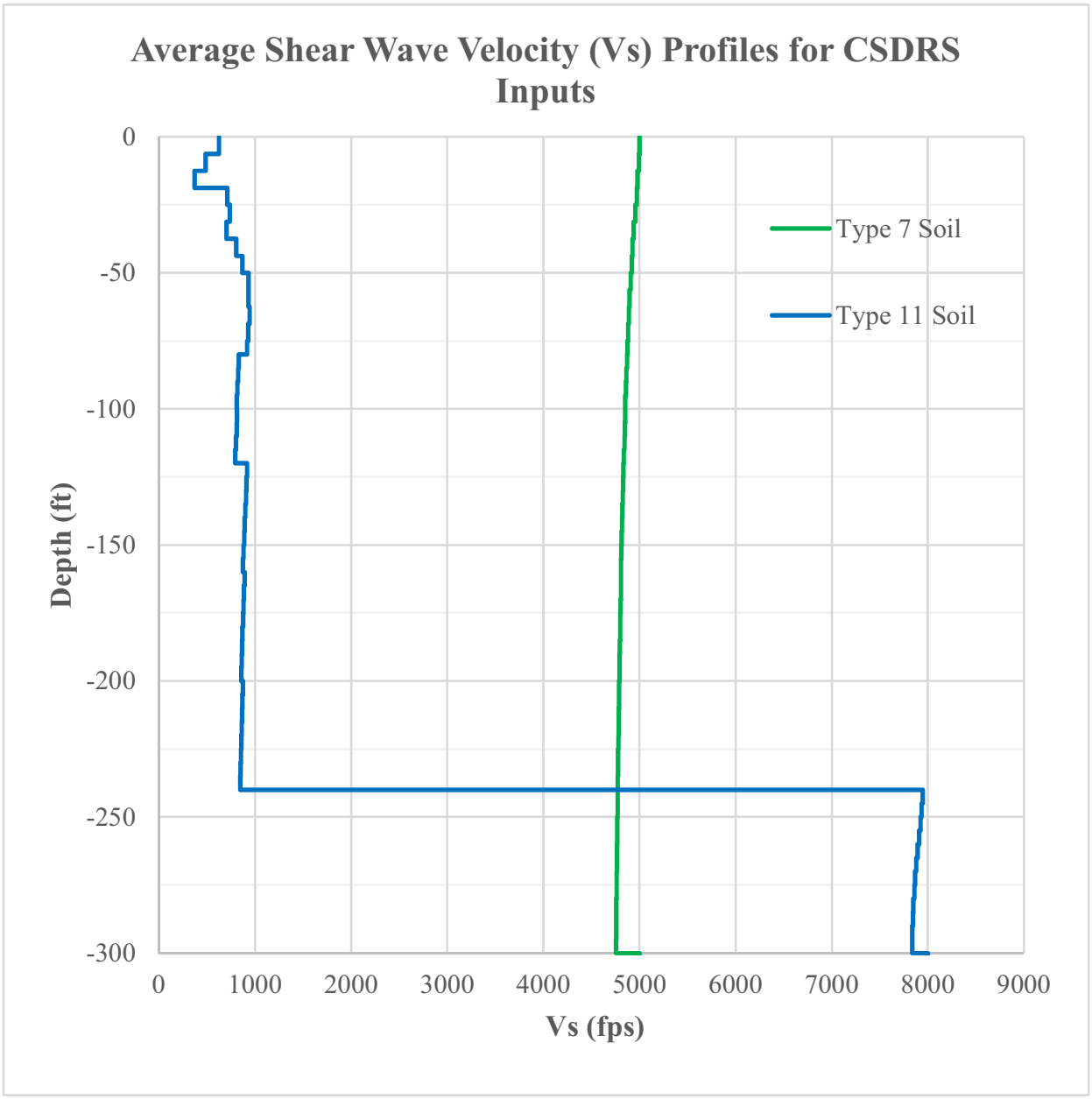
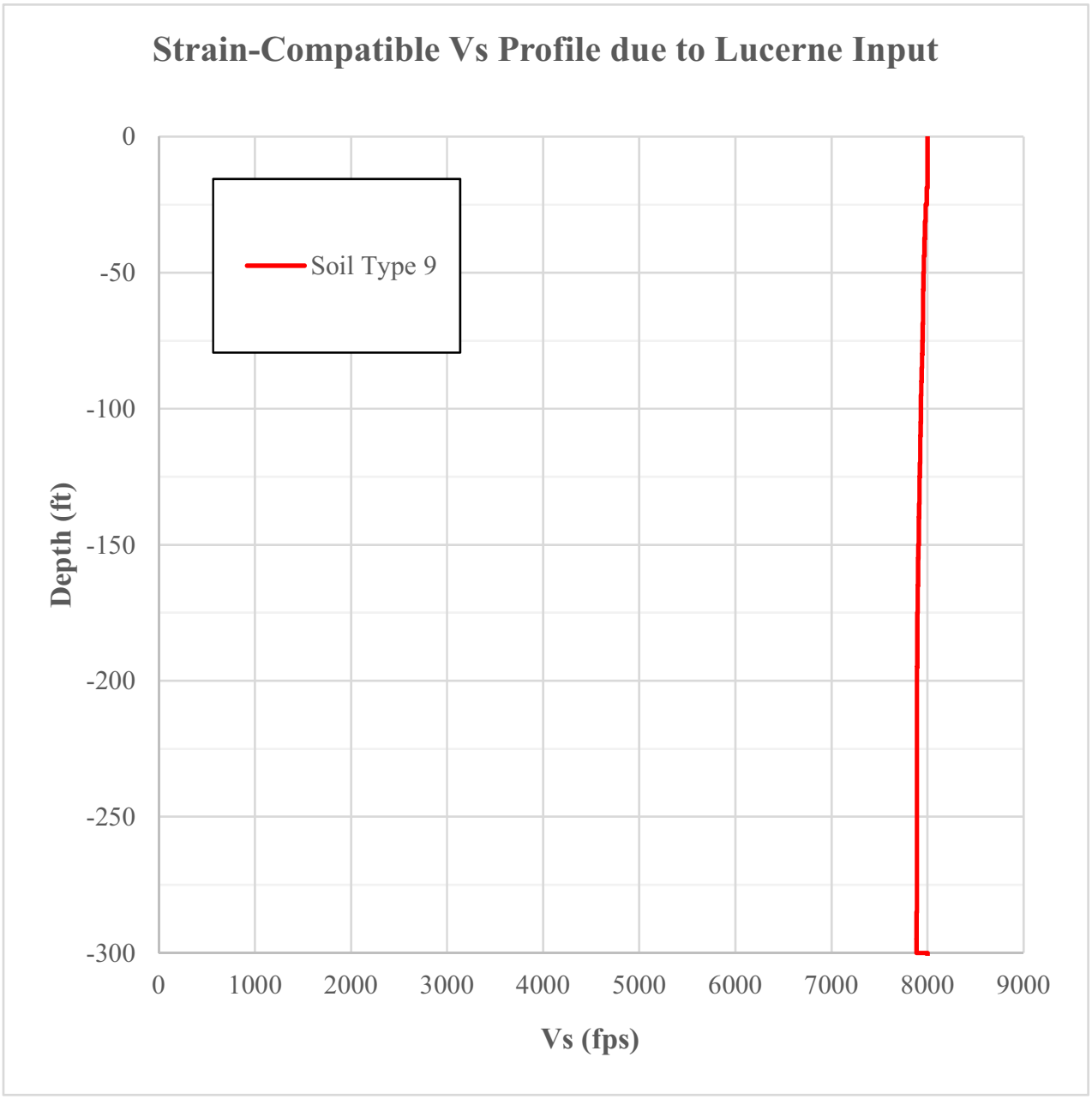


Figure 3.7.1-22: Strain Compatible V_s Profiles for Certified Seismic Design Response Spectra - High Frequency Compatible Input



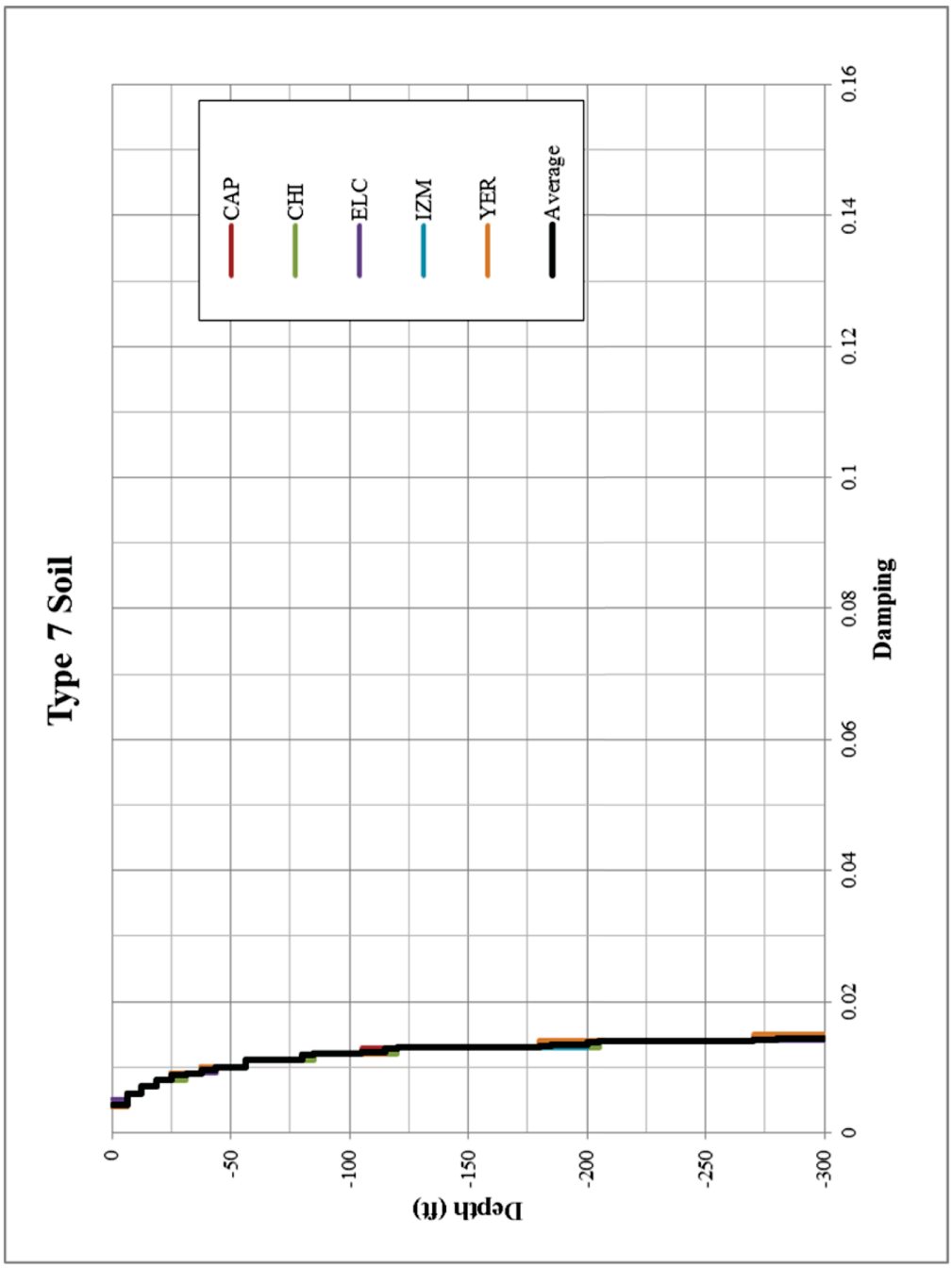


Figure 3.7.1-23: Strain Compatible Damping for Soil Type 7 for Certified Seismic Design Response Spectra Compatible Inputs

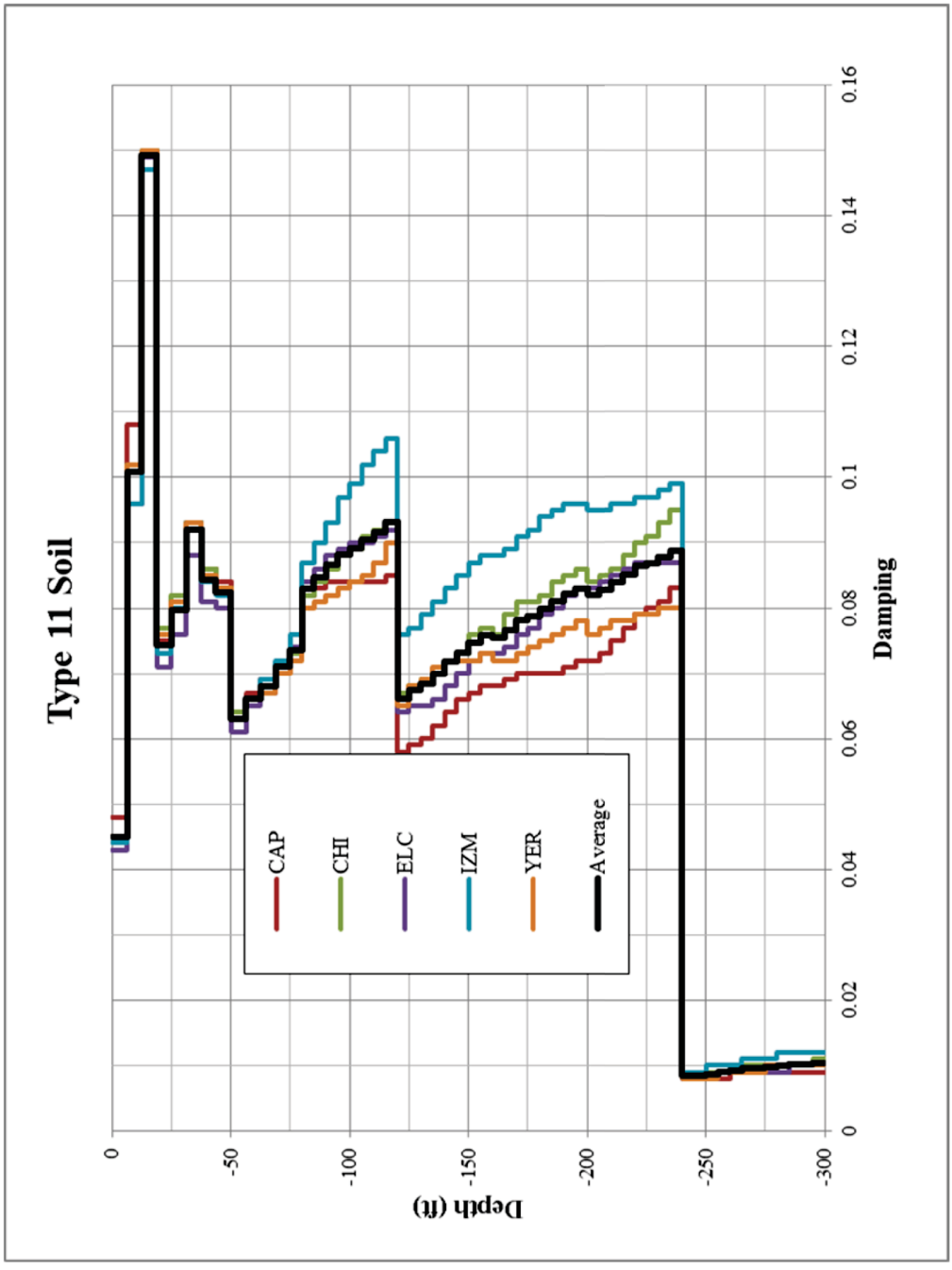


Figure 3.7.1-24: Strain Compatible Damping for Soil Type 11 for Certified Seismic Design Response Spectra Compatible Inputs

Figure 3.7.1-25: Comparison of Average Strain Compatible Damping for Certified Seismic Design Response Spectra Compatible Inputs

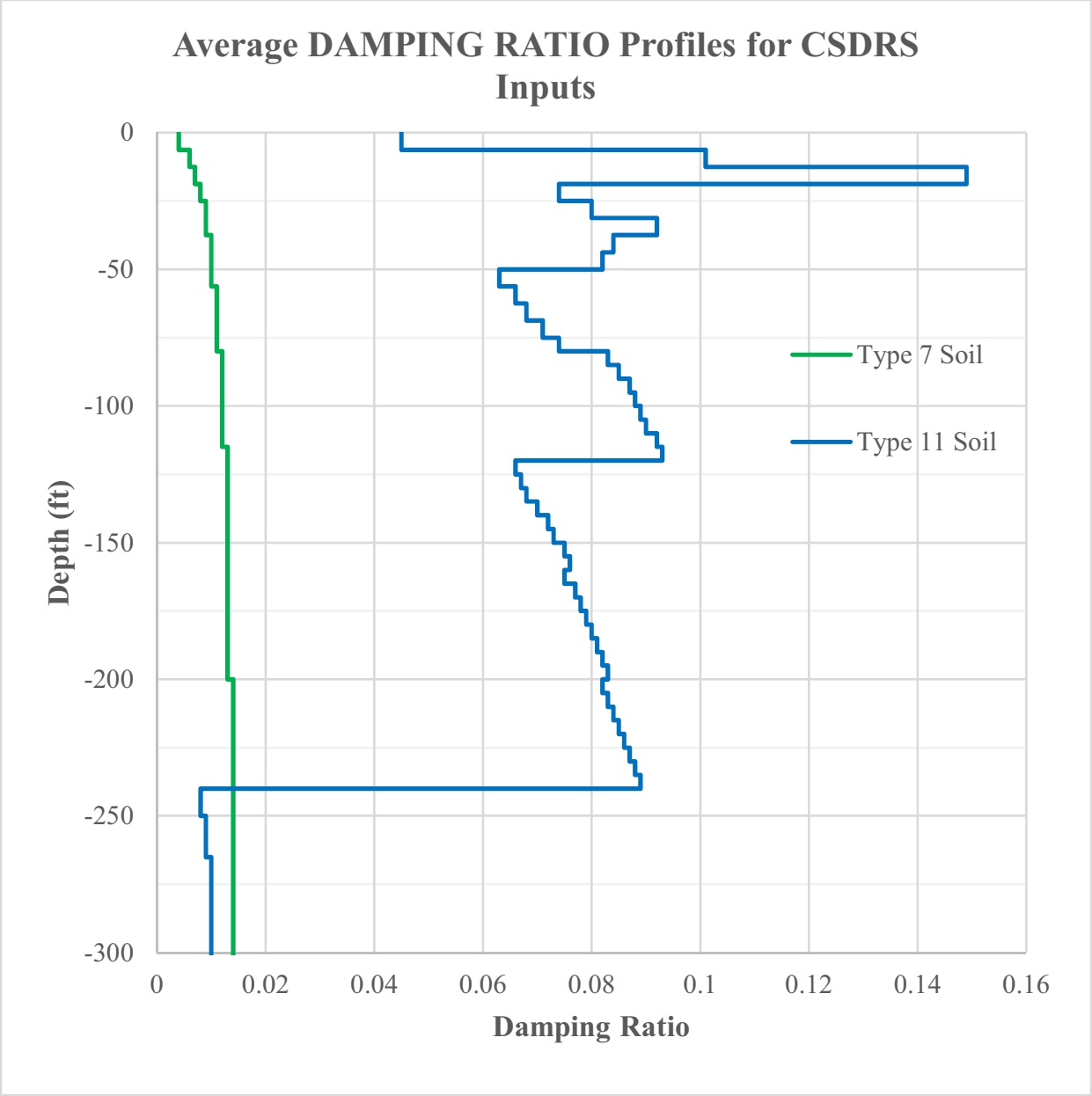
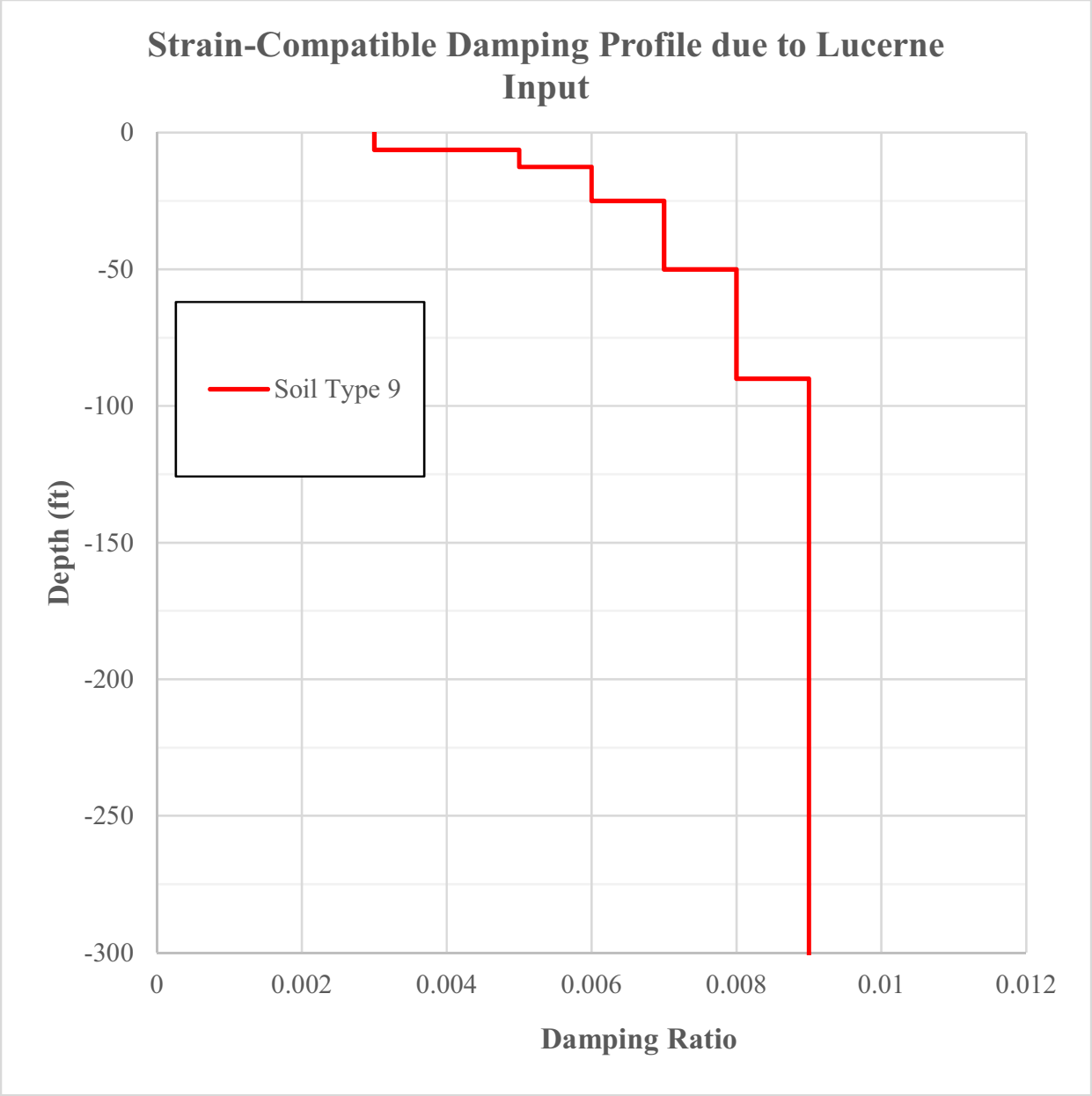


Figure 3.7.1-26: Comparison of Strain Compatible Damping for Certified Seismic Design Response Spectra - High Frequency Compatible Input



3.7.2 Seismic System Analysis

There are two site-independent Seismic Category I (SC-I) structures, portions of the Reactor Building (RXB) and portions of the Control Building (CRB). The RXB is designed for up to six installed NuScale Power Modules (NPMs). The structural analysis is performed with six modules in place. Section 3.7.2.10 provides discussion about the effect on the structure if a seismic event were to occur during operation with less than the full complement of six NPMs.

The above grade portion of the Radioactive Waste Building (RWB) is classified as nonsafety-related, Seismic Category III (SC-III). Below grade, and a small portion above grade of the RWB is classified as RW-IIa depending on the hazardous waste levels stored and processed within the building 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." Section 3.7.2.8 discusses the interaction of the SC-III RWB with the Seismic Category I RXB.

Figure 3.7.2-60 shows the RXB model. Figure 3.7.2-4 shows the CRB model.

3.7.2.1 Seismic Analysis Methods

The seismic analysis of SC-I structures, systems, and components (SSC) uses linear equivalent static analysis, linear dynamic analysis, complex frequency response methods, or nonlinear analysis. The SSC are designed to withstand the effects of the safe shutdown earthquake (SSE) and remain functional in accordance with RG 1.29, "Seismic Design Classification for Nuclear Power Plants." The SC-I aspects of the RXB and the CRB are primarily analyzed using the time-history method, and supplemented with additional analyses described in the following sections. Soil libraries as described in Reference 3.7.2-1, are developed to represent the excavated half space, are introduced in the harmonic analyses as super elements. Together with the seismic load vectors that come with the soil libraries, the soil structure interaction is addressed using the methodology described in Reference 3.7.2-1.

3.7.2.1.1 Computer Programs

Commercially available computer programs ANSYS (Reference 3.7.2-4), SDE SASSI (Reference 3.7.2-2), and ACS SASSI (Reference 3.7.2-3) are used for analysis and development of soil libraries. Software used for performing seismic analyses of SC-I structures, systems, and components conforms with the requirements for computer software as per the NuScale Quality Assurance Program Description (QAPD) (Reference 3.7.2-15). Additionally, Python (Reference 3.7.2-12 and Reference 3.7.2-13) and MATLAB (Reference 3.7.2-14) are utilized throughout for repetitive calculations and presentation of results.

3.7.2.1.1.1 ANSYS

Models are developed using ANSYS and include reinforced concrete (RC) slabs and walls as well as steel-plate composite (SC) walls. A description of the finite element models' characteristics, material properties, and analysis results is provided in subsequent sections.

3.7.2.1.1.2 SASSI

The seismic soil-structure interaction (SSI) analyses are performed using the methodology in the topical report, Improvements in Frequency Domain Soil-Structure-Fluid Interaction Analysis, TR-0118-58005-P-A-R2 (Reference 3.7.2-1). In this methodology, SASSI is used to model the excavated soil around and under the buildings. The SASSI model creates the frequency-dependent soil impedance libraries and seismic load vectors that are imported into ANSYS to be combined with the building models to perform the SSI analyses.

Two SASSI models are used to form the soil impedance libraries and seismic load vectors. The first is a double building (DB) model of the RXB and RWB used to include structure-soil-structure interaction effects. The SDE-SASSI program is used to model the excavated soil around the RXB and RWB.

The second is a single building model of the CRB to evaluate SSI effects. The ACS SASSI program is used to model the soil under the CRB.

In both models, the direct method is used to form the impedance matrices for the soil libraries.

3.7.2.1.1.2.1 Model Dimensions

The soil impedance libraries include soil layers to 300 ft below grade level. Section 3.7.1 describes the parameters (shear wave velocity, density and Poisson's ratio) of the generic soil profiles. Below 300 ft, the parameters remain constant. The variable depth method of SASSI is used to add soil layers to simulate a semi-infinite half space at the bottom of the soil layer base.

In the vertical direction, the finite element model of the RXB and RWB extends to the bottom of the foundation. In the horizontal direction, the finite element model of each building is extended out about 25 ft around the perimeter of the building to model the backfill soil. For the SSI analyses, the soil impedance libraries and seismic load vectors are applied at the boundaries of the finite element model.

The CRB is modeled as a surface-founded structure, ignoring the 5-ft embedment of the foundation; thus, there is no backfill around the CRB model.

The use of the soil impedance libraries in the SSI analysis means that the free-field boundaries extend to the elastic half-space implicitly. However, for the static analyses performed to obtain results due to static loads and differential displacements, the free-field soil is modeled explicitly beyond the backfill soil boundaries. For the double building RXB-RWB static model, the overall length is 2005.5 ft, the width is 768.5 ft and the depth is 360 ft. The vertical depth is deeper than the SSI model. At this depth, the vertical displacement becomes insignificant due to soil stiffness. The horizontal free-field boundaries are extended far enough away so that fixing them has a negligible effect on the static response of the buildings. Similarly, for the CRB static model, the overall length is 1319.7 ft, the width is 1351.1 ft, and the depth is 300 ft.

3.7.2.1.1.2.2 Cut-Off Frequency

For the frequency-domain SSI analysis with Soil Types 7 and 9 the cut-off frequency is established at 100 Hz, the maximum frequency that can be analyzed with a time step of 0.005 seconds.

For Soil Type 11 the cut-off frequency is established at 35.2 Hz. This value is higher than the wave passing frequency of Soil Type 11 calculated as minimum (Vs/(5T)) = 371 fps/(5x6.25') = 12 Hz.

The building models have element sizes that are less than the 6.25 ft layers that are used to determine the wave passing frequency of the soil. Therefore, the wave passing frequencies of the buildings are greater than the wave passing frequencies of the soil.

3.7.2.1.1.2.3 Cracked Model Stiffness

The Building Design Analysis Methodology Topical Report TR-0920-71621-P-A (Reference 3.7.2-9) describes the methodology for modeling the stiffness of cracked concrete elements.

3.7.2.1.2 Finite Element Models

The finite element model of the RXB, crane, and pool water is developed, in general, using solid shell (SOLSH190), shell (SHELL181), beam (BEAM188), and fluid (FLUID30) elements. Surface (SURF154) and mass elements (MASS21) are used to apply the appropriate distributed and concentrated masses to the structure. The finite element models of the CRB and RWB are similar to the RXB except that models use structural shell elements to represent the slabs and walls instead of solid shell elements.

Section 3.7.2.1.2.6 outlines the cracking analysis used to develop the hybrid cracked/uncracked double building models. The hybrid CRB models are also developed using this procedure.

The MATLAB program is used to generate the preliminary finite element mesh. Microsoft Excel is used for loading calculations and to organize ANSYS parametric design language (APDL) scripts for ANSYS model development. The APDL scripts are used to create and modify the geometry and define the building material properties and loads.

3.7.2.1.2.1 Reactor Building

The RXB houses equipment for operating NPMs. The RXB provides anchorages and support for various SSC. The overall dimensions of the building are 231.5 ft, 155.5 ft (185.5 ft with penetration shrouds), 171 ft in east-west, north-south, and vertical directions, respectively. The grade level for the RXB is considered to be at elevation 100 ft.

Section 1.2 provides additional discussion of the RXB and Figure 3.7.2-61 through Figure 3.7.2-86 provide elevation and section views of the building.

The predominant feature of the RXB is the ultimate heat sink (UHS). The UHS includes the spent fuel pool, refueling area pool, and the reactor pool. The dry dock is also assumed to be full of water and part of the UHS for the seismic analysis. This large pool occupies the center of the RXB. Although bay walls extend to the bioshields at elevation 123 ft, the nominal top of the pool is at elevation 79 ft. The normal reactor pool water depth is maintained at 53 ft.

The NPMs and the water in the pool contribute a large amount of weight to the global mass of the RXB and thus are modeled explicitly.

The typical thickness for the main structural interior and exterior SC walls is 4 ft; the primary floor slabs are 2 or 3-ft-thick reinforced concrete. The east and west exterior SC walls are 5 ft thick while the north and south exterior walls are 4 ft thick. The basemat foundation thickness is 8 ft. The foundation top of concrete elevation is 25 ft. The foundation for the reactor pool area and spent fuel pool area is raised and has an elevation of 26 ft at the top of the liner and a corresponding thickness of 9 ft. The roof slab thickness is 3 ft and the top of roof elevation is 187.5 ft (Figure 3.7.2-71).

Reactor Building ANSYS Model

Figure 3.7.2-60 shows the concrete structure of the RXB. The building structure consists of an 8 ft-thick RC basemat. The basemat is 9 ft-thick in the pool region. The building models are developed to include different basemat thicknesses. Floor slabs are 24-36 inches thick. The floor slab at elevation 100 ft includes 3 ft-deep T-beams spanning in the east-west direction. The roof is a composite section comprised of a 36 inch-thick concrete slab and steel girders.

The RXB also includes SC walls with various thicknesses in the east-west and north-south directions. With minor exceptions, the overall thickness of

the SC walls ranges from 4 ft to 5 ft with a constant steel faceplate thickness of 0.75 inch. Descriptions of the SC walls and corresponding material property definitions are presented in subsequent sections.

The RXB structure also includes multiple penetration shrouds on the north and south faces of the exterior walls as shown in Figure 3.7.2-87. Concrete elements are modeled using solid shell (SOLSH190) and shell (SHELL181) elements.

Concrete slabs are divided into three categories:

- 8 ft-thick basemat and 2 to 3-ft-thick main floor slabs. These slabs are modeled using SOLSH190 elements, as it is crucial to achieve correct geometric representation in the pool region. Only the mass of the steel beams, where present, are included for non-composite slabs. Equivalent material properties considering the stiffness of the steel beams under composite portions of elevations 55 ft, 70 ft, and 126 ft slabs are incorporated.
- 6-inch-thick floor slabs. These slabs are not explicitly modeled. Instead, different types of loads on such floors are applied on the walls based on the assumption of a one-way slab.
- 3-ft-thick roof slab modeled using shell (SHELL181) elements. Equivalent material properties considering the stiffness of the steel beams are calculated and applied to the shell elements.

The model has an overall mesh size of approximately 4 ft.

Figure 3.7.2-61 presents three cross section views of the RXB along XZ, YZ, and XY planes. Figure 3.7.2-61 illustrates building structural elements divided into three categories: walls in east-west direction, walls in north-south direction, and horizontal structural components such as basemat, floors, and roof.

Figure 3.7.2-62 through Figure 3.7.2-86 show RXB structural components including basemat, floors, roof, and SC walls on major gridlines. Each component is shown in plan or elevation view. For clarity, some plots include an additional single layer of elements attached to the plotted component in order to mark the boundaries of the component and its interface with other structural members.

3.7.2.1.2.2 NuScale Power Modules

The NPMs are partially immersed in the reactor pool. The NPMs are not permanently bolted or welded to the pool floor or walls, however, there are constraints to keep each NPM in place before, during, and after a seismic event.

The base support is a steel skirt restraint that consists of four built-up stainless steel members bracing the NPM skirt in the lateral directions and

an annular bearing plate that supports the NPM in the vertical direction. The other three geometrical supports are steel lug restraints located on the walls of each bay near the top of the module. The NPM has lugs that align with a slot in the restraint. Each restraint prevents movement in the direction parallel to the wall and allows the NPM to move freely in the upward direction. The lug restraint provides only horizontal restraint in the in-plane direction of the supporting wall.

Figure 3.7.2-7 shows the top view of a restrained NPM.

Figure 3.7.2-5 shows NPM placement within the RXB model.

Figure 3.7.2-6 shows an enlarged view of the NPM and surrounding pool region.

Detailed NuScale Power Module Model Included in the Reactor Building ANSYS Model

To model the NPMs to be included in the RXB ANSYS model, a simplified finite element model of the containment vessel (CNV) and the associated water elements representing the water within each bay (effectively a rectangle of water elements around the NPM) is developed.

The finite element models for these five other NPM components are already developed and are converted to superelements for incorporation into the NPM model:

- top support structure basic model
- reactor pressure vessel basic model
- control rod drive mechanism basic model
- upper reactor vessel internals basic model
- lower reactor vessel internals basic model

Figure 3.7.2-6 shows the CNV model, the pool water elements, and the superelements for one NPM.

Key components of the NPM are described in TR-121515 (Reference 3.7.2-16).

Piping and valves (except for the main steam and feedwater piping and valves), manways, instruments, pressurizer heaters, and other small internal components such as bolts are not explicitly modeled. These features do not affect the gross structural behavior of the model and removing them allows for simplified meshing techniques to be used. The piping is relatively flexible when compared to the vessels, and it does not drive the response of the CNV or reactor pressure vessel.

3.7.2.1.2.3 Reactor Building Crane

The Reactor Building crane (RBC) is a bridge crane used to transport modules between the operating locations and the refueling and disassembly area and the dry dock. The RBC travels on rails on the top of the reactor pool walls at elevation 145 ft. When not in use, the RBC is parked over the refueling pool. Section 9.1 provides additional RBC description.

The Reactor Building crane ANSYS model is coupled to the RXB model at 32 nodes using constraint equations. These constraints are applied in the vertical direction (UZ degree-of-freedom), with just two of the east nodes coupled in the east-west direction (UX degree-of-freedom) and two of the south nodes coupled in the north-south direction (UY degree-of-freedom).

Figure 3.7.2-8 shows the RBC model. For the RXB analysis, the RBC does not hold the NPM load and the crane is located in the western side of the reactor pool area as shown in Figure 3.7.2-9. This configuration, generates a larger response in the building. The RXB analysis generates in-structure response spectra (ISRS) that are then used as input to the RBC seismic analysis.

3.7.2.1.2.4 Ultimate Heat Sink Pool

The UHS pool contributes a large amount of weight to the global mass of the RXB. This fluid mass impacts the dynamic characteristics of the building. Figure 3.7.2-10 provides a visualization of the hydrodynamic pressure field in the pool and on the CNV.

Water mass regions are modeled by fluid finite elements. This fluid element is defined by eight nodes having three translational degrees of freedom at each node plus a pressure degree of freedom. This element is used to model fluids contained within vessels having no net flow rate and is well suited for calculating hydrostatic pressures and fluid-solid interactions.

3.7.2.1.2.5 Control Building

Section 1.2 describes CRB features and components. Figure 3.7.2-11 through Figure 3.7.2-16 show plan and elevation views of the CRB model.

The CRB is an RC structure comprised of an SC-I portion and an SC-II portion. The main control room is located in the SC-I portion while the Technical Support Center is located in the SC-II portion. The SC-I portion of the CRB is the focus of this section. It has overall dimensions of approximately 120 ft, 55 ft, and 50 ft in the east-west (X), north-south (Y), and vertical (Z) directions, respectively.

The SC-I portion of the CRB consists of a 5-ft-thick basemat and 3-ft-thick exterior walls. The interior walls at gridlines 3 and G are 3 ft and 2 ft thick,

respectively. The floors are 2 ft thick. Concrete elements are modeled using shell elements. An overall mesh size of 16 inches is used for modeling.

The partition walls are not modeled explicitly, however, their mass is included as a distributed mass on the slabs.

The CRB is embedded 5 ft so it is modeled as a surface founded structure. The ACS SASSI program requires that an excavated soil region be defined in order to form the soil library, so two layers of soil are excavated and then reinserted for the CRB to sit on. The basemat and soil elements are then joined using target/contact elements with options set to ensure a bonded contact.

With these target/contact pairs, the elements comprising the soil are allowed to be coarser than the CRB elements. Material properties are assigned to the soil matching the upper layer properties for each of the three soil types under consideration. Thus, three CRB models with uncracked concrete properties are built: one each with Soil-7, 9, and 11 properties. Figure 3.7.2-4 shows one of the three CRB models.

The cracking analysis and hybrid CRB models are developed following the same procedure outlined in Section 3.7.2.1.2.6 for the double-building. The only difference is the CRB is comprised entirely of RC members; thus, consideration for SC members is not required.

3.7.2.1.2.6 Double-Building Model

The standalone RXB and RWB models are combined with a backfill model that surrounds the building models and ensures mesh conformity. The combined model, named the DB model, fills the excavation volume that is used in building the soil libraries.

The DB model interacts with the soil library through an interface that maps the contribution of the latter on the DB. The soil library is built with a mesh coarser than the one used in the DB. This incompatibility is addressed through a system made of spring, shell, and contact elements placed on the excavation volume exterior surfaces. Figure 3.7.2-1 shows a representation of the SSI interface model. In the figure, the blue node represents the master node of the soil library substructure.

Because the soil libraries are built for an excavation volume that exceeds the size of the volume to be filled with the engineered backfill, the material models of the solid elements inside the DB model that are not supposed to be filled with engineered backfill are given the properties of the free-field soil that is evaluated.

The NPM and RBC models are imported into the DB model. These models are simplified versions of their more detailed versions and are used in the DB model to address their contribution in the analyses performed and to

extract relevant data to be used in detailed analyses with more accurate models. The details of the simplified NPM model and the process followed to import them into RXB are explained in TR-121515 (Reference 3.7.2-16).

The DB model to be used for the soil-separation design-basis case is built by reducing the stiffness of the engineered backfill in the top 25 ft below grade by 99 percent. This model is built using the DB model that is compatible with the Soil-7 library.

This process results in four DB models with uncracked concrete properties. These are the DB models that are compatible with Soil-7, 9, and 11 soil libraries and the soil separation model that is compatible with the Soil-7 soil library.

Cracking Analysis

The cracking analysis is performed by first extracting the peak element forces from the seismic analyses of the uncracked DB models; then the structural members that are identified as cracked are updated by changing their material properties to represent the cracked concrete as outlined in Section 4.0 of TR-0920-71621-P-A (Reference 3.7.2-9) to form the hybrid cracked/uncracked models.

Hybrid Double-Building Models

The DB models that are made of both cracked and uncracked structural members are grouped into ISRS and design categories.

As outlined in TR-0920-71621-P-A (Reference 3.7.2-9), in developing the hybrid DB models for ISRS calculation, the damping values of the cracked RC and SC members are specified as 7 percent and 5 percent respectively. For the uncracked RC and SC members, the damping values are set to 4 percent and 3 percent respectively. For design calculations, the damping values for uncracked RC and SC members are set to be same as the cracked ones (5 percent for SC and 7 percent for RC members). Thus, two different hybrid models are generated per DB model: one for ISRS calculation and the other for design calculations. The difference between the two hybrid models is the damping value assigned to uncracked members.

3.7.2.2 Natural Frequencies and Responses

Modal analysis is performed by calculating the first 1957 modes of the DB model with uncracked concrete properties. Table 3.7.2-1 through Table 3.7.2-3 list the first five modes with the highest mass participation ratios in X, Y, and Z directions. Figure 3.7.2-17 shows the RXB modal mass participation ratios. Figure 3.7.2-18a through Figure 3.7.2-18c show the major RXB modes in the X, Y, and Z directions. These modes are from a modal analysis of the uncracked RXB model with a fixed base and without the crane, NPMs, pool water, and backfill. Nodal

displacement contour plots are provided for each mode shape. For clarity, only RXB structural elements (SOLSH190 and SHELL181) are shown.

For the SC-I CRB, modal analyses are performed by calculating the first 350 modes. Table 3.7.2-4 through Table 3.7.2-6 list the first five modes with the highest mass participation ratios in X, Y, and Z directions. Figure 3.7.2-107 through Figure 3.7.2-109 show the CRB major modes in X, Y, and Z directions. Figure 3.7.2-19 shows the CRB modal mass participation ratios. The major frequencies in X and Y directions are 16.17 Hz and 11.34 Hz, respectively. The cumulative mass participation does not exceed 65 percent for any X, Y, or Z direction because fixed boundary conditions are applied to the basemat with its large amount of mass.

3.7.2.3 Procedures Used for Analysis Modeling

The general approach for the structural analysis is:

1) Create building models for the double-building and Control Building with major equipment in ANSYS.

   a) Develop the NPM model for the RXB.

   b) Develop the Reactor Building crane model for the RXB.

   c) Incorporate the NPMs and Reactor Building crane into the RXB and combine with the Radioactive Waste Building to form the double-building.

   d) Perform cracking analysis of the double-building and Control Building models.

   e) Thermal accident model of RXB is generated after cracking.

2) Perform multiple runs of ANSYS using the different combinations of the certified seismic design response spectra (CSDRS) and certified seismic design response spectra - high frequency (CSDRS-HF) (discussed in Section 3.7.1.1), soil profiles (discussed in Section 3.7.1.3), building models and material damping values (discussed in Section 3.7.1.2).

3) Perform static analyses with uncracked models.

4) Combine the results to create bounding design values.

3.7.2.4 Soil-Structure Interaction

Soil-structure interaction analysis follows the methodology in TR-0118-58005-P-A-R2 (Reference 3.7.2-1). In this methodology, the SSI is performed in the frequency domain using a multi-step approach. For each soil type, soil impedances and seismic load vectors are calculated using SASSI to form a soil library. These soil impedances and seismic load vectors are then

imported into an ANSYS model for the SSI analysis. If applicable, the ANSYS model contains fluid elements for modeling fluid-structure interaction.

The CSDRS, CSDRS-HF, and associated time histories sets are developed in Section 3.7.1.1. The soil types are developed in Section 3.7.1.3.

Figure 3.7.2-2 shows the SASSI model of the excavated soil for the DB model. The deepest portion, under the RXB, consists of 12 layers. Each layer is 7 ft thick, except layer 3 is 6 ft thick and layer 12 is 8 ft thick.

Figure 3.7.2-3 shows the SASSI model of the excavated soil under the CRB.The excavated soil model consists of two 75-inch-thick layers. Because the CRB is modeled as surface founded, a soil model with identical properties and meshing as the excavated soil is put back into the excavation.

Figure 3.7.2-4 shows the combined CRB plus soil model. The non-SC-I portion of the CRB is not included in the model, and the soil region is enlarged to include the surcharge effects of the non-SC-I portion.

3.7.2.4.1 Methodology for Combining Seismic Response Results

Seismic responses are obtained due to input motion in each global direction. The combined seismic responses are obtained by algebraic summation of the responses due to each particular input motion direction, as shown below.

$$Out_j(t) = (\sum\nolimits_{u=X,Y,Z}) Out_{j,u}(t) \qquad \text{Eq. 3.7-1}$$

where, $Out_{j,u}(t)$ is the time domain response due to input motion in u direction, and $Out_j(t)$ is the combined time domain response for structural response component j (Table 3.7.2-9).

As an example, the combined acceleration responses in the time domain are obtained as:

$$\begin{bmatrix} Acc_X(t) \\ Acc_Y(t) \\ Acc_Z(t) \end{bmatrix} = \begin{bmatrix} Acc_{X,x}(t) + Acc_{X,y}(t) + Acc_{X,z}(t) \\ Acc_{Y,x}(t) + Acc_{Y,y}(t) + Acc_{Y,z}(t) \\ Acc_{Z,x}(t) + Acc_{Z,y}(t) + Acc_{Z,z}(t) \end{bmatrix} \qquad \text{Eq. 3.7-2}$$

In Equation 3.7-1, the summation is performed at each time step.

Using the combined acceleration responses, ISRS are obtained in the X, Y, and Z directions. The ANSYS time-history post-processor (POST26) command RESP is used to obtain absolute acceleration response spectra for user-specified damping ratios and frequencies. These frequencies range from 0.1 Hz to 100 Hz, at 0.1 Hz intervals.

Section 3.7.2.5.1 describes the post-processing of the ISRS, for example, averaging, enveloping, and broadening.

3.7.2.4.2 Maximum Forces and Moment in Steel-Composite Walls

Element-based results are used to design SC walls. The forces and moments from the SOLSH elements are described in Appendix 3B.1.1. Appendix 3B.1.1 shows the finite element forces obtained in the local element coordinate system.

3.7.2.4.3 Maximum Forces and Moments in Section Cuts

Section cuts are used to design RC members including walls, floor slabs, basemat, and roof. For the section cuts, the forces and moments are the force and moment resultants resolved at the geometric center of the cut plane. Figure 3B-2 shows the force and moment resultants that are obtained in the section cut local coordinate system.

3.7.2.4.4 Relative Displacements at Selected Locations

Relative displacements at selected locations in the RXB and CRB are calculated using the post-processing steps described in Section 3.7.2.4.1 and Section 3.7.2.5.1. The displacements are calculated relative to the top center of the basemat and are in the global directions.

Figure 3.7.2-88 and Figure 3.7.2-89 show some of the selected nodes in the RXB. Table 3.7.2-14a and Table 3.7.2-14b provide the RXB relative displacements.

Figure 3.7.2-98 through Figure 3.7.2-100 show the selected nodes in the CRB. Table 3.7.2-14c and Table 3.7.2-14d provide the CRB relative displacements.

3.7.2.4.5 Design Approach

Design of structural components is based on ACI 349-13 (Reference 3.7.2-10) for RC members and AISC N690-18 (Reference 3.7.2-11) for SC walls. Design methodologies are discussed in TR-0920-71621-P-A (Reference 3.7.2-9).

The design calculations are performed to determine section thickness and reinforcement layout of RC members, and thickness of steel faceplates and size and spacing of tie reinforcement for SC walls.

There are four case-soil type pairs forming the design basis: Baseline-Soil-7, Baseline-Soil-9, Baseline-Soil-11, and Soil-Separation-Soil-7. The Soil-Separation-Soil-7 case applies to the RXB only.

The demand-to-capacity ratio (DCR) values for different design condition failure modes are calculated independently for each seismic input motion. The

resulting DCR values are averaged for the same design-basis case. Then, the averaged DCR values are enveloped for all design-basis cases and load combinations. The initial design properties are adjusted (e.g., by providing additional reinforcement and changing the thickness of steel faceplates) such that the enveloped DCR values stay below the 1.0 limit for each design condition.

The design of Reactor Building and CRB critical sections is provided in Appendix 3B.

3.7.2.5 Development of In-Structure Floor Response Spectra

Development of ISRS follows the guidance in RG 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components.” The ISRS are obtained from SSI analyses of the DB hybrid model and CRB hybrid model for soil types 7, 9, 11, and 7 with soil separation for the DB.

COL Item 3.7-5: An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific analysis that assesses the effects of soil separation. The applicant will confirm that the in-structure response spectra in the soil separation cases are bounded by the in-structure response spectra described in Section 3.7.2.

Six input motions are used for ISRS and relative displacement development. As documented in Section 3.7.1.1, five of the six are compatible with the CSDRS and include earthquake seeds based on the Capitola, Chi-Chi, El Centro, Izmit, and Yermo earthquake records. The CSDRS-HF consists of the input motion based on the Lucerne earthquake seed. The CSDRS compatible input motions are used for soil types 7 and 11. The CSDRS-HF compatible input motion is used for Soil Type 9. For the analysis of the DB model, the input motions between Soil 7 and Soil 11 are different because the in-column motions are being applied at the RXB base elevation.

The X, Y, and Z response transfer functions for nodes of interest are generated from the ANSYS harmonic analysis results files. Acceleration and displacement time histories are then developed from these transfer functions.

3.7.2.5.1 Averaging and Combining Analysis Cases

The algebraic sum of the time histories due to each direction of input is used to obtain the directionally combined time histories. The ISRS of the combined time histories are then calculated for six damping values: 2 percent, 3 percent, 4 percent, 5 percent, 7 percent, and 10 percent. The ISRS are averaged for the five CSDRS input motions. Averaging is not performed for CSDRS-HF because there is only one input motion.

The ISRS at selected nodes that belong to the same group are enveloped. For example, the ISRS of all nodes on the same floor are enveloped to obtain that floor's ISRS. For walls, ISRS at the selected nodes per elevation are enveloped. For the CSDRS, the ISRS from Soil7, Soil11, Soil7-SS

(Soil-Separation) are enveloped. After enveloping, the ISRS are broadened by ±15 percent, in accordance with RG 1.122 and ASCE 4-16 (Reference 3.7.2-6).

Maximum relative displacements are calculated in the same way, at the same locations. First, the maximum relative displacements for each input motion and at each node are calculated, then the average of the maximum relative displacements due to each of the input motions for each node is calculated; finally, an envelope of the displacements over a group of nodes (i.e., the nodes per elevation for each wall, or the nodes on the same floor) is determined.

3.7.2.5.2 Reactor Building In-Structure Response Spectra

Figure 3.7.2-88 shows the selected basemat locations for determining the ISRS. The resulting ISRS at the basemat are shown in Figure 3.7.2-90 for the CSDRS input case and in Figure 3.7.2-92 for the CSDRS-HF input case.

COL Item 3.7-6: An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific analysis that assesses the effects of non-vertically propagating seismic waves on the free-field ground motions and seismic responses of Seismic Category I structures, systems, and components.

3.7.2.5.3 Reactor Building Crane In-Structure Response Spectra

The seismic analysis of the Reactor Building crane uses ISRS and time histories matching the ISRS as input. The ISRS are generated based on 20 points selected on the RX-B and RX-D walls on which the crane moves. Figure 3.7.2-20 shows these locations on the reactor pool wall at the crane rail slab at elevation 145 ft 6 inches. Figure 3.7.2-21 and Figure 3.7.2-22 show the enveloping ISRS for these locations. The seismic analysis of the RBC is completed per ASME NOG-1 (Reference 3.7.2-5).

3.7.2.5.4 NuScale Power Module Skirt and Lug Supports In-Structure Response Spectra

At the NPM skirts, ISRS are generated from the time histories at nodes directly beneath each corresponding NPM. Figure 3.7.2-88, nodes 8 through 13, shows the location of these nodes.

Figure 3.7.2-91 and Figure 3.7.2-93 show the resulting ISRS.

Figure 3.7.2-89 shows selected nodes on the Reactor Building for NPM lug supports for ISRS. The spectra are shown in Figure 3.7.2-94 through Figure 3.7.2-97.

3.7.2.5.5 Control Building In-Structure Response Spectra

The ISRS and displacements are calculated at the following floor locations:

- Elevation 100 ft (basemat)
- Elevation 123 ft (middle floor)
- Elevation 153 ft 3 inches (roof)

Figure 3.7.2-98 through Figure 3.7.2-100 show the floor nodes' locations. Figure 3.7.2-101 through Figure 3.7.2-106 show the ISRS.

3.7.2.6 Three Components of Earthquake Motion

The three components of earthquake motion are developed as separate time histories as discussed in Section 3.7.1.1. When performing a harmonic analysis, the representative response time history of interest of the SSC is obtained by performing separate analyses for each of the three components of earthquake motion (and summing them algebraically). In the case of non-linear time-history analysis, for stability evaluation for example, the three components of earthquake motion are applied simultaneously. In the latter case, the three components of earthquake motion are statistically independent.

Seismic demand is obtained for the three orthogonal (two horizontal and one vertical) components of earthquake motion in accordance with ASCE 4-16 (Reference 3.7.2-6). The orthogonal axes are aligned with the principal axes of the structure.

3.7.2.7 Combination of Modal Responses

Modal responses in seismic response analysis is combined in accordance with RG 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

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

A failure of a nearby structure could adversely affect the SC-I portions of the RXB and CRB. These nearby structures are assessed (or analyzed if necessary) to ensure that there is not a credible potential for adverse interactions. Figure 1.2-4 provides a site plan showing the standard plant layout. The non-SC-I structures that are adjacent to the Seismic Category I RXB and CRB are the Radioactive Waste Building (SC-III), that is adjacent to RXB and the SC-II portion of the CRB that is directly to the north of the SC-I portion of the CRB.

COL Item 3.7-7: An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to a site-specific safe shutdown earthquake will not collapse and adversely affect Seismic Category I portions of the Reactor Building and Control Building.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra

Uncertainties in seismic modeling, due to variation in input parameters such as soil column and earthquake spectrum and structural properties such as material strength, cracking, mass properties, and specific locations of SSC are accounted for in three ways:

- a conservative design approach

  The design considers ground motions that bound most sites, and performs multiple ANSYS analyses using different combinations of soil profiles and cracked and uncracked properties. Bounding results are used in the design.

- ISRS broadening

  The bounding ISRS are broadened as specified in RG 1.122. The envelope ISRS is broadened 15 percent on a linear frequency scale.

- site-specific analysis

  A site-specific analysis is performed to show that the design provides sufficient capacity to resist the site-specific demand.

3.7.2.10 Sensitivity Studies on Soil Separation, Empty Dry Dock, and Modularity

The sensitivity studies of the RXB structural response are performed using the uncracked DB models that are compatible with Soil-7 library for the following cases:

- baseline case

  This is the reference case where the uncracked DB models that are built to be compatible with Soil-7 library are used without modification (i.e., no soil separation, six NPMs are present, and dry dock is full of water). This case is referred to as Baseline7.

- empty dry dock case

  Empty dry dock refers to the case with no water in the dry dock area.

- modularity case

  Modularity refers to the case with a reduced number of NPMs in the Reactor Building. To create the most eccentric NPM responses on the pool and support walls, two NPMs, located on the north side of the pool, are removed, starting from west-most to east.

- soil separation case

  Soil separation refers to the case where there is no contact between the backfill soil and the RXB. The model for this case is generated by reducing the Young's Moduli of the soil layers in the top 25 ft to 1 percent of their original values.

3.7.2.10.1 Methodology for Sensitivity Studies

Sensitivity of the structural response on the effects of different cases is evaluated by comparing the following output quantities to the same output from the baseline case:

- ISRS curves at selected set of nodes. The ISRS curves are compared in terms of the magnitude and the frequency content of the peaks.
- reactions at the base of the structure, the NPMs, and NPM lug supports. The comparison is performed using the peak values of the reactions.
- forces and moments at selected sets of section cuts. The comparison is performed using the peak values of the section cut forces and moments.
- demand-to-capacity ratios for different design conditions calculated at section cuts selected for the force and moment comparison.

The sensitivity of structural response on the effects of different cases is considered significant if the difference from the baseline case is greater than 10 percent.

3.7.2.10.1.1 Calculation and Comparison of In-Structure Response Spectra

The results of SSI analyses are used to calculate absolute acceleration ISRS at 48 selected nodes. Table 3.7.2-7 lists these nodes and their locations in the DB. Averaging is preferred when the nodes are connected to a single component, such as the roof or bioshield; enveloping is preferred otherwise. Figure 3.7.2-23 to Figure 3.7.2-29 show the locations of the nodes, which are organized according to the node groupings provided in Table 3.7.2-8.

Node acceleration time histories are calculated for the models using the CSDRS and CSDRS-HF compatible in-layer acceleration input motions.

The CSDRS compatible acceleration input motions are developed for Capitola (CP), Chi-Chi (CC), El Centro (EC), Izmit (IZ) and Yermo (YM) earthquake time series, while CSDR-HF uses the Lucerne (LUC) earthquake time series.

Figure 3.7.2-30 through Figure 3.7.2-32 present the acceleration time histories and response spectra calculated at a node on RXB roof, for the baseline Soil-7 case. Within figures, a comparison is made between the time histories before and after oversampling for verification purposes.

In-structure response spectra are calculated following the steps below:

1) Absolute acceleration response spectra are calculated for 5 percent damping ratio.

2) For each node, an averaged ISRS is calculated (Figure 3.7.2-33) by taking the events of interest per CSDRS compatible case. Because

CSDRS-HF compatible cases employ only Soil-9 soil library and LUC event, averaging is not performed for them.

3) The ISRS generation procedure is in accordance with the Time Series Method outlined in Section 6.1.2.1 (b) of Reference 3.7.2-6. The ISRS calculation procedure for different groups of nodes is outlined in Table 3.7.2-8.

3.7.2.10.1.2 Calculation and Comparison of NuScale Power Module Support Reactions

The results of SSI analyses are used to calculate the reaction forces at the supports of the selected NPMs and the Reactor Building crane. Because each NPM is connected to the building through six spring elements (Figure 3.7.2-34), the internal element forces of the spring elements are extracted for the reaction forces. All 36 springs (24 for modularity case) are selected to output their element forces (Figure 3.7.2-35).

Table 3.7.2-10 lists selected spring elements used for calculating the reaction forces of selected NPMs. The time-history element forces due to five earthquake motions (Capitola, Chi-Chi, El Centro, Izmit, and Yermo for Soil7) in the X, Y, and Z directions are output. The element forces due to the three directions of each earthquake excitation are then algebraically combined to obtain the total element forces in time-history. The maximum absolute values of the total time-history forces are obtained for each input motion. For the four Soil-7 related cases, the average of the maximum reactions due to five input motions is used for sensitivity evaluation.

Figure 3.7.2-36 presents the transfer functions and the time-history response comparison of spring element 36 for the modularity case with Soil-7 properties and input motion based on the Capitola record.

3.7.2.10.1.3 Calculation and Comparison of Section-Cut Forces and Design Outputs

A set of 44 section cuts is selected based on the critical locations and also based on locations where sensitivity cases are expected to affect the structural response and design. Figure 3.7.2-37 through Figure 3.7.2-43 show the locations of section cuts.

At these section cuts, the SSI analyses are post-processed to calculate force and moment time-histories. The sensitivity of the structural response is evaluated by comparing the absolute peak force and moment values to the corresponding values calculated from the baseline case.

The section cut forces and moments from the baseline case are used in accordance with the RC design methodology and the associated RC design evaluation scripts to calculate a "reference design" (section reinforcement) that results in DCR values that are typically less than 0.8 for different design limit states (e.g., axial-moment interaction and in-plane

shear). The "reference design" is calculated only using the seismic demands from the baseline case without considering the effects of non-seismic loads (e.g., gravity, hydrostatic, thermal) that are part of seismic load combinations. In this context, the "reference design" is considered as a basis for evaluating the sensitivity of design output to changes in forces and moment from different sensitivity cases. The section design (reinforcement) determined as part of the "reference design" is used to calculate the DCR values for the forces and moments from different sensitivity cases. The sensitivity of design outputs is evaluated by comparing the DCR values calculated for different cases.

The design calculations are performed considering that structural members of the Reactor Building are RC sections even though walls are currently designed as SC walls. This consideration is taken in this sensitivity evaluation to simplify the design calculations and to have the same set of design outputs for all members considered in the process. The conclusions on the sensitivity of design based on RC members are applicable to the sensitivity of design based on SC wall members because the primary design limit states (e.g., axial-moment interaction, in-plane and out-of-plane shear) are common to both RC and SC wall components, with some differences in the relationships used in calculating the member capacities.

The sensitivity evaluations for section cut forces and design outputs are performed using SSI analyses based on Soil-7 conditions and five acceleration time-history input motions matching the CSDRS. In this approach, the peak values of force and DCR values are averaged over five input motions and used in sensitivity evaluations.

3.7.2.10.2 Sensitivity Study Results

3.7.2.10.2.1 In-Structure Response Spectra Results

For clarity, the ISRS results from the sensitivity analyses are presented only for 5 percent damping.

3.7.2.10.2.1.1 Empty Dry Dock In-Structure Response Spectra Results

Figure 3.7.2-47 to Figure 3.7.2-50 illustrate the ISRS calculated for 5 percent damping for different node groups for the baseline and empty dry dock cases. The presented ISRS are normalized with respect to the peak value for the baseline case. A graphical comparison of ISRS calculated for full and empty dry dock shows that the response at the dry dock gate is greater when it is empty, however insignificant elsewhere (Figure 3.7.2-48). The graphical comparison of baseline and empty dry dock ISRS shows that emptying the dry dock has local effects that diminish with distance.

3.7.2.10.2.1.2 Modularity In-Structure Response Spectra Results

Figure 3.7.2-51 to Figure 3.7.2-54 present the ISRS calculated for 5 percent damping for different node groups for the baseline and modularity cases. The ISRS are normalized with respect to the peak value for baseline case. A graphical comparison of ISRS calculated for the Reactor Building with six and four NPMs shows that the effects of reducing the number of NPMs is localized with minor impacts on the ISRS outputs.

3.7.2.10.2.1.3 Soil Separation In-Structure Response Spectra Results

Figure 3.7.2-55 to Figure 3.7.2-59 present the ISRS calculated for 5 percent damping for different node groups for the baseline and soil separation cases. The presented ISRS are normalized with respect to the peak value for baseline case. A graphical comparison of ISRS calculated for the two cases shows that soil separation affects ISRS throughout the structure.

3.7.2.10.2.2 NuScale Power Module Support Reaction Results

Table 3.7.2-11 shows the enveloped reaction forces of different types of constraints (X-direction shear lugs, Y-direction shear lugs, X-direction constraint on the basemat, Y-direction constraint on the basemat, and Z-direction constraint on the basemat). Table 3.7.2-11 shows the comparison ratios of sensitivity reactions to the enveloped baseline reaction. The maximum ratio is 1.04 for Soil-7 modularity. Because all ratios are below 1.05, the baseline cases (Soil-7) are sufficient for NPM reaction design, and the soil separation, dry dock, and modularity cases have no significant impact on the envelope of baseline NPM reactions.

3.7.2.10.2.3 Section Cut Force and Design Evaluation Results

Table 3.7.2-12a and Table 3.7.2-12b summarize the calculated force ratios for the sensitivity cases. The results are grouped according to the sensitivity case and each group is assigned a number. Force ratios that are greater than 1.1 are highlighted with red-colored text and blue-colored background. The three sensitivity cases considered are:

- drydock effect (Soil7 Drydock /Baseline7)
- modularity effect (Soil7 Modularity /Baseline7)
- soil-separation effect (Soil7 Soil-Separation /Baseline7)

The key outcomes and conclusions are summarized as follows:

- Results from the Drydock7 case show that the force outputs are 10 percent greater than the force output from the Baseline7 case only for some structural components that are local to the empty dry dock area. This case is considered significant for force outputs only in a localized region around the empty dry dock. The section cut forces in

this localized region can be shown to be governed by static loads (e.g., hydrostatic) and static load combinations and show that the dry dock case need not be included as part of the design basis.

- Results from the Modularity7 case show that the force outputs are typically within 10 percent of the force outputs from the Baseline7 case. This case is considered to be not significant for force outputs.
- Results from the Soil_Separation7 case indicate that the force outputs are typically 10 percent greater than the force outputs from the Baseline7 case. This case is considered to be significant for force outputs.

A comparison of DCR values is used to evaluate the sensitivity of design outputs on the sensitivity cases. Table 3.7.2-13a and Table 3.7.2-13b summarize the calculated design outputs (DCR values and the required in-plane shear reinforcement ratio to meet the threshold DCR limit of 0.8) for all sensitivity cases. The design output values that are 10 percent greater than the Baseline7 case are highlighted with red-colored text and blue-colored background.

The same trends observed for the calculated force ratios are seen for the DCR comparison. The dry dock and modularity effects are minor and localized while the soil separation effects are more pervasive. Both Soil-7 conditions with and without soil separation sufficiently envelope the structure's response.

COL Item 3.7-8: An applicant that references the NuScale Power Plant US460 standard design will demonstrate that the site-specific seismic demand is bounded by the Final Safety Analysis Report capacity for an empty dry dock condition.

3.7.2.10.3 Foundation Uplift

Section 3.8.5 presents the foundation uplift evaluation.

3.7.2.10.4 Use of Constant Vertical Static Factors

Constant vertical static factors are not used in the design of the SC-I and SC-II structures. Vertical seismic loads are generated from the ANSYS analysis.

3.7.2.11 Accidental Torsion

Accidental torsion is considered for the Reactor Building, Control Building, and Radioactive Waste Building in accordance with ASCE 7-16 (Reference 3.7.2-7). The effect of accidental torsion is accounted for by increasing the demand forces and moments due to east-west and north-south CSDRS (and CSDRS-HF) by 5 percent.

3.7.2.12 Comparison of Results

The response spectrum method is not used in the evaluation of the site-independent SC-I structures. The ANSYS analysis is a time-history analysis method. Therefore, a direct comparison is not applicable.

3.7.2.13 Methods for Seismic Analysis of Dams

US460 design does not include dams.

3.7.2.14 Determination of Dynamic Stability of Seismic Category I Structures

Section 3.8.5 discusses bearing pressure, lateral wall pressure, overturning, sliding, and flotation.

3.7.2.15 Analysis Procedure for Damping

Section 3.7.1 describes the damping ratios used for seismic analysis of the RXB and CRB. For analyses of Seismic Category I SSC, the damping values of RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," are used. For the soil and rock materials, the damping ratio is obtained based on strain-compatible soil properties generated for each soil profile. Soil material damping ratios are shown on Table 3.7.1-9 through Table 3.7.1-11 for each soil type considered. Soil damping ratio is limited to 15 percent.

Damping values for linear elastic analysis depends on the level of cracking expected during the SSE. TR-0920-71621-P-A (Reference 3.7.2-9) provides additional description.

3.7.2.16 Site-Specific Seismic Analysis

Site-specific seismic analysis is performed by the applicant to confirm that the site-independent SC-I structures can be constructed without modification, or to identify where modifications are necessary. Section 3.8.4 documents this comparison. The site-specific analysis is performed using the site-specific SSE and the site-specific soil profile developed in Section 3.7.1. Appendix 3B critical sections include RXB exterior walls that are subject to earth pressures. Therefore, by comparing seismic demand in these walls, site-specific versus lateral certified standard soil pressures are also compared.

COL Item 3.7-9: An applicant that references the NuScale Power Plant US460 standard design will perform a soil-structure interaction analysis of the Reactor Building and the Control Building using the NuScale ANSYS models for those structures. The applicant will confirm that the site-specific seismic demands of the standard design for critical structures, systems, and components in Appendix 3B are bounded by the corresponding design certified seismic demands and, if not, the standard design for critical structures, systems, and components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands. Seismic

demands investigated shall include forces, moments, deformations, in-structure response spectra, and seismic stability of the structures.

3.7.2.17 References

3.7.2-1 NuScale Power, LLC, “Improvements in Frequency Domain Soil-Structure Fluid Interaction Analysis,” TR-0118-58005-P-A, Revision 2.

3.7.2-2 SDE SASSI, v2.1.1, Carl J. Costantino and Associates, LLC., Spring Valley, New York.

3.7.2-3 ACS SASSI Version 4.3.1, Ghiocel Predictive Technologies, Inc., Pittsford, New York.

3.7.2-4 ANSYS (Release 19.2) [Computer Program]. (2019). Canonsburg, PA, ANSYS Incorporated.

3.7.2-5 American Society of Mechanical Engineers, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder), ASME NOG-1, 2004, New York, NY.

3.7.2-6 American Society of Civil Engineers, Seismic Analysis of Safety-Related Nuclear Structures, ASCE/SEI 4-16, 2016, Reston, Virginia.

3.7.2-7 American Society of Civil Engineers, Minimum Design Loads and Associated Criteria for Buildings and Other Structures, ASCE/SEI 7-16, 2016 Reston, Virginia.

3.7.2-8 American Society of Civil Engineers, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, ASCE 43-05.

3.7.2-9 NuScale Power, LLC, "Building Design and Analysis Methodology for Safety-Related Structures," TR-0920-71621-P-A, Revision 1.

3.7.2-10 American Concrete Institute, Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary, ACI 349-13.

3.7.2-11 American Institute of Steel Construction, Specification for Safety-Related Steel Structures for Nuclear Facilities, AISC N690-18.

3.7.2-12 Python Software Foundation, Python v3.7.4, July 2019.

3.7.2-13 Python Software Foundation. Python Language Reference, version 2.7. Available at http://www.python.org.

3.7.2-14 MATLAB, 2021. 9.10. 0.1602886 (R20221a), Natick, Massachusetts, The MathWorks Inc.

3.7.2-15 NuScale Power, LLC, "Quality Assurance Program Description," Revision 0, MN-122626, November 2022.

3.7.2-16 NuScale Power, LLC, "US460 NuScale Power Module Seismic Analysis," TR-121515, Revision 0.

Table 3.7.2-1: Double Building Model Mass Participation Factors in X Direction for Five Major Modes

Mode Number	Frequency (Hz)	Eff. Mass / Total Mass (%)
41	7.2	13.40
39	6.77	5.52
49	7.56	1.63
96	9.26	1.39
98	9.29	1.33

Table 3.7.2-2: Double Building Model Mass Participation Factors in Y Direction for Five Major Modes

Mode Number	Frequency (Hz)	Eff. Mass / Total Mass (%)
30	5.14	10.9
75	8.75	4.87
117	9.75	2.22
77	8.8	2.04
152	11.61	1.58

Table 3.7.2-3: Double Building Model Mass Participation Factors in Z Direction for Five Major Modes

Mode Number	Frequency (Hz)	Eff. Mass / Total Mass (%)
300	14.9	1.5
301	14.92	1.17
351	15.7	1.04
278	14.67	0.86
28	4.29	0.73

Table 3.7.2-4: Control Building Model Mass Participation Factors in X Direction for Five Major Modes

Mode Number	Frequency (Hz)	Eff. Mass / Total Mass (%)
13	16.17	39%
16	18.1	10%
48	33.17	1%
46	32.43	1%
38	28.38	1%

Table 3.7.2-5: Control Building Model Mass Participation Factors in Y Direction for Five Major Modes

Mode Number	Frequency (Hz)	Eff. Mass / Total Mass (%)
7	11.34313	42%
8	12.19966	11%
24	22.86103	3%
25	23.86423	2%
37	27.73076	1%

Table 3.7.2-6: Control Building Model Mass Participation Factors in Z Direction for Five Major Modes

Mode Number	Frequency (Hz)	Eff. Mass / Total Mass (%)
86	44.96395	8%
89	46.41699	4%
4	8.52987	4%
1	5.22661	4%
2	7.31213	3%

Table 3.7.2-7: List of Nodes Selected to Calculate In-Structure Response Spectra in the Reactor Building for Sensitivity

ID Number	Node ID in the Model	X (in.)	Y (in.)	Z (in.)
1	3981	1341.00	-9.00	120.00
2	20143	1939.00	-675.00	468.00
3	20885	2553.00	-104.00	468.00
4	73601	339.00	214.50	468.00
5	21838	1939.00	675.00	468.00
6	21814	886.25	675.00	468.00
7	20119	886.25	-675.00	468.00
8	29801	1939.00	-675.00	648.00
9	30551	2553.00	-104.00	648.00
10	73799	339.00	214.50	648.00
11	31504	1939.00	675.00	648.00
12	31480	886.25	675.00	648.00
13	29777	886.25	-675.00	648.00
14	41119	1939.00	-675.00	828.00
15	41637	2553.00	-104.00	828.00
16	42290	1939.00	675.00	828.00
17	42266	886.25	675.00	828.00
18	41095	886.25	-675.00	828.00
19	48683	1939.00	-675.00	1008.00
20	49304	2553.00	-104.00	1008.00
21	50039	1939.00	675.00	1008.00
22	50015	886.25	675.00	1008.00
23	48659	886.25	-675.00	1008.00
24	76558	300.75	-228.00	1008.00
25	74083	339.00	214.50	1008.00
26	57445	1939.00	-675.00	1320.00
27	58034	1939.00	675.00	1320.00
28	61953	1898.00	675.00	1566.00
29	61418	1898.00	-675.00	1566.00
30	61504601	886.25	675.00	1566.00
31	61508123	886.25	-675.00	1566.00
32	66744	424.50	-9.00	2046.00
33	66765	1341.00	-9.00	2046.00
34	66789	2376.00	-9.00	2046.00
35	16561	1125.00	375.00	411.00
36	125169	1125.00	99.00	411.00
37	84499	1125.00	375.00	699.88
38	125213	1125.00	99.00	699.88
39	49615	1125.00	375.00	1008.00
40	125207	1125.00	99.00	1008.00
41	55366	1563.00	138.00	1260.00
42	55462	1563.00	417.00	1260.00
43	55368	1857.00	138.00	1284.00
44	55466	1857.00	417.00	1284.00
45	55370	2151.00	138.00	1284.00
46	55470	2151.00	417.00	1284.00
47	55371	2421.00	138.00	1260.00
48	55473	2421.00	417.00	1260.00

Table 3.7.2-8: Group of Nodes to Combine In-Structure Response Spectra for Sensitivity Studies

Group Name	Description	IDs in the Group	ISRS Combination Method
EL 26	Node on top of the basemat	1	N/A
EL 55	Nodes on top of the slabs at 55 ft elevation	2, 3, 4, 5, 6, 7	Envelope
EL 70	Nodes on top of the slabs at 70 ft elevation	8, 9, 10, 11, 12, 13	Envelope
EL 85	Nodes on top of the slabs at 85 ft elevation	14, 15, 16, 17, 18	Envelope
EL 100	Nodes on top of the slabs at 100 ft elevation	19, 20, 21, 22, 23, 24, 25	Envelope
EL 126	Nodes on top of the slabs at 126 ft elevation	26, 27	Envelope
EL 146	Nodes on top of the slabs at 146 ft elevation	28, 29, 30, 31	Envelope
RXB_Roof	Nodes on top of the RXB Roof	32, 33, 34	Average
Drydock gate	Nodes around the dry dock gate location	35, 36, 37, 38, 39, 40	Average
Bioshield	Nodes around the bioshield above the 3 NPMs to the north of the pool	41, 42, 43, 44, 45, 46, 47, 48	Average

Table 3.7.2-9: Transfer Function Components of Structural Responses for Each Input Motion Direction, u

Response	$\boldsymbol{TF_u}(\omega)$
Acceleration	$[Acc_X, Acc_Y, Acc_Z]_u$
Displacement	$[Disp_X, Disp_Y, Disp_Z]_u$
Element forces	$[N_{11}, N_{22}, N_{12}, M_{11}, M_{22}, M_{12}, Q_{13}, Q_{23}]_u$
Section cut forces	$[F_X, F_Y, F_Z, M_X, M_Y, M_Z]_u$

Table 3.7.2-10: List of Spring Elements Selected to Calculate Reactions of NuScale Power Modules

ElemID	Elem#	X (in.)	Y (in.)	Z (in.)
elem_1	9057839	1787.875	264	935.6
elem_2	9057840	1584.125	264	935.6
elem_3	9057841	1686	365.875	935.6
elem_4	9057842	1686	264	120
elem_5	9057843	1686	264	120
elem_6	9057844	1686	264	120
elem_7	9057846	2081.875	264	935.6
elem_8	9057847	1878.125	264	935.6
elem_9	9057848	1980	365.875	935.6
elem_10	9057849	1980	264	120
elem_11	9057850	1980	264	120
elem_12	9057851	1980	264	120
elem_13	9057853	2375.875	264	935.6
elem_14	9057854	2172.125	264	935.6
elem_15	9057855	2274	365.875	935.6
elem_16	9057856	2274	264	120
elem_17	9057857	2274	264	120
elem_18	9057858	2274	264	120
elem_19	9057860	2375.875	-264	935.6
elem_20	9057861	2172.125	-264	935.6
elem_21	9057862	2274	-365.875	935.6
elem_22	9057863	2274	-264	120
elem_23	9057864	2274	-264	120
elem_24	9057865	2274	-264	120
elem_25	9057867	2081.875	-264	935.6
elem_26	9057868	1878.125	-264	935.6
elem_27	9057869	1980	-365.875	935.6
elem_28	9057870	1980	-264	120
elem_29	9057871	1980	-264	120
elem_30	9057872	1980	-264	120
elem_31	9057874	1787.875	-264	935.6
elem_32	9057875	1584.125	-264	935.6
elem_33	9057876	1686	-365.875	935.6
elem_34	9057877	1686	-264	120
elem_35	9057878	1686	-264	120
elem_36	9057879	1686	-264	120

Table 3.7.2-11: Sensitivity Ratios for NuScale Power Module Reactions

Component	Baseline	Soil7_SS	Soil7_Drydoc	Soil7_Modularity
	Baseline	Baseline	Baseline	Baseline
Shear_Lug_X	1	0.850	1.011	1.015
Shear_Lug_Y	1	0.894	0.989	1.002
Base_X	1	0.851	0.992	1.010
Base_Y	1	0.785	0.993	0.990
Base_Z	1	0.887	1.003	1.039

Table 3.7.2-12a: Comparison of Absolute Maximum Section Cut Forces for All Sensitivity Cases

		1						2					
		Ratio = (Drydock$_7$)/(Baseline$_7$)						Ratio = (Modularity$_7$)/(Baseline$_7$)					
	Cut ID	P	V_{ip}	V_{op}	M_{tor}	M_{op}	M_{ip}	P	V_{ip}	V_{op}	M_{tor}	M_{op}	M_{ip}
Walls at Basemat EL	Cut_1	1.06	1.00	1.00	1.03	1.01	1.01	1.00	1.00	1.03	1.01	1.04	0.99
	Cut_2	1.01	1.01	0.99	1.00	1.04	1.02	1.01	0.99	1.00	0.97	0.99	0.98
	Cut_3	1.00	1.00	1.01	0.98	1.01	1.00	1.00	1.00	1.00	1.08	1.00	1.00
	Cut_4	1.05	1.01	1.53	0.57	1.53	0.96	1.00	1.01	1.00	1.02	1.01	1.00
	Cut_5	1.55	1.08	1.41	1.26	2.57	1.24	1.05	1.00	1.01	1.01	1.06	1.00
	Cut_6	1.01	1.01	0.97	1.04	0.93	1.04	1.00	1.00	1.02	1.03	1.01	1.00
	Cut_7	1.01	1.00	1.03	0.99	1.00	1.00	1.00	1.00	1.01	0.98	1.00	1.01
	Cut_8	0.99	1.03	1.03	1.01	1.02	1.03	1.00	1.00	1.00	1.01	0.98	1.04
	Cut_9	1.00	1.00	1.01	1.00	1.01	1.00	1.00	1.00	1.01	1.00	1.01	1.00
Basemat Slab	Cut_10	0.99	1.13	1.00	0.97	1.02	0.98	1.06	1.05	1.00	1.08	1.01	0.99
	Cut_11	1.01	0.99	1.02	1.02	1.01	0.96	1.03	1.01	0.99	1.02	1.02	1.00
	Cut_12	1.00	1.03	1.01	1.01	1.00	1.00	1.03	1.02	1.02	1.05	1.01	1.07
	Cut_13	0.99	1.04	1.03	1.13	0.99	1.02	1.07	1.03	1.02	1.01	1.01	1.01
FL 55ft	Cut_14	1.00	1.05	1.01	1.02	1.07	0.93	1.00	1.00	1.00	1.00	1.02	1.01
	Cut_15	1.01	0.95	1.00	1.02	1.02	0.96	0.99	1.02	1.00	1.01	1.01	1.02
FL 70ft	Cut_16	1.02	1.02	1.01	1.03	1.01	1.00	1.00	1.00	1.01	1.01	1.00	1.00
	Cut_17	1.03	0.97	1.00	1.04	1.01	0.99	1.00	1.01	1.00	1.01	1.00	1.01
	Cut_18	1.02	0.91	0.99	1.02	1.00	0.98	0.99	1.01	0.99	0.99	1.01	1.04
FL 85ft	Cut_19	1.02	0.97	1.01	1.03	1.00	0.97	1.00	1.00	1.01	1.01	1.01	1.00
	Cut_20	1.03	1.02	1.03	1.01	0.99	1.09	1.00	1.02	1.00	1.00	1.00	1.01
Floor at EL 100ft	Cut_21	0.99	1.00	0.99	1.04	1.00	1.01	1.04	0.97	1.00	0.99	1.01	0.99
	Cut_22	1.04	1.04	1.00	1.01	1.01	1.02	1.20	1.13	1.01	1.00	1.01	1.01
	Cut_23	0.97	1.03	1.00	1.02	1.02	1.08	1.01	1.00	1.00	1.00	1.00	1.01
	Cut_24	1.01	1.03	1.00	1.02	0.99	0.97	1.01	1.01	1.01	1.02	1.00	1.00
	Cut_25	1.02	0.98	0.99	1.02	1.00	1.02	1.01	1.00	1.00	1.03	1.00	1.01
	Cut_26	1.01	1.05	1.00	1.01	1.00	1.04	1.04	1.03	1.00	0.99	1.00	1.05
	Cut_27	1.01	1.00	1.02	1.01	1.05	1.00	1.01	1.00	1.01	1.00	0.99	1.00
	Cut_28	1.06	1.14	0.88	1.66	1.04	1.07	1.04	0.99	1.01	1.01	1.00	1.00
	Cut_29	1.11	1.03	0.97	1.74	0.82	1.00	1.00	0.99	1.00	1.02	1.02	1.00
Roof at EL 188ft	Cut_30	1.00	1.01	1.01	1.01	0.65	1.01	1.00	1.00	1.00	1.02	0.75	1.00
	Cut_31	1.03	1.00	1.00	1.01	1.00	1.00	1.01	1.00	1.00	1.00	1.00	1.00
	Cut_32	1.13	0.99	1.02	1.01	1.01	1.00	1.04	1.00	1.01	1.02	1.00	1.00
	Cut_33	1.00	1.00	1.00	1.06	1.01	1.00	1.01	0.99	1.00	1.01	1.00	0.99
	Cut_34	1.00	0.99	1.00	0.99	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
	Cut_35	1.00	1.00	1.01	1.00	1.01	1.00	0.99	1.00	1.00	1.00	1.00	1.00
	Cut_36	1.00	1.01	1.01	1.00	1.00	1.00	0.99	1.02	1.00	1.00	1.00	1.01
	Cut_37	1.00	1.01	1.01	1.00	0.99	1.00	1.00	1.02	1.01	1.00	1.00	0.99
Roof at EL 146.5ft	Cut_38	1.03	1.00	1.01	1.24	0.99	1.04	1.03	1.02	1.01	1.04	1.04	1.01
	Cut_39	1.00	1.00	1.02	1.01	1.01	0.98	0.99	1.02	0.99	1.00	1.00	1.00
	Cut_40	1.00	0.97	0.99	1.00	1.01	1.00	1.00	1.00	1.00	1.00	1.01	1.00
	Cut_41	1.01	1.01	1.00	1.01	1.00	1.03	1.00	1.01	1.00	0.99	1.01	1.01
	Cut_42	0.99	1.00	1.00	0.99	0.99	1.00	1.00	1.01	1.00	0.99	1.00	1.01
	Cut_43	1.01	1.02	1.02	1.03	1.03	1.00	1.00	1.01	1.01	1.01	1.03	1.00
	Cut_44	1.00	0.99	1.03	1.03	1.05	1.02	1.00	1.02	1.04	1.02	0.99	1.00

Table 3.7.2-12b: Comparison of Absolute Maximum Section Cut Forces for All Sensitivity Cases

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		Ratio = (Soil_Separation$_7$)/(Baseline$_7$)					
	Cut ID	P	V_{ip}	V_{op}	M_{tor}	M_{op}	M_{ip}
Walls at Basemat EL	Cut_1	**1.12**	0.99	1.00	1.01	**1.36**	0.97
	Cut_2	0.81	1.04	**1.64**	**1.32**	**1.26**	**1.10**
	Cut_3	**1.12**	1.01	**1.21**	0.87	**1.13**	1.02
	Cut_4	1.08	0.98	0.94	1.01	0.96	0.99
	Cut_5	1.10	0.98	0.97	0.88	1.01	0.98
	Cut_6	1.01	1.05	0.95	1.04	0.97	1.09
	Cut_7	**1.24**	0.98	**1.19**	0.88	**1.13**	1.01
	Cut_8	**1.14**	1.03	1.03	**1.37**	0.86	1.05
	Cut_9	0.96	**1.17**	1.04	0.98	0.99	**1.19**
Basemat Slab	Cut_10	**1.12**	0.94	1.03	0.64	1.03	**1.42**
	Cut_11	**1.19**	1.08	**1.52**	1.00	0.97	**1.25**
	Cut_12	**1.74**	1.06	**1.58**	**1.16**	**1.47**	0.82
	Cut_13	**1.14**	1.01	**1.37**	1.04	**1.30**	0.81
FL 55ft	Cut_14	1.05	1.09	1.04	**1.38**	**1.12**	1.00
	Cut_15	0.94	0.91	**1.27**	1.00	**1.27**	1.10
FL 70ft	Cut_16	0.98	0.87	1.05	**1.51**	1.06	0.89
	Cut_17	0.86	1.00	**1.33**	**1.13**	**1.33**	0.93
	Cut_18	**1.10**	0.82	1.01	1.06	**1.21**	0.84
FL 85ft	Cut_19	0.79	0.67	**1.11**	**1.38**	0.99	0.77
	Cut_20	0.95	**1.15**	0.97	1.03	0.94	**1.36**
Floor at EL 100ft	Cut_21	**2.30**	**1.59**	**1.12**	0.96	1.04	**1.60**
	Cut_22	**2.15**	**1.69**	1.01	1.04	1.07	**1.38**
	Cut_23	**1.73**	**1.55**	0.99	**1.15**	0.95	**1.39**
	Cut_24	0.35	0.98	0.98	1.10	0.96	1.00
	Cut_25	0.39	**1.26**	0.98	**1.11**	0.98	0.95
	Cut_26	**2.56**	**1.47**	0.99	0.99	1.00	**1.27**
	Cut_27	0.82	**1.42**	1.09	1.09	**1.31**	**1.51**
	Cut_28	**1.46**	**1.43**	**1.25**	1.08	**1.46**	0.80
	Cut_29	**1.70**	**1.43**	**1.16**	1.07	0.99	**1.46**
Roof at EL 188ft	Cut_30	0.91	0.91	1.05	0.95	0.76	1.01
	Cut_31	0.94	0.96	1.03	0.98	1.03	0.95
	Cut_32	0.90	0.89	**1.13**	0.99	1.04	0.87
	Cut_33	0.93	0.88	1.01	0.90	1.01	0.88
	Cut_34	0.94	0.95	0.99	1.04	1.02	0.92
	Cut_35	0.88	0.89	1.08	0.97	0.97	0.88
	Cut_36	0.89	0.89	**1.10**	1.08	1.00	0.94
	Cut_37	0.91	0.82	**1.10**	1.09	0.97	0.92
Roof at EL 146.5ft	Cut_38	0.87	0.86	0.95	**1.12**	0.94	0.97
	Cut_39	1.09	0.88	0.96	0.93	0.98	**1.39**
	Cut_40	0.97	1.01	0.94	1.00	1.04	0.99
	Cut_41	0.93	**1.14**	0.88	1.07	0.90	1.08
	Cut_42	0.93	**1.17**	0.92	1.02	0.93	**1.15**
	Cut_43	**1.22**	0.91	0.94	0.99	1.08	0.94
	Cut_44	1.01	0.97	1.07	1.10	1.00	1.06

Table 3.7.2-13a: Comparison of Demand-to-Capacity Ratio Values at Each Section Cut for all Sensitivity Cases

		1				2				3			
		Baseline$_7$ - DCR				Drydock$_7$ - DCR				Modularity$_7$ - DCR			
	Cut ID	PM_{op}	PV_{op}	V_{ip}	$\rho_{ip}x10^3$	PM_{op}	PV_{op}	V_{ip}	$\rho_{ip}x10^3$	PM_{op}	PV_{op}	V_{ip}	$\rho_{ip}x10^3$
Walls at Basemat EL	Cut_1	0.33	0.03	0.80	3.437	0.34	0.03	0.80	3.419	0.33	0.03	0.80	3.369
	Cut_2	0.24	0.16	0.80	3.593	0.24	0.15	0.80	3.568	0.24	0.15	0.80	3.517
	Cut_3	0.87	0.70	0.80	4.491	0.87	0.70	0.80	4.466	0.86	0.69	0.80	4.423
	Cut_4	0.76	0.21	0.80	0.119	0.72	0.23	0.77	0.088	0.76	0.21	0.80	0.120
	Cut_5	0.13	0.18	0.74	0.000	**0.24**	**0.24**	0.75	0.000	0.14	0.18	0.74	0.000
	Cut_6	0.49	0.42	0.80	4.536	0.47	0.37	0.80	4.567	0.49	0.42	0.80	4.535
	Cut_7	0.75	0.55	0.80	4.193	0.75	0.55	0.80	4.198	0.74	0.55	0.80	4.196
	Cut_8	0.03	0.01	0.03	0.000	0.03	0.01	0.03	0.000	0.03	0.01	0.03	0.000
	Cut_9	0.57	0.31	0.80	1.633	0.57	0.31	0.80	1.630	0.57	0.31	0.80	1.619
Basemat Slab	Cut_10	0.18	0.26	0.13	0.000	0.17	0.26	0.12	0.000	0.18	0.26	0.13	0.000
	Cut_11	0.10	0.38	0.43	0.000	0.09	0.39	0.43	0.000	0.10	0.38	0.43	0.000
	Cut_12	0.40	0.43	0.26	0.000	0.39	0.43	0.26	0.000	0.40	0.43	0.27	0.000
	Cut_13	0.33	0.37	0.27	0.000	0.33	0.37	0.28	0.000	0.33	0.37	0.28	0.000
FL 55ft	Cut_14	0.29	0.12	0.80	1.504	0.29	0.13	0.80	1.229	0.29	0.13	0.80	1.510
	Cut_15	0.25	0.15	0.80	2.384	0.26	0.15	0.80	2.282	0.25	0.15	0.80	2.339
FL 70ft	Cut_16	0.34	0.14	0.80	2.143	0.34	0.14	0.80	2.025	0.34	0.14	0.80	2.151
	Cut_17	0.22	0.16	0.80	1.701	0.22	0.16	0.80	1.684	0.22	0.16	0.80	1.709
	Cut_18	0.15	0.06	0.80	1.098	0.14	0.06	0.80	0.771	0.14	0.06	0.80	1.127
FL 85ft	Cut_19	0.35	0.13	0.80	1.843	0.35	0.13	0.80	1.789	0.35	0.13	0.80	1.854
	Cut_20	0.20	0.08	0.80	0.282	0.20	0.08	0.80	0.299	0.20	0.08	0.80	0.282
Floor at EL 100ft	Cut_21	0.53	0.24	0.80	0.473	0.52	0.24	0.80	0.475	0.53	0.24	0.80	0.332
	Cut_22	0.13	0.12	0.76	0.133	0.13	0.12	0.77	0.124	0.13	0.12	0.78	**0.147**
	Cut_23	0.10	0.04	0.80	2.402	0.10	0.04	0.80	2.451	0.10	0.04	0.80	2.319
	Cut_24	0.63	0.48	0.47	0.000	0.62	0.48	0.46	0.000	0.63	0.48	0.47	0.000
	Cut_25	0.56	0.51	0.76	0.000	0.56	0.51	0.75	0.000	0.57	0.51	0.76	0.000
	Cut_26	0.16	0.11	0.79	0.152	0.16	0.11	0.79	**0.191**	0.15	0.11	0.79	0.157
	Cut_27	0.13	0.05	0.80	3.055	0.13	0.05	0.80	3.119	0.13	0.05	0.80	2.945
	Cut_28	0.56	0.22	0.80	1.142	0.57	0.21	0.80	1.186	0.56	0.22	0.80	1.114
	Cut_29	0.93	0.29	0.80	2.183	0.83	0.25	0.80	2.364	0.88	0.29	0.80	2.132
Roof at EL 188ft	Cut_30	0.04	0.08	1.21	7.071	0.04	0.08	1.21	7.071	0.04	0.08	1.20	7.071
	Cut_31	0.21	0.30	0.86	7.071	0.22	0.30	0.85	7.071	0.21	0.30	0.86	7.071
	Cut_32	0.76	0.07	1.02	7.071	0.77	0.07	1.02	7.071	0.76	0.07	1.02	7.071
	Cut_33	0.09	0.14	1.35	7.071	0.09	0.14	1.35	7.071	0.09	0.14	1.34	7.071
	Cut_34	0.33	0.05	1.05	7.071	0.33	0.05	1.05	7.071	0.33	0.05	1.05	7.071
	Cut_35	0.33	0.05	0.99	7.071	0.32	0.05	0.99	7.071	0.33	0.05	0.98	7.071
	Cut_36	0.50	0.03	0.26	7.071	0.50	0.03	0.26	7.071	0.50	0.03	0.27	7.071
	Cut_37	0.46	0.03	0.25	7.071	0.46	0.03	0.25	7.071	0.46	0.03	0.25	7.071
Roof at EL 146.5ft	Cut_38	0.42	0.15	0.80	1.521	0.43	0.16	0.80	1.537	0.43	0.15	0.80	1.544
	Cut_39	0.42	0.25	0.88	7.071	0.42	0.26	0.87	7.071	0.41	0.25	0.89	7.071
	Cut_40	0.69	0.11	0.80	1.467	0.69	0.10	0.80	1.465	0.69	0.10	0.80	1.444
	Cut_41	0.47	0.60	0.31	0.000	0.46	0.60	0.31	0.000	0.47	0.60	0.31	0.000
	Cut_42	0.43	0.52	0.59	0.000	0.43	0.52	0.59	0.000	0.43	0.52	0.59	0.000
	Cut_43	0.14	0.14	0.80	5.878	0.14	0.14	0.80	5.928	0.14	0.14	0.80	5.783
	Cut_44	0.55	0.09	0.80	3.072	0.54	0.09	0.80	3.059	0.54	0.09	0.80	3.135

Table 3.7.2-13b: Comparison of Demand-to-Capacity Ratio Values at Each Section Cut for all Sensitivity Cases

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		Soil_Separation$_7$ - DCR			
	Cut ID	PM_{op}	PV_{op}	V_{ip}	$\rho_{ip}x10^3$
Walls at Basemat EL	Cut_1	0.37	0.03	0.80	3.281
	Cut_2	0.16	0.26	0.80	3.429
	Cut_3	1.03	1.06	0.80	4.392
	Cut_4	0.82	0.19	0.79	0.052
	Cut_5	0.13	0.16	0.71	0.000
	Cut_6	0.46	0.36	0.80	4.513
	Cut_7	0.79	0.63	0.80	3.983
	Cut_8	0.03	0.01	0.03	0.000
	Cut_9	0.52	0.29	0.80	2.432
Basemat Slab	Cut_10	0.17	0.25	0.12	0.000
	Cut_11	0.10	0.50	0.44	0.000
	Cut_12	0.58	0.60	0.26	0.000
	Cut_13	0.39	0.45	0.26	0.000
FL 55ft	Cut_14	0.31	0.12	0.80	1.868
	Cut_15	0.30	0.18	0.80	2.110
FL 70ft	Cut_16	0.33	0.13	0.80	1.541
	Cut_17	0.25	0.19	0.80	1.383
	Cut_18	0.15	0.06	0.80	0.328
FL 85ft	Cut_19	0.31	0.12	0.79	0.241
	Cut_20	0.19	0.08	0.80	0.499
Floor at EL 100ft	Cut_21	0.76	0.28	0.80	1.441
	Cut_22	0.17	0.12	0.80	1.657
	Cut_23	0.13	0.04	0.80	4.724
	Cut_24	0.37	0.44	0.41	0.000
	Cut_25	0.38	0.48	0.80	0.237
	Cut_26	0.19	0.11	0.80	0.883
	Cut_27	0.14	0.05	0.80	5.620
	Cut_28	0.81	0.31	0.80	2.044
	Cut_29	1.13	0.38	0.80	4.147
Roof at EL 188ft	Cut_30	0.04	0.08	1.04	7.071
	Cut_31	0.22	0.31	0.73	7.071
	Cut_32	0.77	0.07	0.85	7.071
	Cut_33	0.08	0.14	1.13	7.071
	Cut_34	0.32	0.05	0.90	7.071
	Cut_35	0.28	0.05	0.82	7.071
	Cut_36	0.45	0.03	0.21	7.071
	Cut_37	0.41	0.03	0.20	7.071
Roof at EL 146.5ft	Cut_38	0.34	0.14	0.80	1.055
	Cut_39	0.43	0.23	0.74	7.071
	Cut_40	0.71	0.09	0.80	1.489
	Cut_41	0.40	0.49	0.34	0.000
	Cut_42	0.37	0.45	0.64	0.000
	Cut_43	0.15	0.13	0.80	5.240
	Cut_44	0.52	0.09	0.80	2.914

Table 3.7.2-14a: Relative Displacement for RXB Floors due to Certified Seismic Design Response Spectra

Group	Max Displacement @ Each Node				Enveloped Displacement		
	Node	Disp_X (in.)	Disp_Y (in.)	Disp_Z (in.)	Ux (in.)	Uy (in.)	Uz (in.)
Basemat @ EL25'	1	0.0147	0.0131	0.1063	0.0197	0.0569	0.1482
	2	0.0140	0.0227	0.1137			
	3	0.0098	0.0470	0.1482			
	4	0.0018	0.0013	0.0128			
	5	0.0133	0.0569	0.1095			
	6	0.0131	0.0111	0.1271			
	7	0.0197	0.0224	0.1128			
NPM Base Supports @ EL26'	8	0.0082	0.0175	0.0516	0.0127	0.0399	0.0967
	9	0.0100	0.0264	0.0699			
	10	0.0119	0.0396	0.0967			
	11	0.0096	0.0176	0.0404			
	12	0.0116	0.0265	0.0666			
	13	0.0127	0.0399	0.0926			
Floor @ EL55'	14	0.1150	0.1754	0.1886	0.1433	0.1917	0.2087
	15	0.1115	0.1857	0.1216			
	16	0.1140	0.1851	0.1439			
	17	0.1181	0.1909	0.1996			
	18	0.1111	0.1775	0.1578			
	19	0.1433	0.1881	0.1183			
	20	0.1236	0.1783	0.2087			
	21	0.1174	0.1656	0.1339			
	22	0.1173	0.1765	0.1377			
	23	0.1212	0.1917	0.1815			
Floor @ EL70'	24	0.1634	0.2372	0.1915	0.1871	0.2732	0.2135
	25	0.1580	0.2732	0.1233			
	26	0.1608	0.2671	0.1384			
	27	0.1655	0.2546	0.2069			
	28	0.1652	0.2376	0.1602			
	29	0.1871	0.2513	0.1224			
	30	0.1757	0.2401	0.2135			
	31	0.1634	0.2458	0.1440			
	32	0.1637	0.2569	0.1320			
	33	0.1683	0.2551	0.1861			
Floor @ EL85'	34	0.2093	0.2984	0.1952	0.2370	0.3464	0.3901
	35	0.2194	0.3464	0.3706			
	36	0.2214	0.3425	0.3763			
	37	0.2197	0.3138	0.2136			
	38	0.2370	0.3121	0.1286			
	39	0.2113	0.3001	0.2178			
	40	0.2109	0.3203	0.3582			
	41	0.2182	0.3310	0.3901			
	42	0.2174	0.3142	0.1902			

Table 3.7.2-14a: Relative Displacement for RXB Floors due to Certified Seismic Design Response Spectra (Continued)

Group	Max Displacement @ Each Node				Enveloped Displacement		
	Node	Disp_X (in.)	Disp_Y (in.)	Disp_Z (in.)	Ux (in.)	Uy (in.)	Uz (in.)
Floor @ EL100'	43	0.2712	0.3628	0.1976	0.3486	0.4843	0.3696
	44	0.2764	0.4843	0.1979			
	45	0.2772	0.4446	0.3696			
	46	0.2848	0.3700	0.2205			
	47	0.2491	0.3581	0.1734			
	48	0.2487	0.3656	0.1413			
	49	0.3486	0.3677	0.1322			
	50	0.2700	0.3586	0.2219			
	51	0.2652	0.4731	0.2023			
	52	0.2754	0.4310	0.3690			
	53	0.2840	0.3702	0.1962			
Floor @ EL126'	54	0.3716	0.6624	0.1587	0.3716	0.6624	0.1587
	55	0.3694	0.6308	0.1463			
Floor @ EL145'	56	0.4204	0.8411	0.1687	0.5359	1.6341	0.9159
	57	0.4238	0.8039	0.1543			
	58	0.5359	1.2526	0.2201			
	59	0.5228	1.6341	0.9159			
Roof @ EL187'	60	0.5186	1.7325	1.0154	0.5394	1.7325	1.0154
	61	0.5241	1.5092	0.8779			
	62	0.5394	1.0594	0.1692			
	63	0.4105	1.1569	0.1665			
	64	0.3960	1.1069	0.1939			

Table 3.7.2-14b: Relative Displacement for RXB Floors due to Certified Seismic Design Response Spectra - High Frequency

Group	Max Displacement @ Each Node				Enveloped Displacement		
	Node	Disp_X (in.)	Disp_Y (in.)	Disp_Z (in.)	Ux (in.)	Uy (in.)	Uz (in.)
Basemat @ EL25'	1	0.0044	0.0025	0.0060	0.0044	0.0068	0.0077
	2	0.0034	0.0037	0.0064			
	3	0.0028	0.0049	0.0069			
	4	0.0007	0.0008	0.0012			
	5	0.0033	0.0068	0.0077			
	6	0.0040	0.0027	0.0071			
	7	0.0032	0.0032	0.0066			
NPM Base Supports @ EL26'	8	0.0015	0.0021	0.0039	0.0025	0.0044	0.0078
	9	0.0018	0.0032	0.0054			
	10	0.0023	0.0044	0.0072			
	11	0.0017	0.0016	0.0040			
	12	0.0021	0.0026	0.0052			
	13	0.0025	0.0041	0.0078			
Floor @ EL55'	14	0.0336	0.0368	0.0246	0.0646	0.0571	0.0632
	15	0.0400	0.0571	0.0535			
	16	0.0377	0.0433	0.0632			
	17	0.0349	0.0311	0.0502			
	18	0.0271	0.0403	0.0242			
	19	0.0646	0.0335	0.0468			
	20	0.0288	0.0392	0.0272			
	21	0.0360	0.0423	0.0474			
	22	0.0381	0.0369	0.0607			
	23	0.0342	0.0307	0.0458			
Floor @ EL70'	24	0.0559	0.0550	0.0301	0.0635	0.0678	0.0772
	25	0.0590	0.0678	0.0703			
	26	0.0555	0.0515	0.0496			
	27	0.0489	0.0446	0.0615			
	28	0.0461	0.0570	0.0329			
	29	0.0635	0.0456	0.0503			
	30	0.0463	0.0556	0.0361			
	31	0.0550	0.0524	0.0772			
	32	0.0572	0.0483	0.0441			
	33	0.0500	0.0425	0.0561			
Floor @ EL85'	34	0.0800	0.0739	0.0399	0.0821	0.0808	0.2993
	35	0.0753	0.0808	0.2364			
	36	0.0719	0.0613	0.2993			
	37	0.0672	0.0616	0.0653			
	38	0.0614	0.0625	0.0623			
	39	0.0701	0.0717	0.0430			
	40	0.0755	0.0696	0.2554			
	41	0.0821	0.0653	0.2647			
	42	0.0731	0.0603	0.0669			

Table 3.7.2-14b: Relative Displacement for RXB Floors due to Certified Seismic Design Response Spectra - High Frequency (Continued)

Group	Max Displacement @ Each Node				Enveloped Displacement		
	Node	Disp_X (in.)	Disp_Y (in.)	Disp_Z (in.)	Ux (in.)	Uy (in.)	Uz (in.)
Floor @ EL100'	43	0.0977	0.0892	0.0430	0.1050	0.1136	0.3427
	44	0.0920	0.0903	0.1935			
	45	0.0914	0.0799	0.3427			
	46	0.0874	0.0811	0.0726			
	47	0.0939	0.0834	0.0659			
	48	0.0925	0.0865	0.0565			
	49	0.0952	0.0835	0.0770			
	50	0.0960	0.0842	0.0544			
	51	0.1002	0.1136	0.1870			
	52	0.1050	0.0865	0.2898			
	53	0.1013	0.0826	0.0761			
Floor @ EL126'	54	0.1427	0.1384	0.0906	0.1495	0.1384	0.0906
	55	0.1495	0.1288	0.0906			
Floor @ EL145'	56	0.1737	0.1836	0.1122	0.2567	0.4800	0.2624
	57	0.1808	0.1933	0.0887			
	58	0.2567	0.3492	0.0900			
	59	0.2503	0.4800	0.2624			
Roof @ EL187'	60	0.2486	0.5125	0.2951	0.2578	0.5125	0.2951
	61	0.2517	0.4414	0.2524			
	62	0.2578	0.2878	0.1217			
	63	0.1674	0.2819	0.1053			
	64	0.1694	0.2886	0.1044			

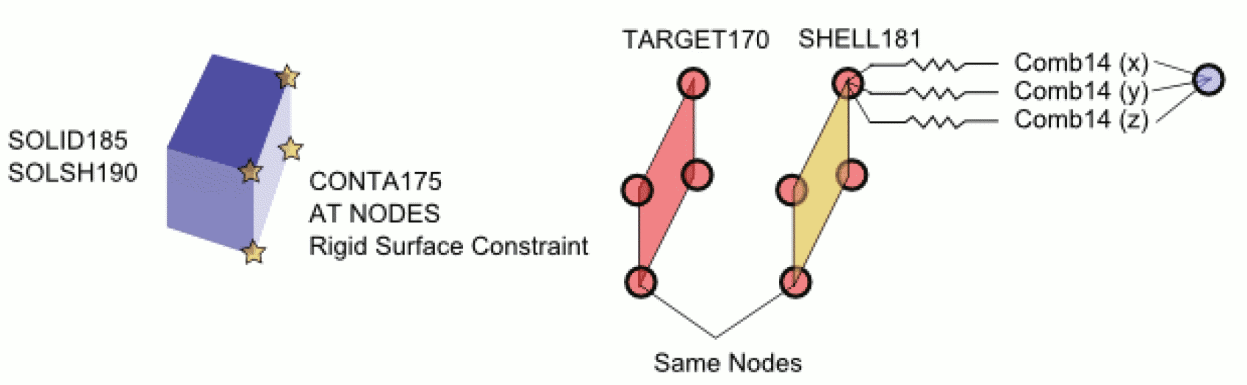
Table 3.7.2-14c: Relative Displacements at CRB Floors Due to Certified Seismic Design Response Spectra Input Motions

	Envelope of S7 and S11		
Location	**X (in.)**	**Y (in.)**	**Z (in.)**
floor at EL 100' (basemat)	0.013	0.022	0.376
floor at EL 123'	0.088	0.346	0.460
floor at EL 150'-3" (roof)	0.171	0.690	0.792

Table 3.7.2-14d: Relative Displacements at CRB Floors Due to Certified Seismic Design Response Spectra - High Frequency Input Motions

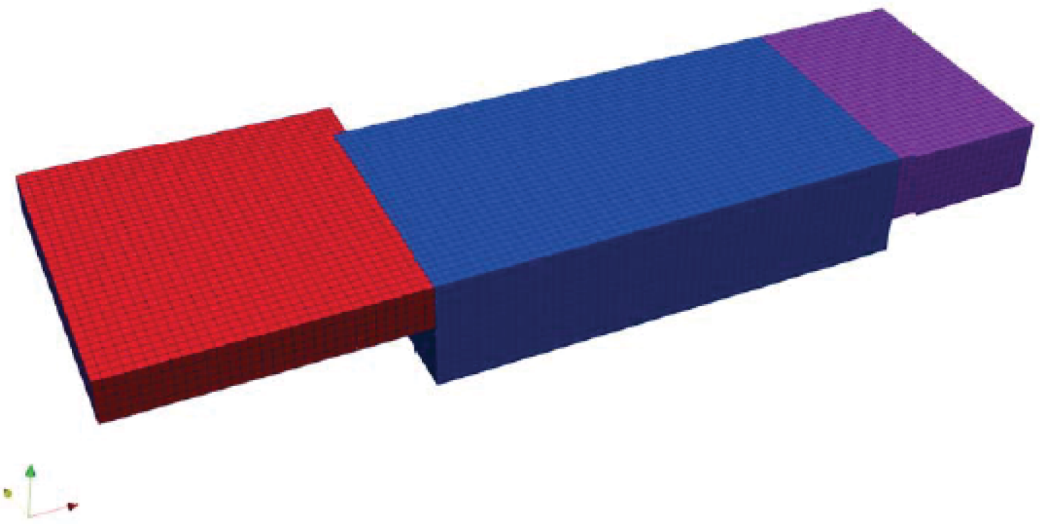
	Envelope of S9		
Location	**X (in.)**	**Y (in.)**	**Z (in.)**
floors at EL 100' (basemat)	0.003	0.003	0.029
floors at EL 123'	0.052	0.110	0.223
floors at EL 150'-3" (roof)	0.083	0.169	0.254

Figure 3.7.2-1: Soil-Structure Interaction Interface Model Representation



Note: Blue Node is a Master Node for the Soil Library Substructure

Figure 3.7.2-2: Double-Building Excavated Soil Model



Note: RXB excavated soil shown in blue. RWB excavated soil shown in red

Figure 3.7.2-3: Finite Element Mesh of Excavated Soil and Putback Soil under the Control Building

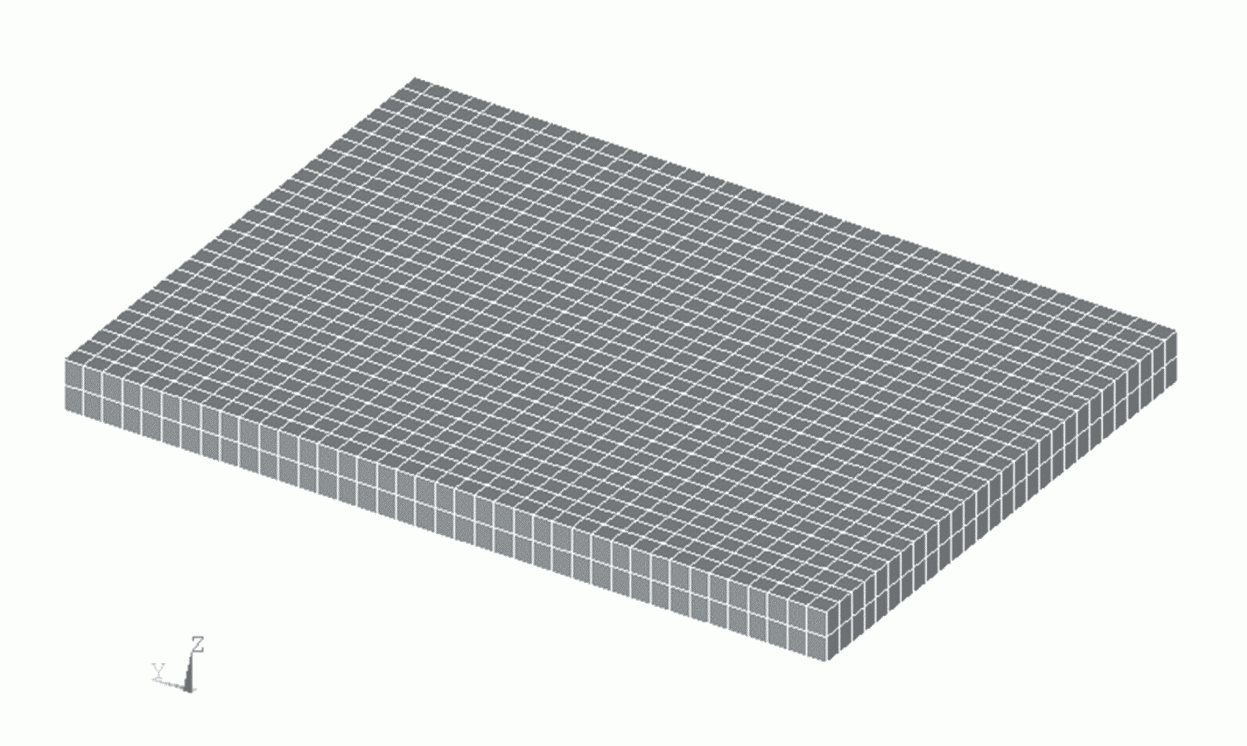


Figure 3.7.2-4: Combined Control Building plus Putback Soil Model

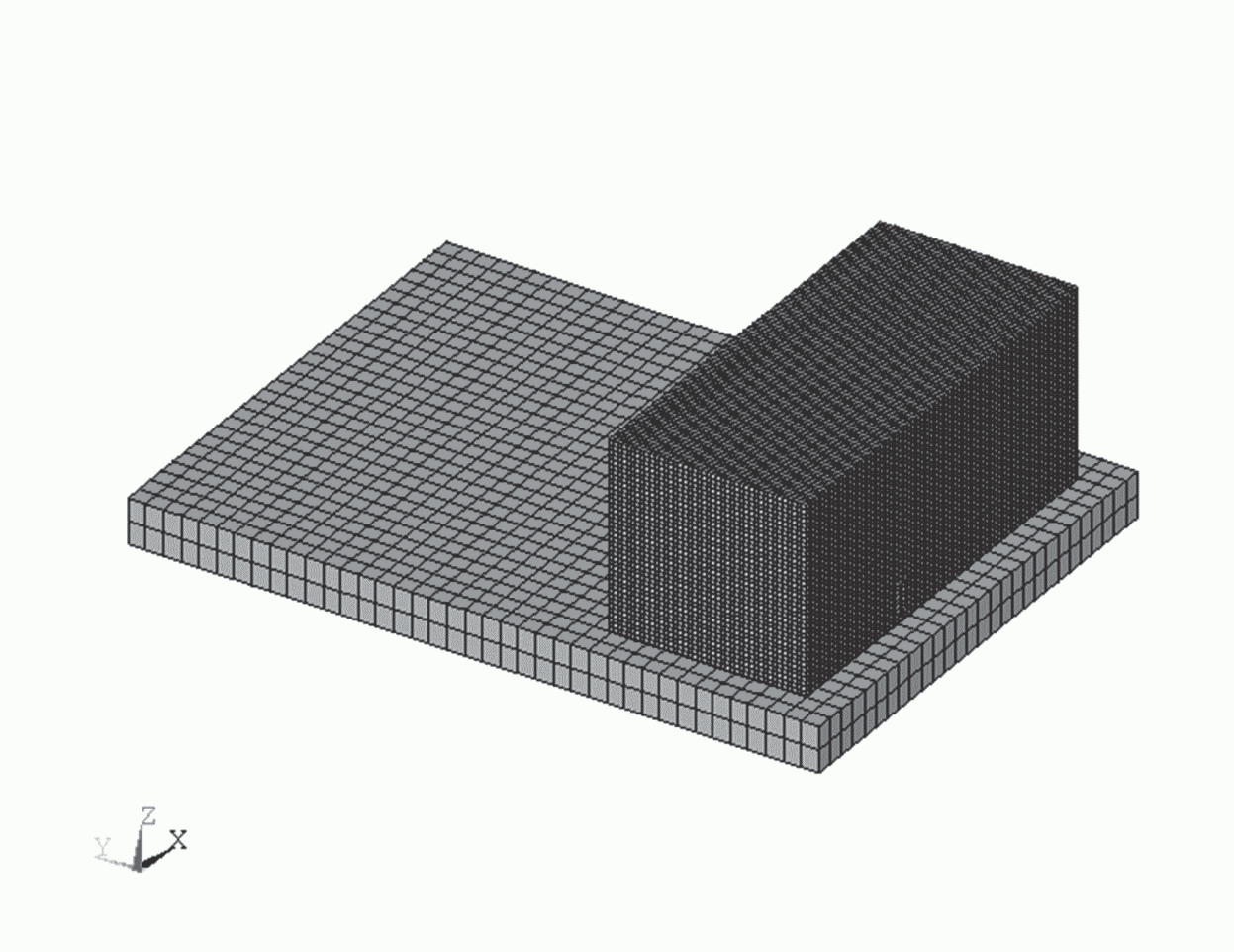


Figure 3.7.2-5: NuScale Power Module Number Location

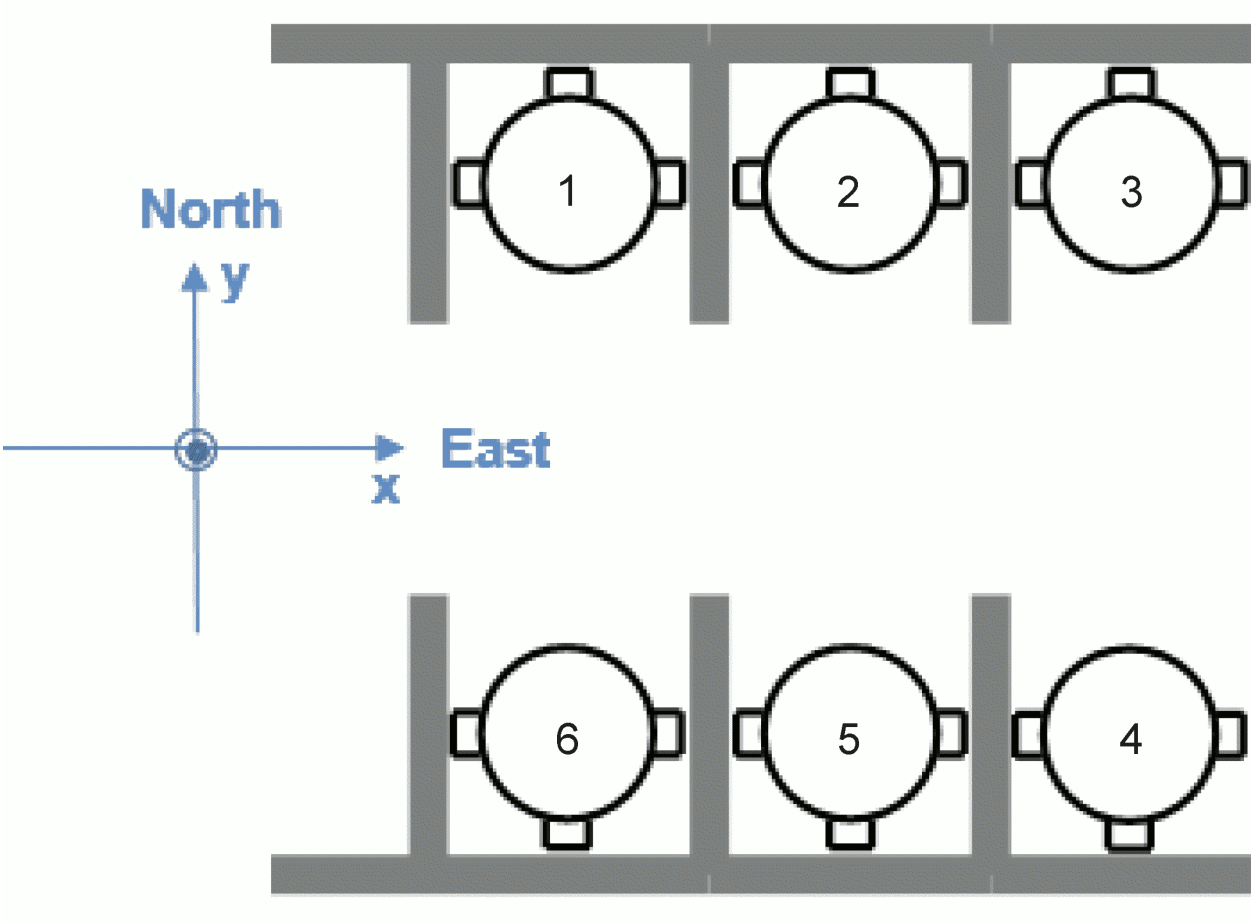


Figure 3.7.2-6: Combined NuScale Power Module Model

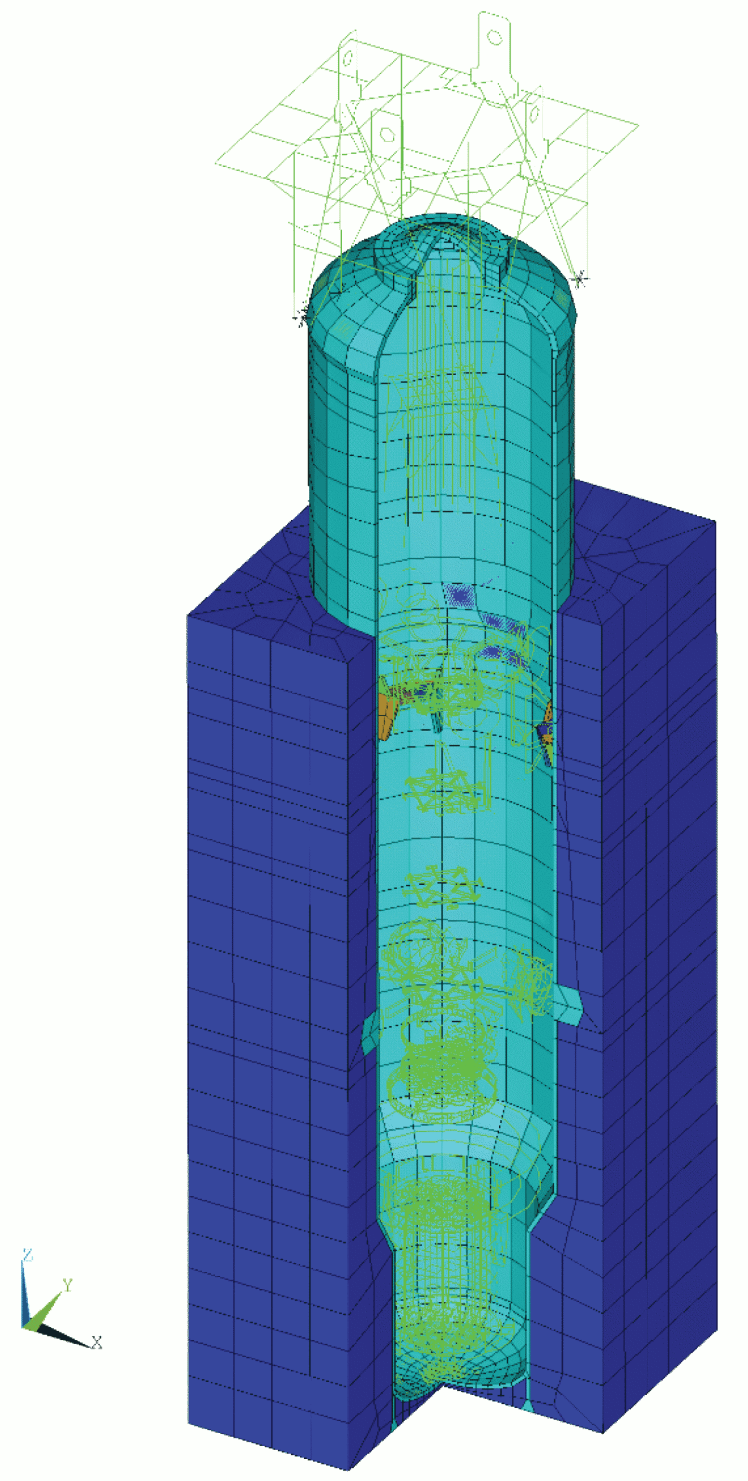


Figure 3.7.2-7: NuScale Power Module Lug and Reactor Building Corbel Connections

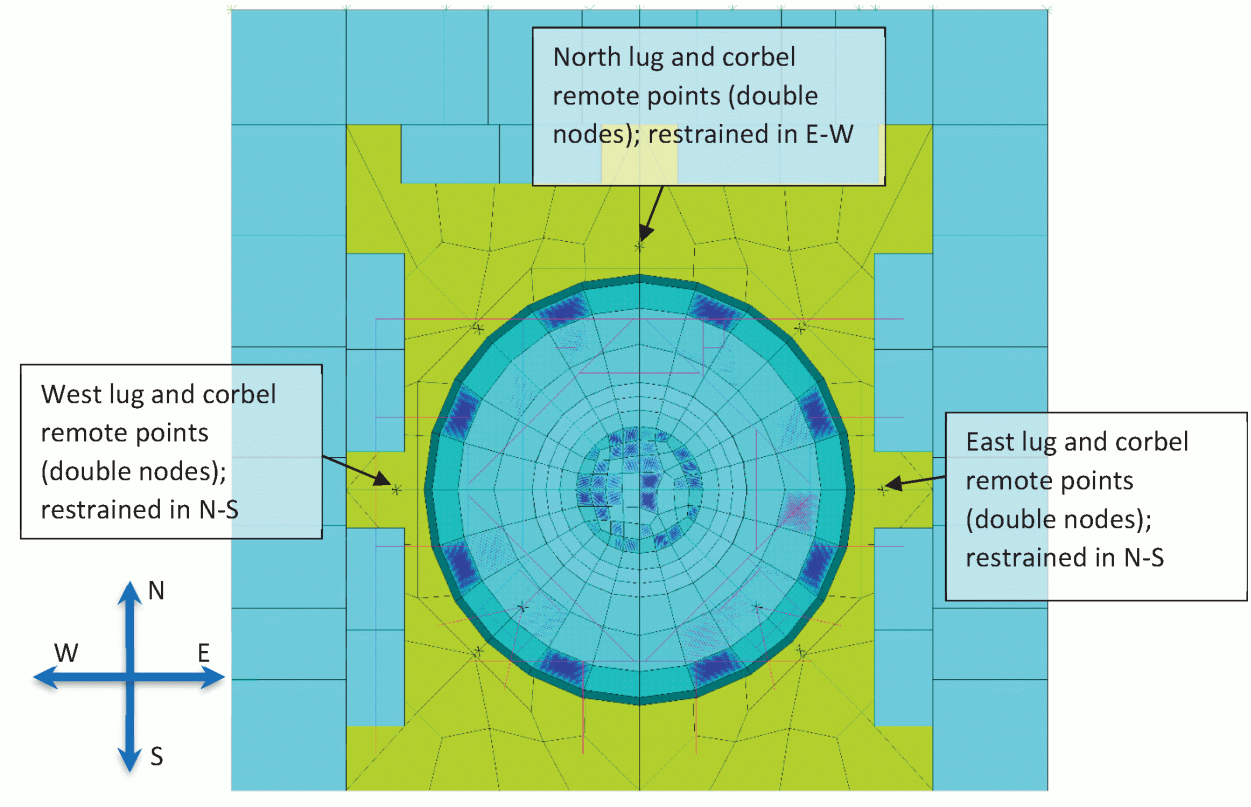


Figure 3.7.2-8: Reactor Building Crane Model inserted in RXB

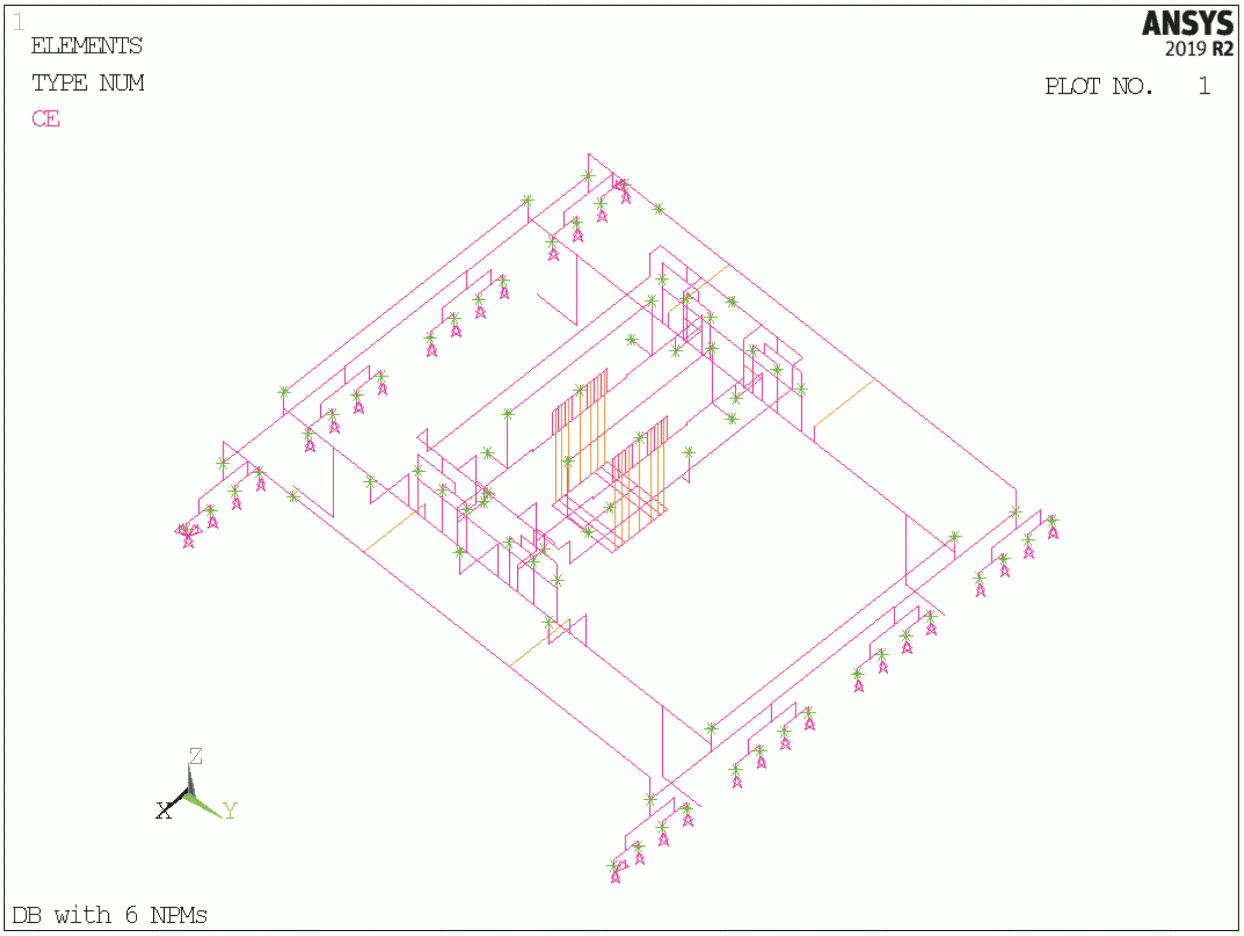


Figure 3.7.2-9: Crane Location in the Reactor Building Model

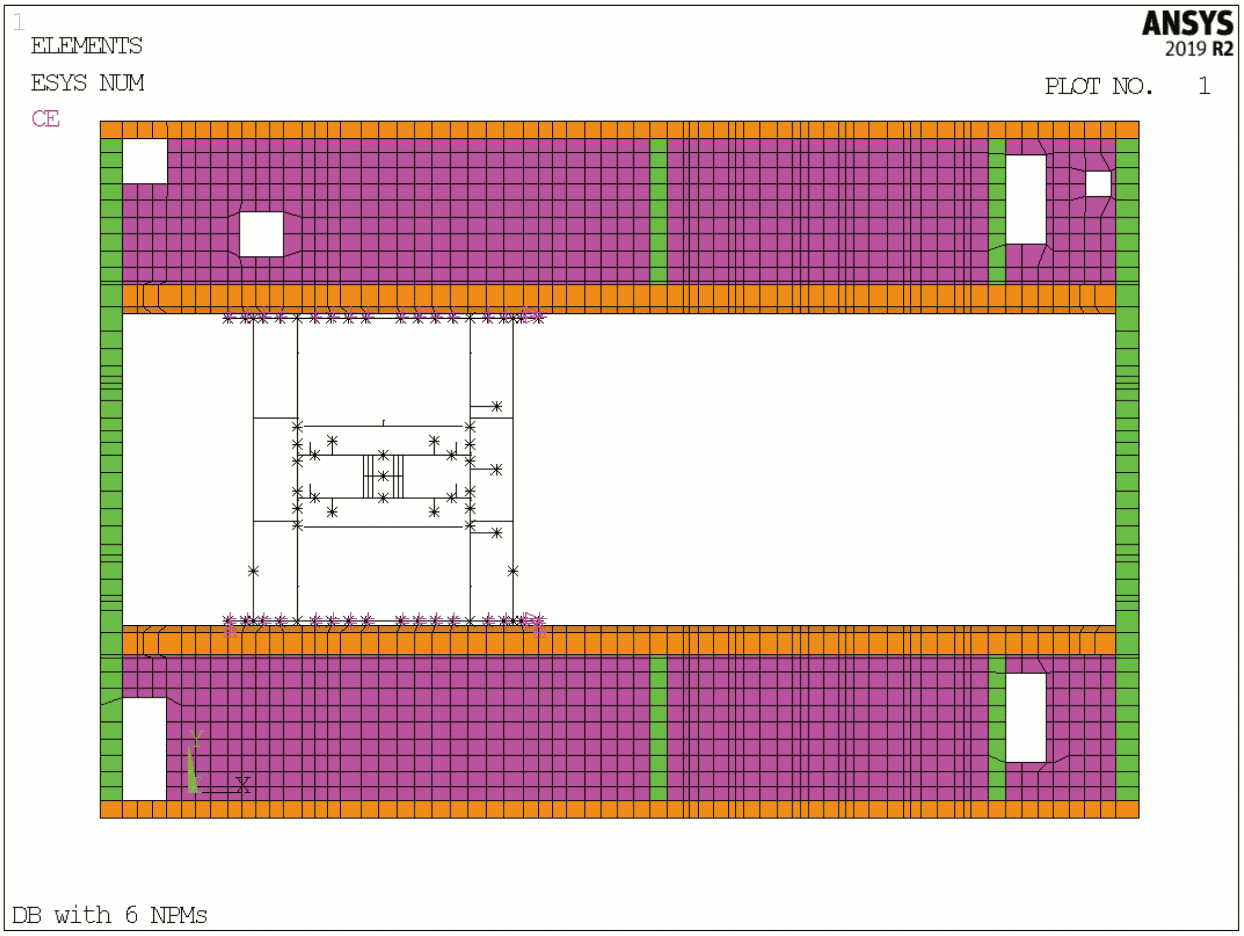


Figure 3.7.2-10: Pressure Field in the Pool (a) in psi and Displacement Field on Containment Vessels (b) in inches Due to 1g Acceleration Applied in Z-Direction

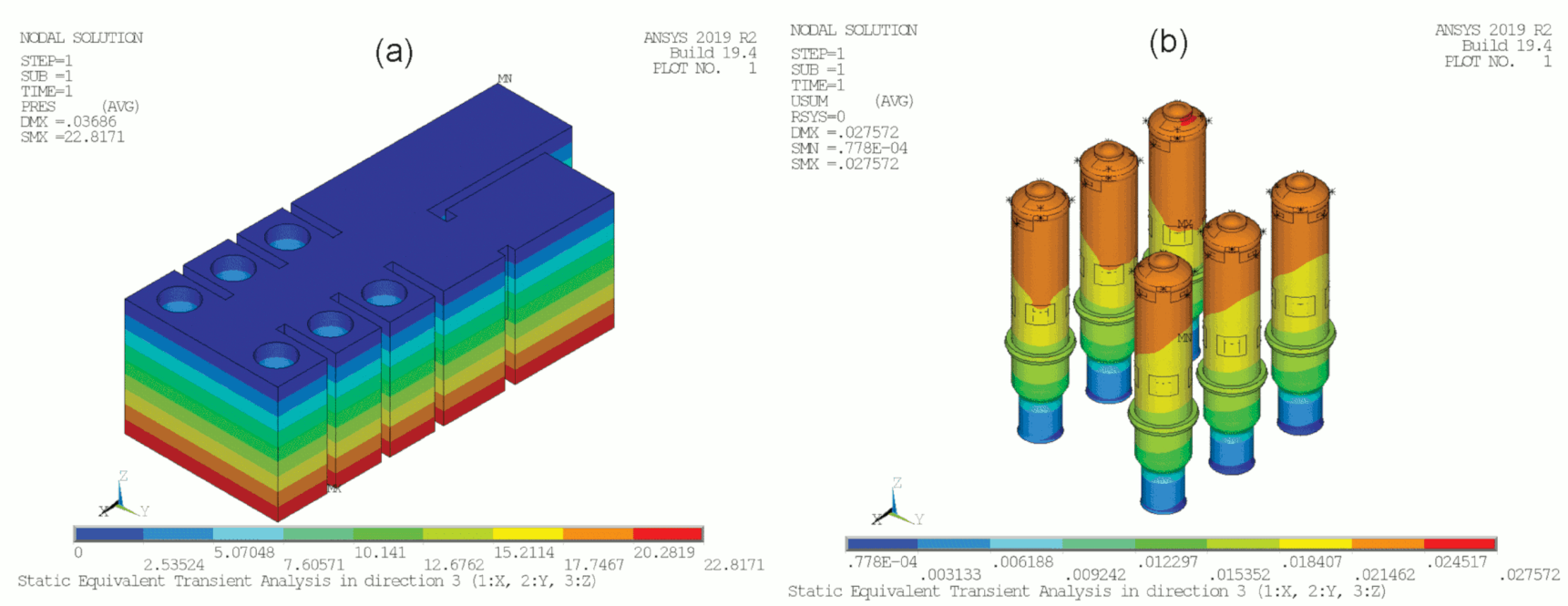


Figure 3.7.2-11: Control Building Plan View at Elevation 100’

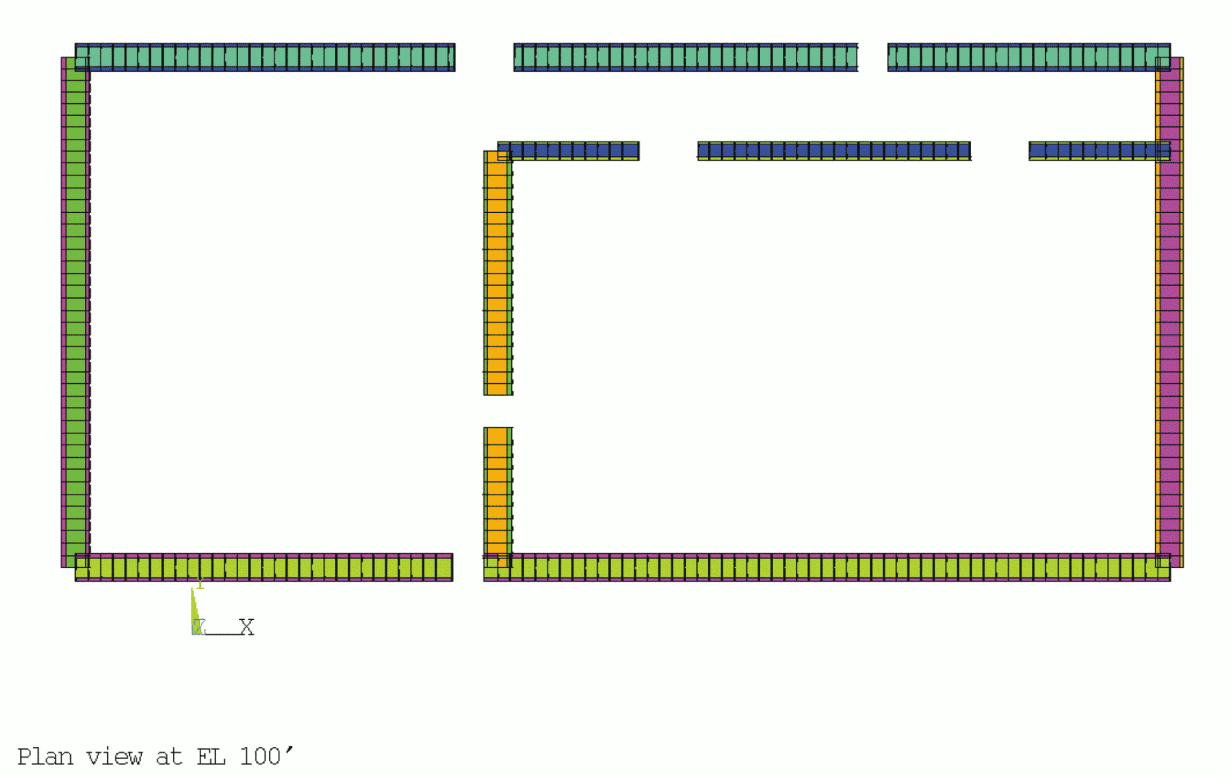


Figure 3.7.2-12: Control Building Plan View at Elevation 123'

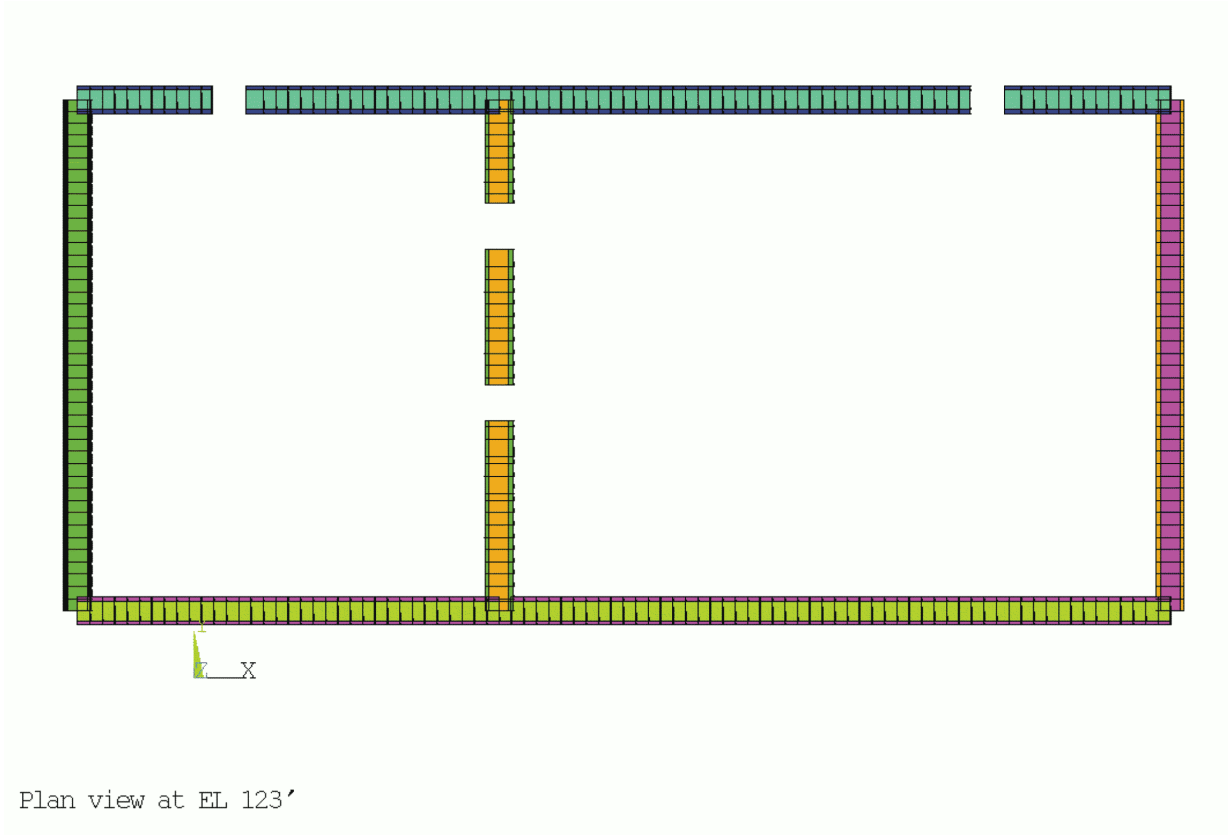


Figure 3.7.2-13: Control Building Side View (North Wall Looking North)

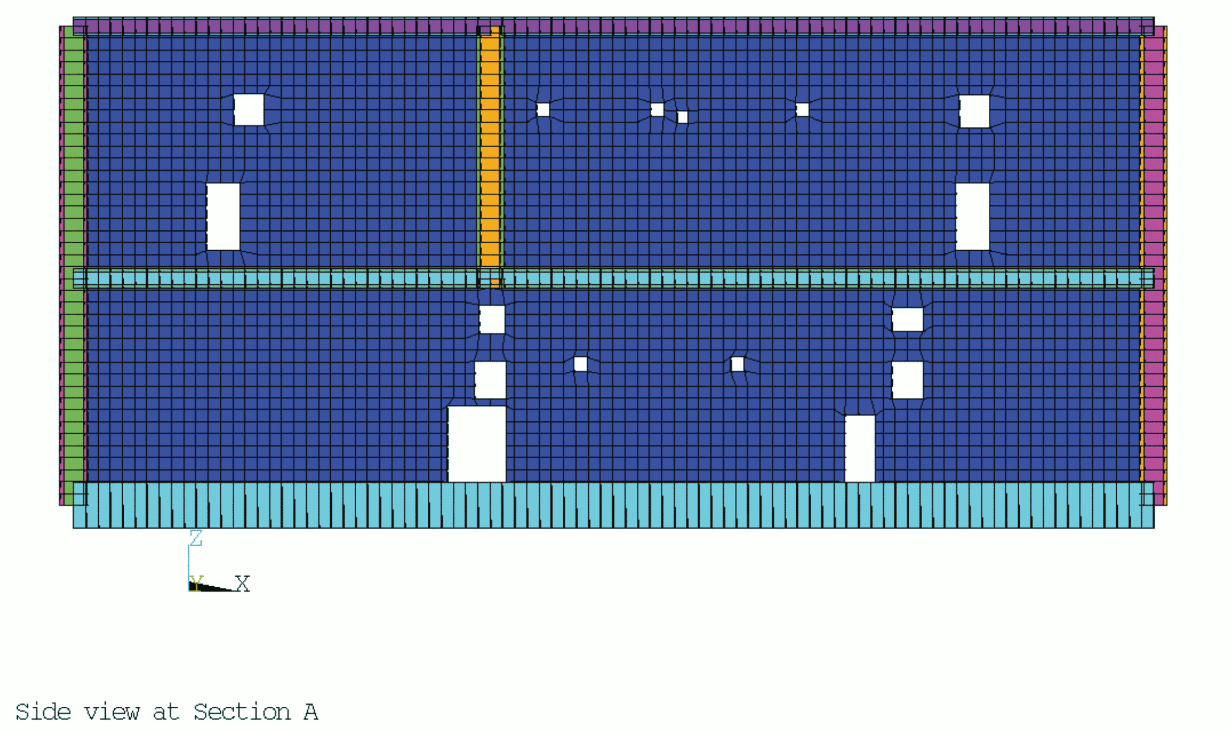


Figure 3.7.2-14: Control Building Side View (Wall at Gridline G Looking North)

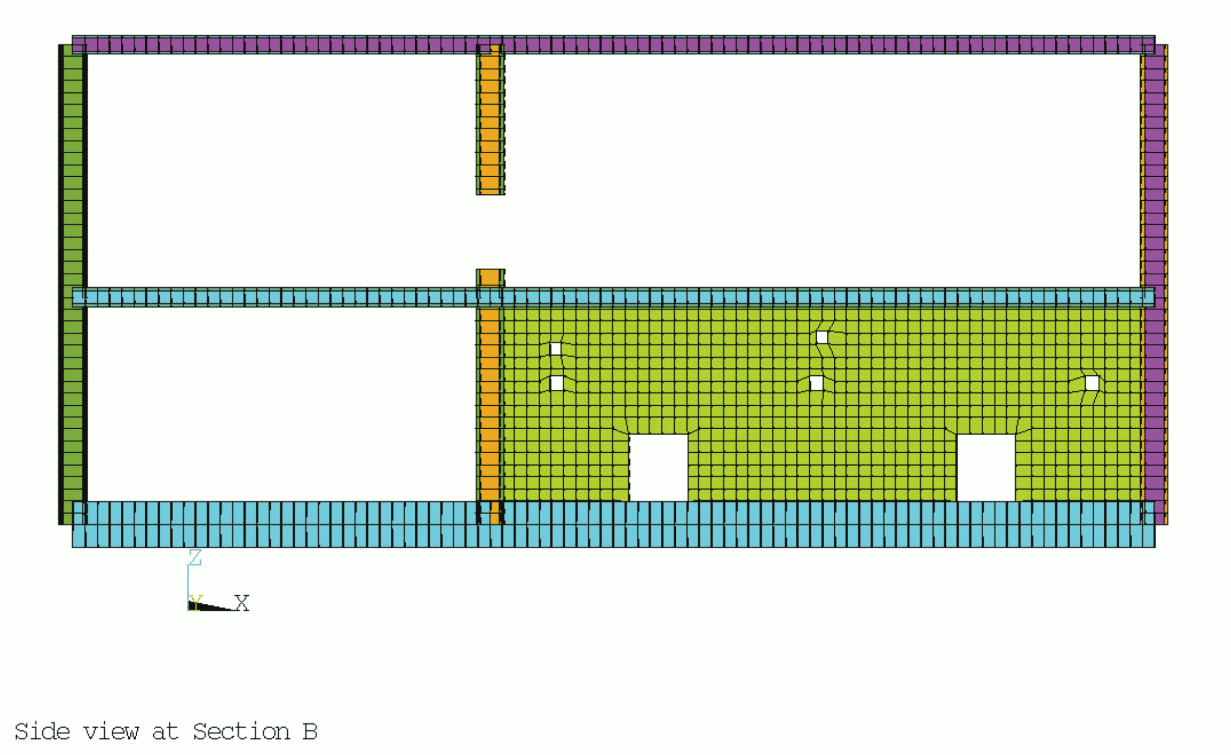


Figure 3.7.2-15: Control Building Side View (South Wall Looking North)

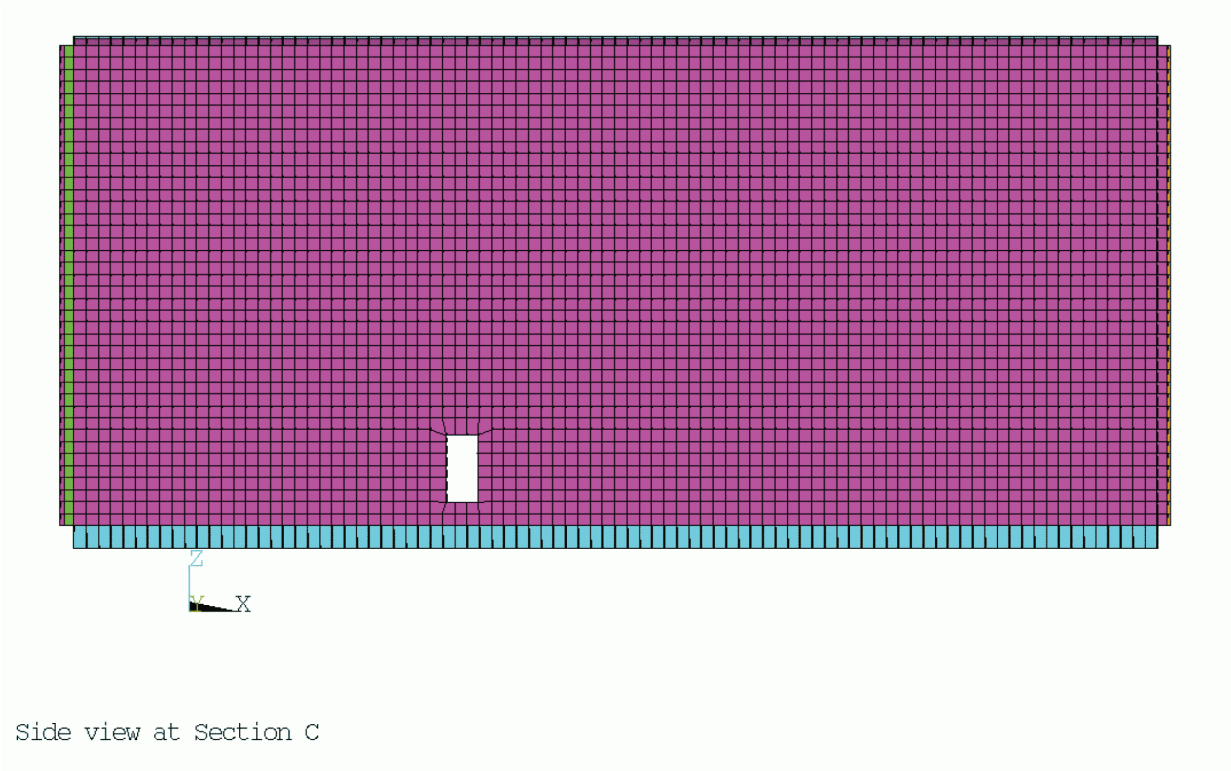


Figure 3.7.2-16: Control Building Side View (Wall at Gridline 3 Looking West)

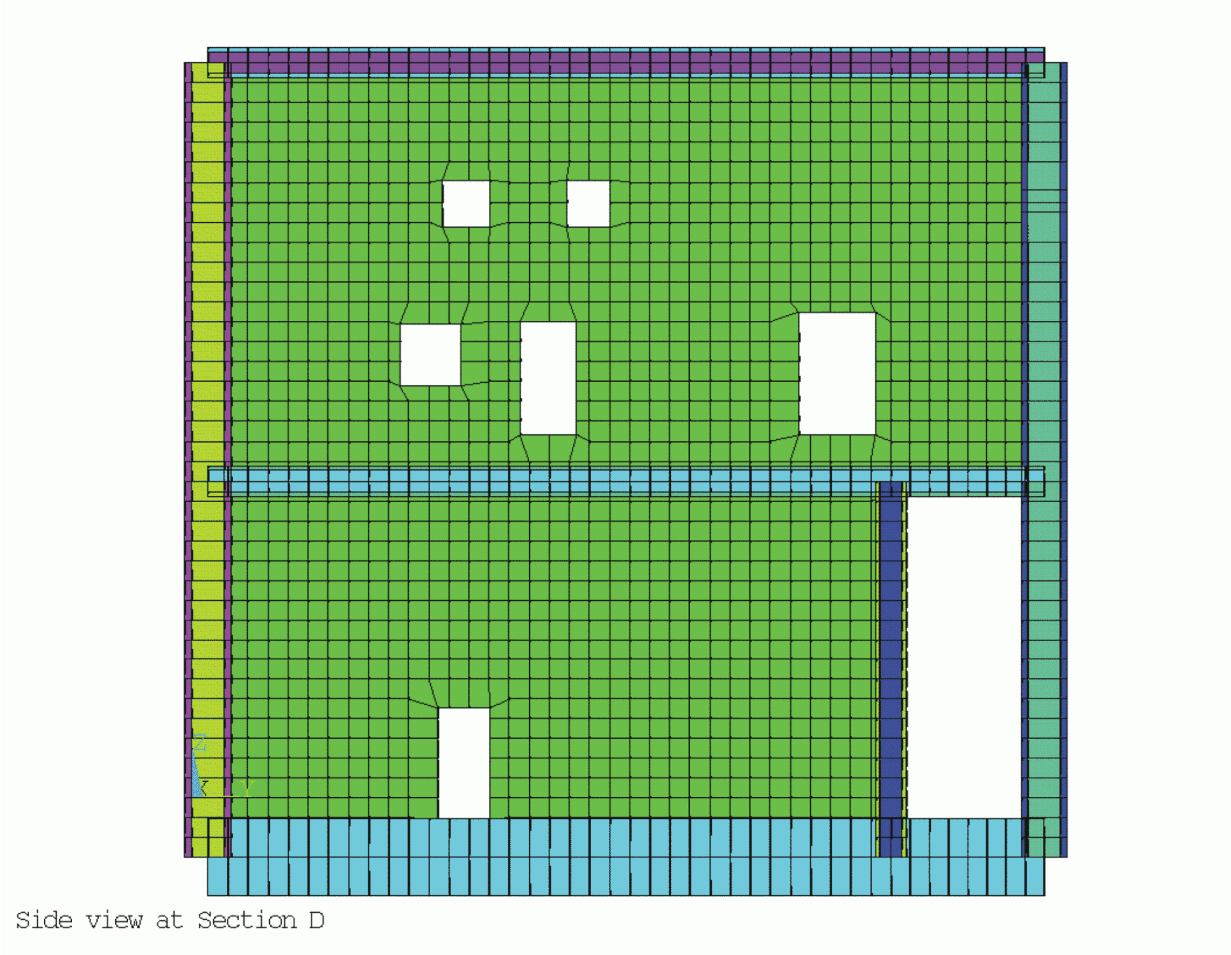


Figure 3.7.2-17: Double Building Modal Analysis Mass Participation Ratios in X, Y, and Z Directions

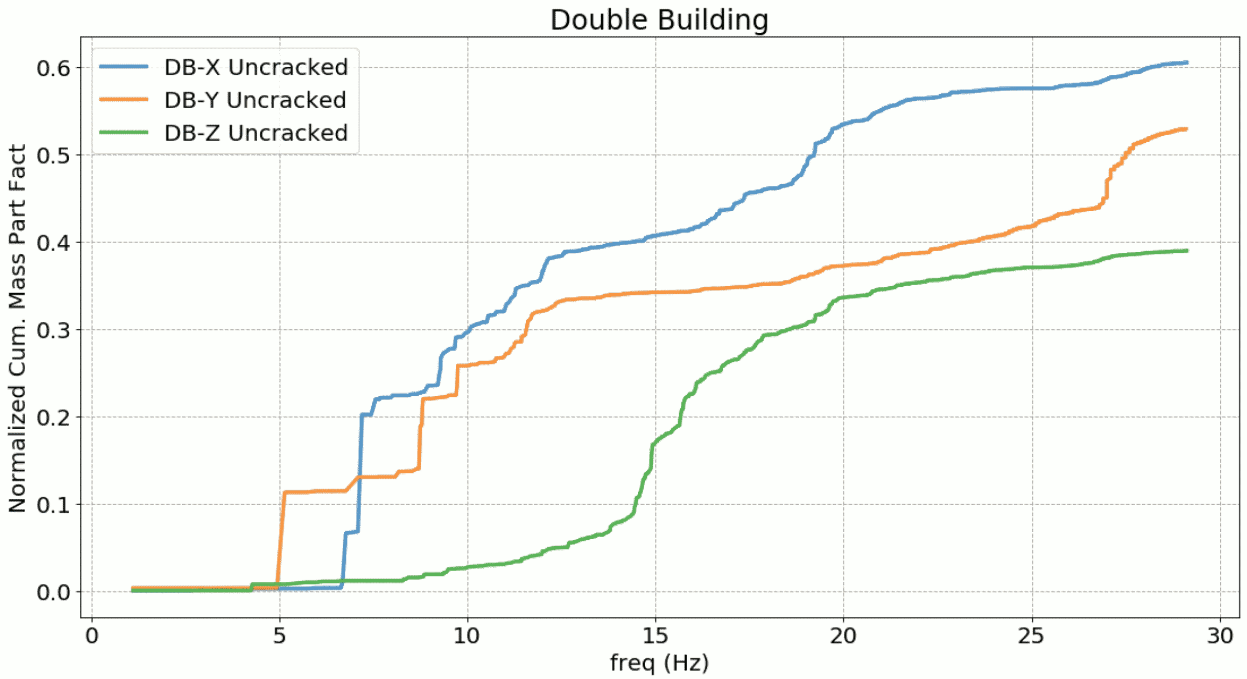
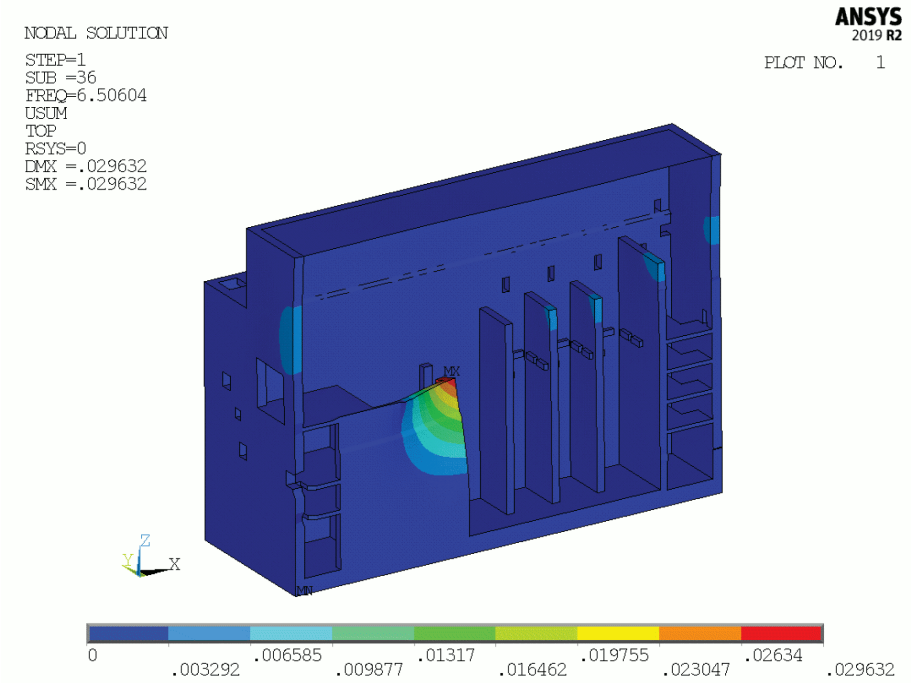
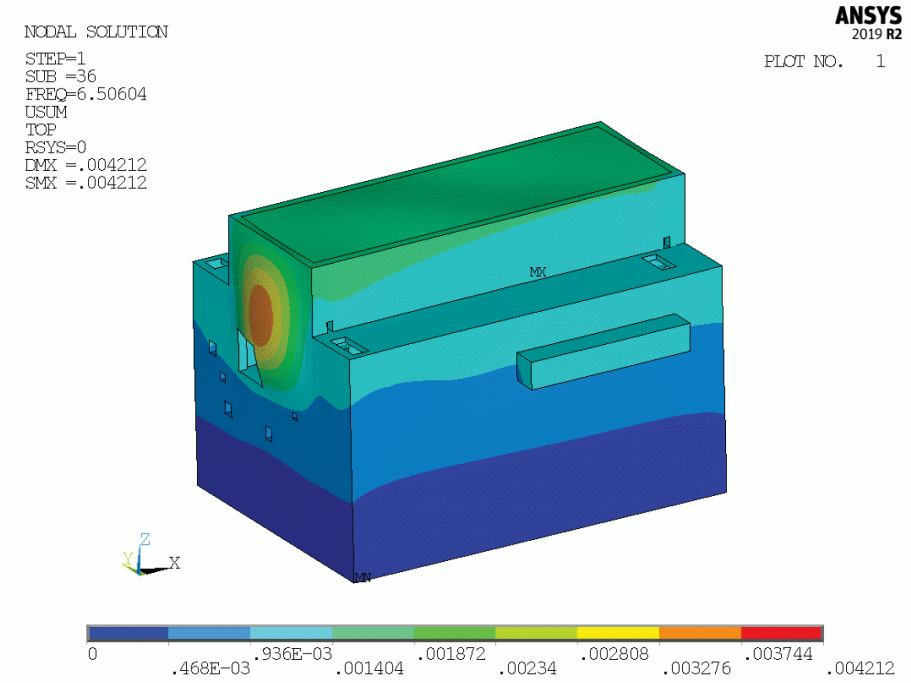


Figure 3.7.2-18a: Reactor Building ANSYS Model
(X Direction, Mode 36 - Frequency: 6.51 Hz)



(a) 3D view



(b) 3D view (wall at RX_C unselected)

Figure 3.7.2-18b: Reactor Building ANSYS Model
(Y-Direction, Mode 32, Frequency: 4.51 Hz)

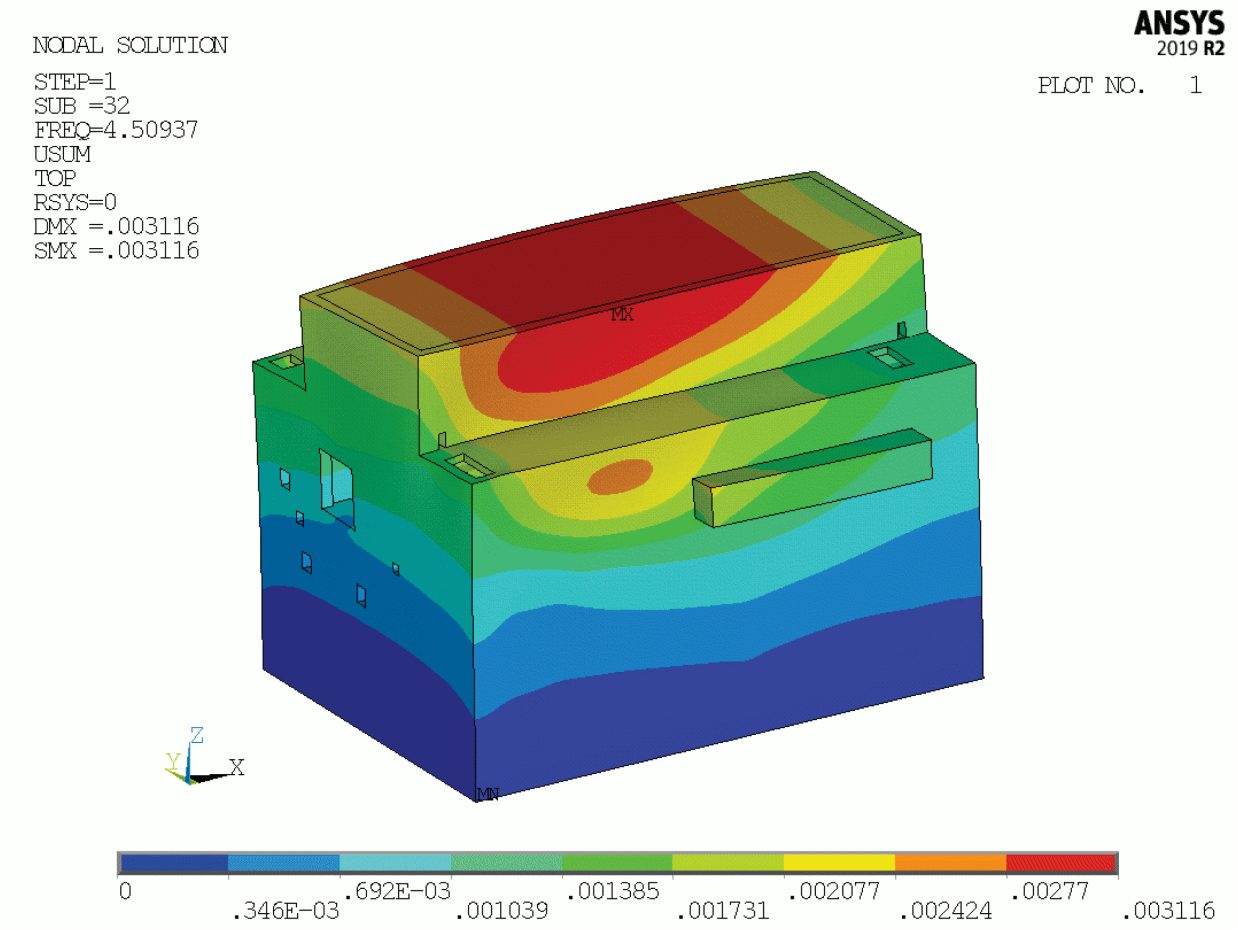


Figure 3.7.2-18c: Reactor Building ANSYS Model (Z Direction, Mode 94 - Frequency: 14.95 Hz)

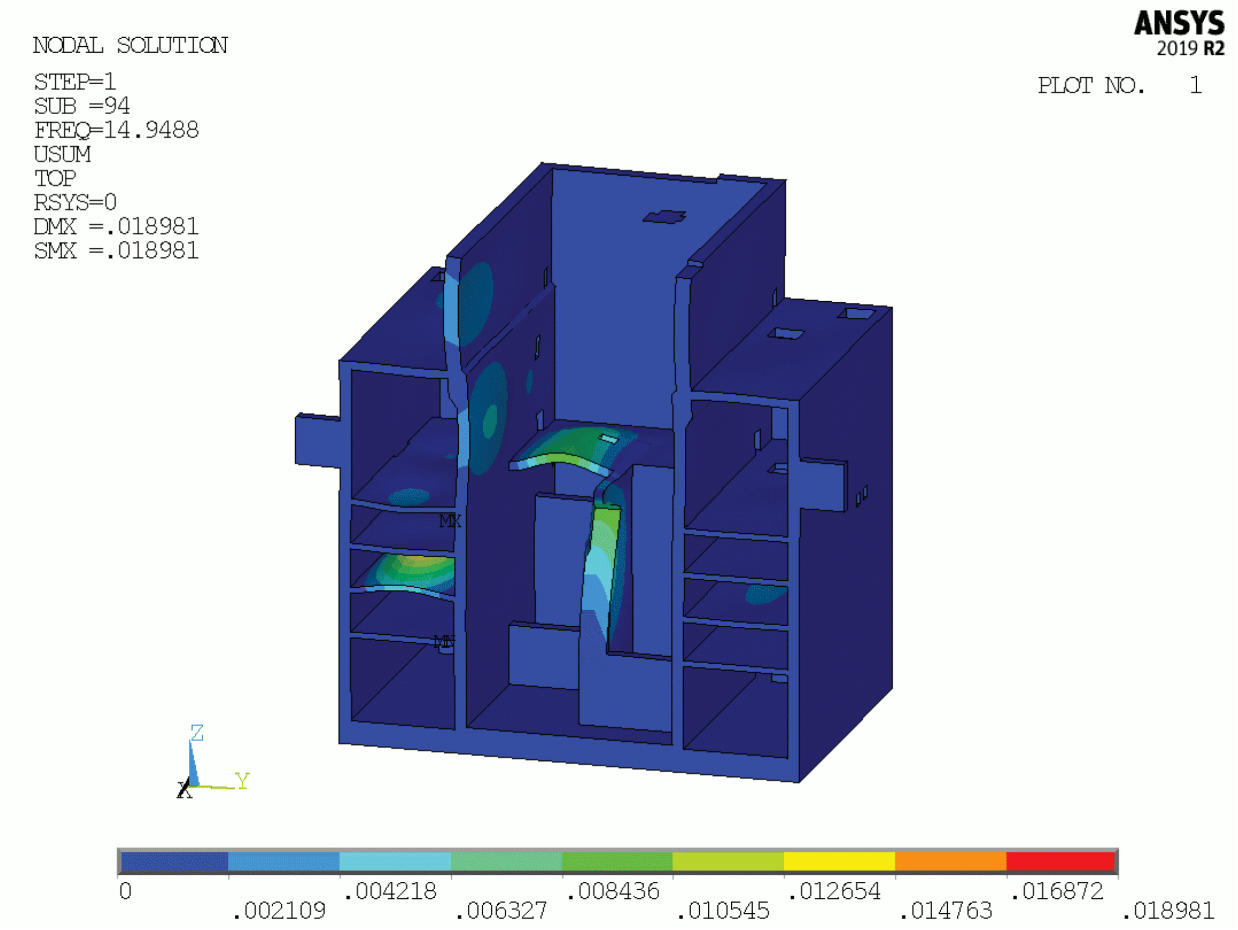


Figure 3.7.2-19: Control Building Modal Mass Participation Ratios in X, Y, and Z Directions

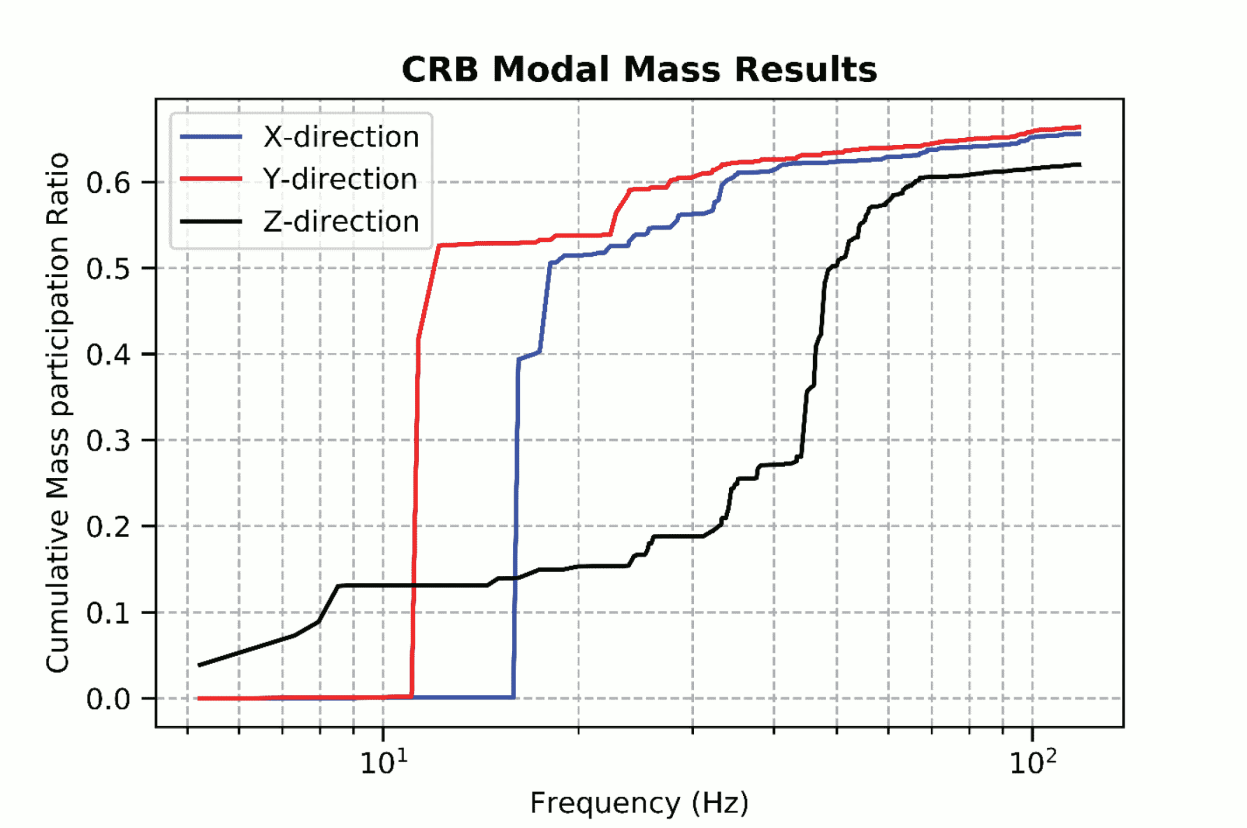


Figure 3.7.2-20: Selected Nodes for Reactor Building Crane Support In-Structure Response Spectra

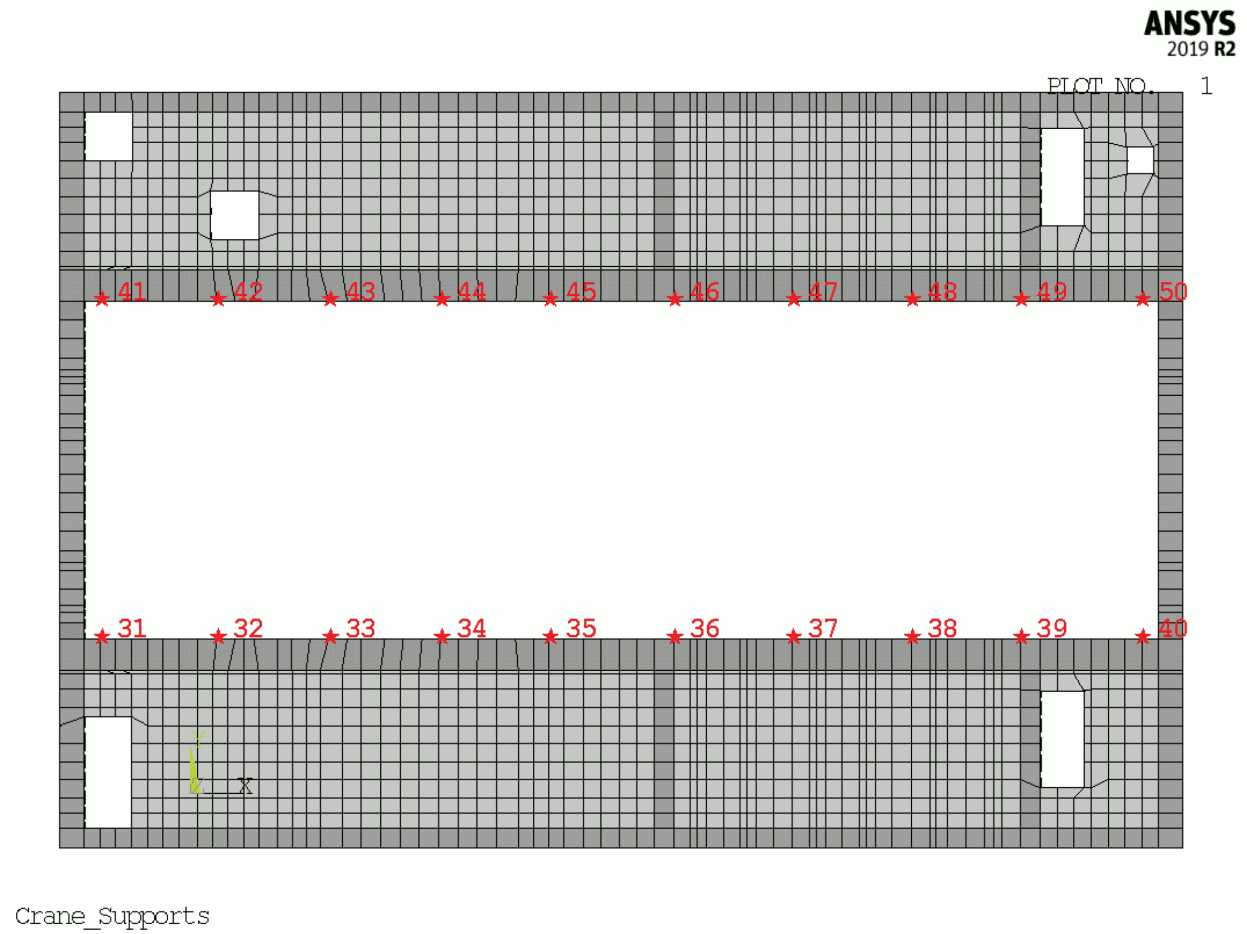


Figure 3.7.2-21: East-West, North-South, and Vertical In-Structure Response Spectra Due to Certified Seismic Design Response Spectra for Reactor Building Crane Supports

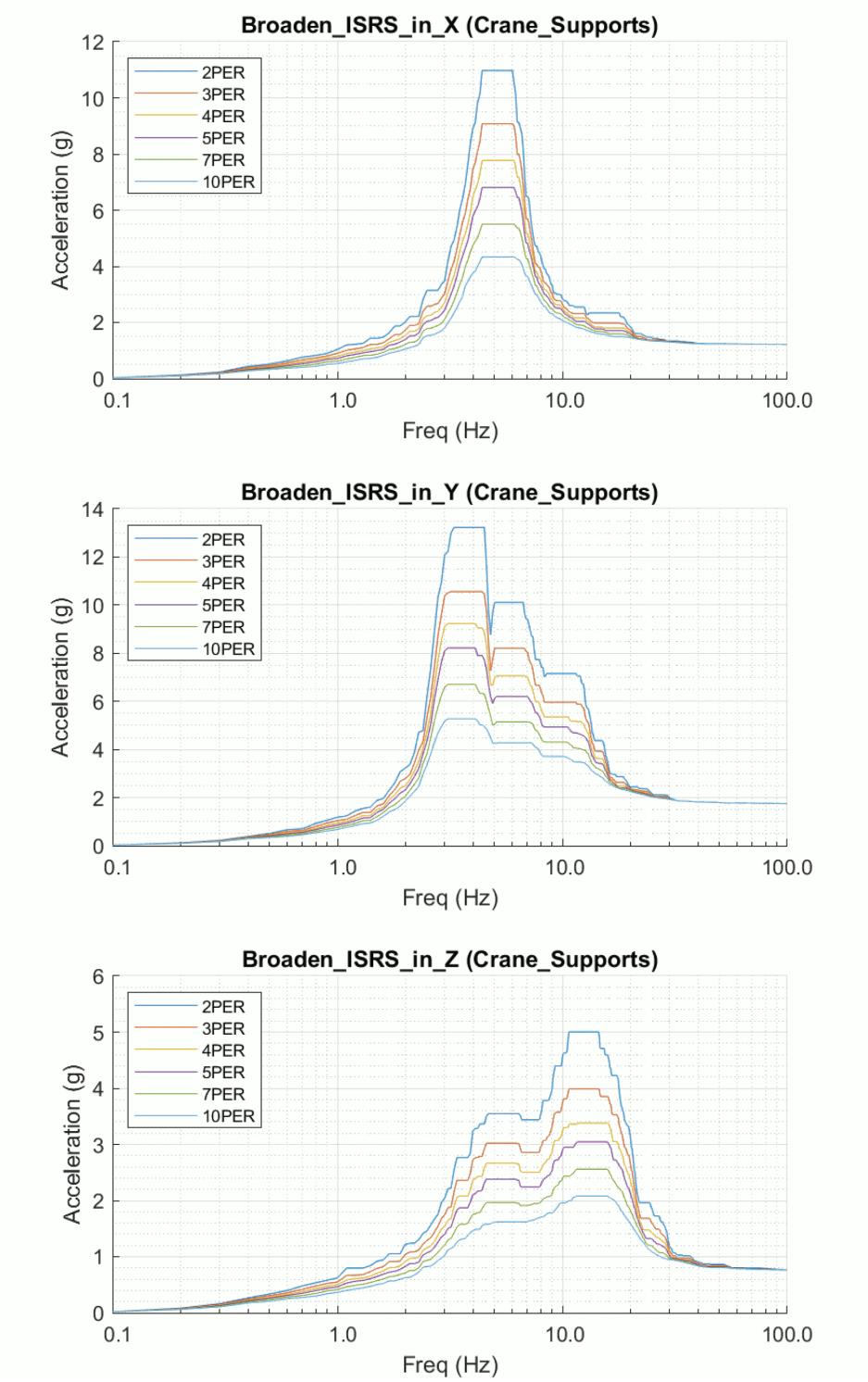


Figure 3.7.2-22: East-West, North-South, and Vertical In-Structure Response Spectra Due to Certified Seismic Design Response Spectra - High Frequency for Reactor Building Crane Supports

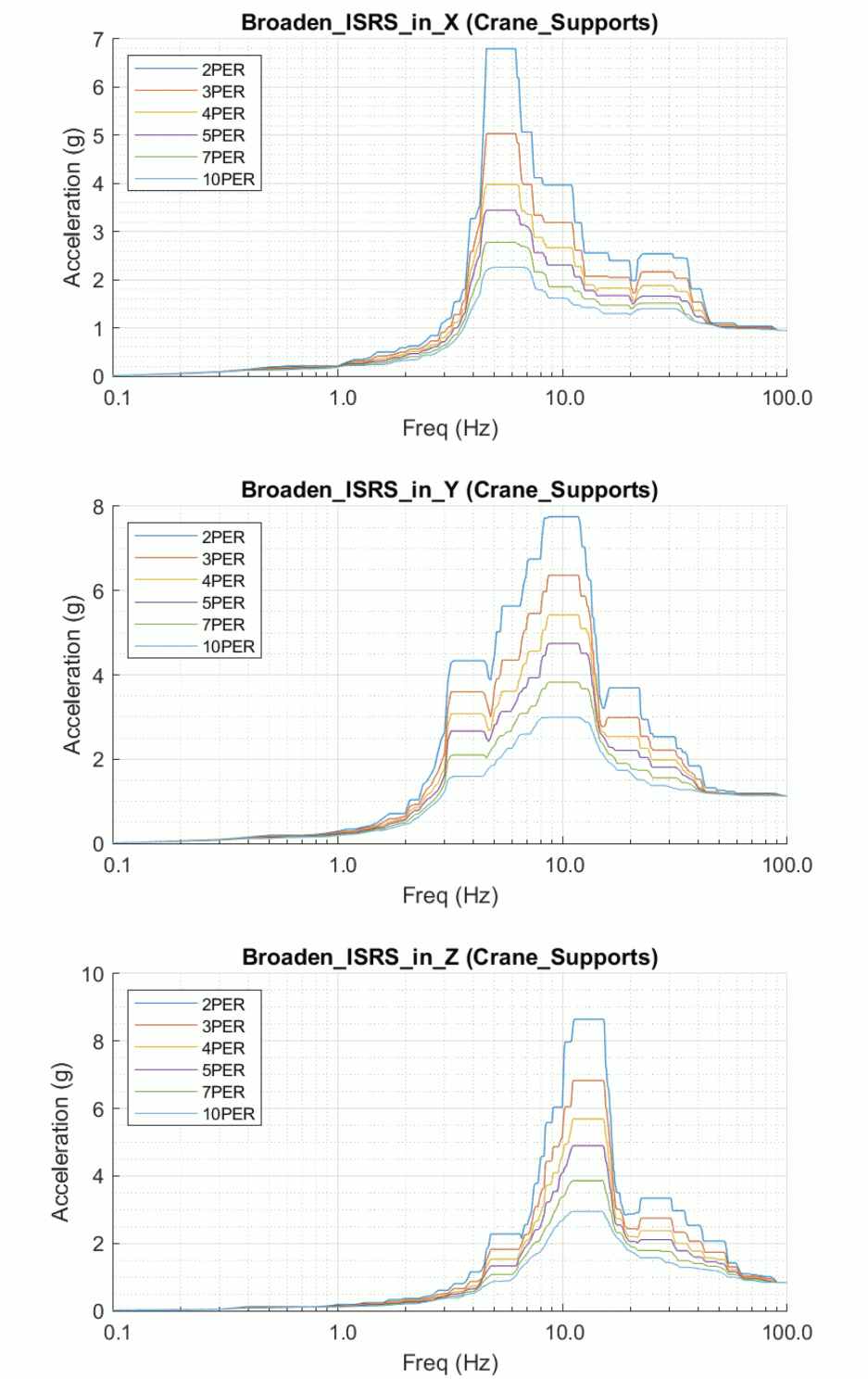


Figure 3.7.2-23: Selected Node at Elevation 26 ft for Reactor Building Sensitivity Studies

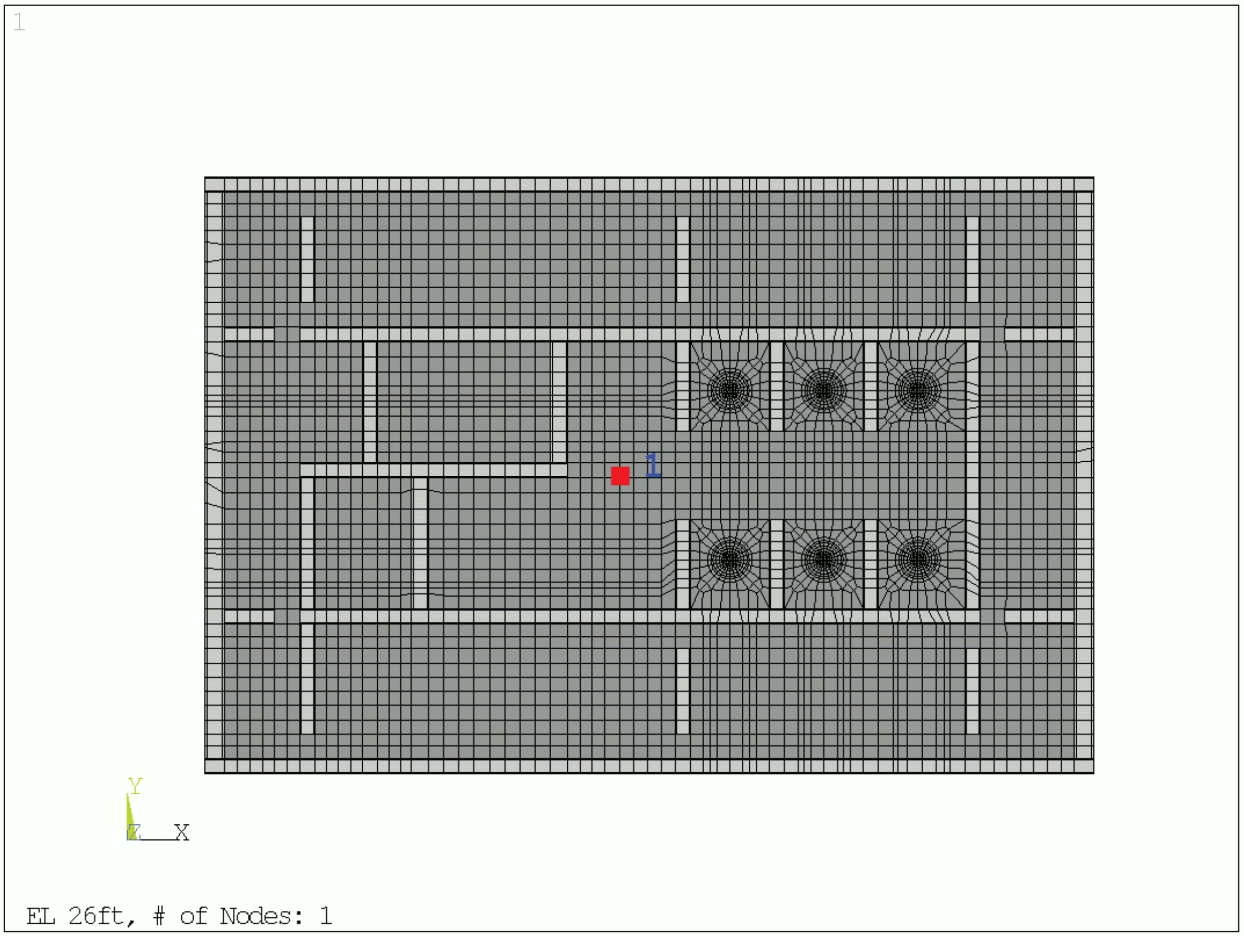
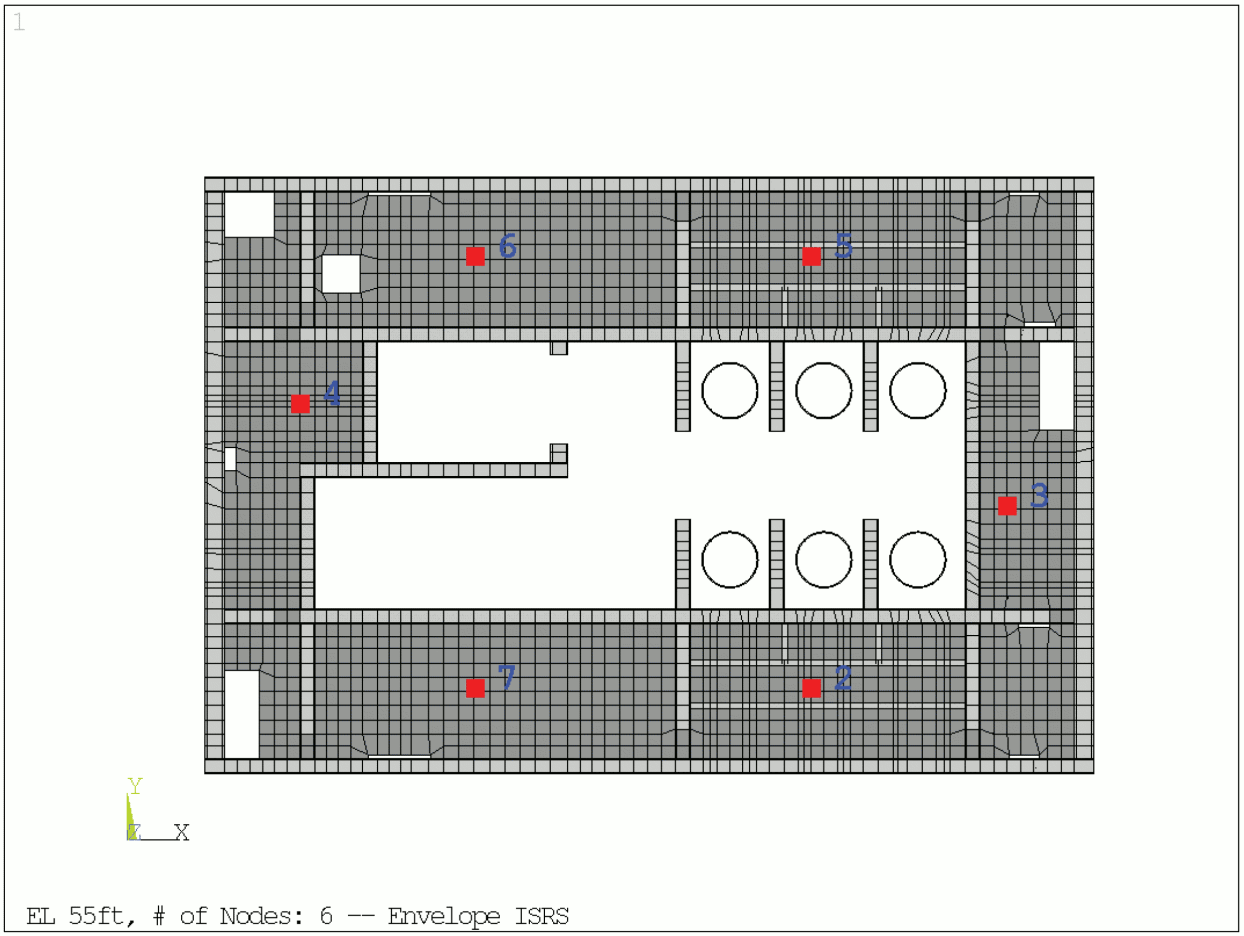
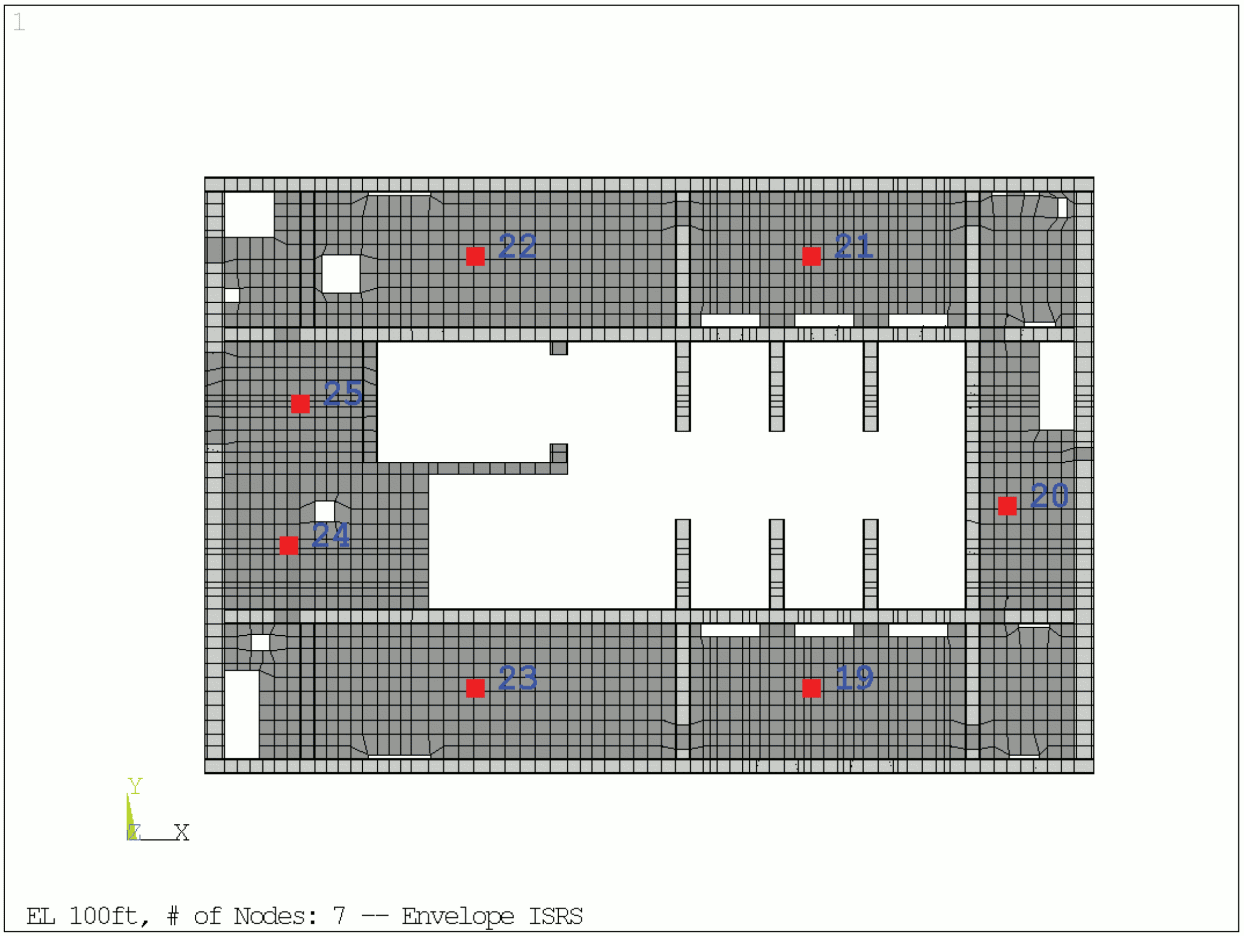


Figure 3.7.2-24: Selected Nodes at Elevation 55 ft for Reactor Building Sensitivity Studies



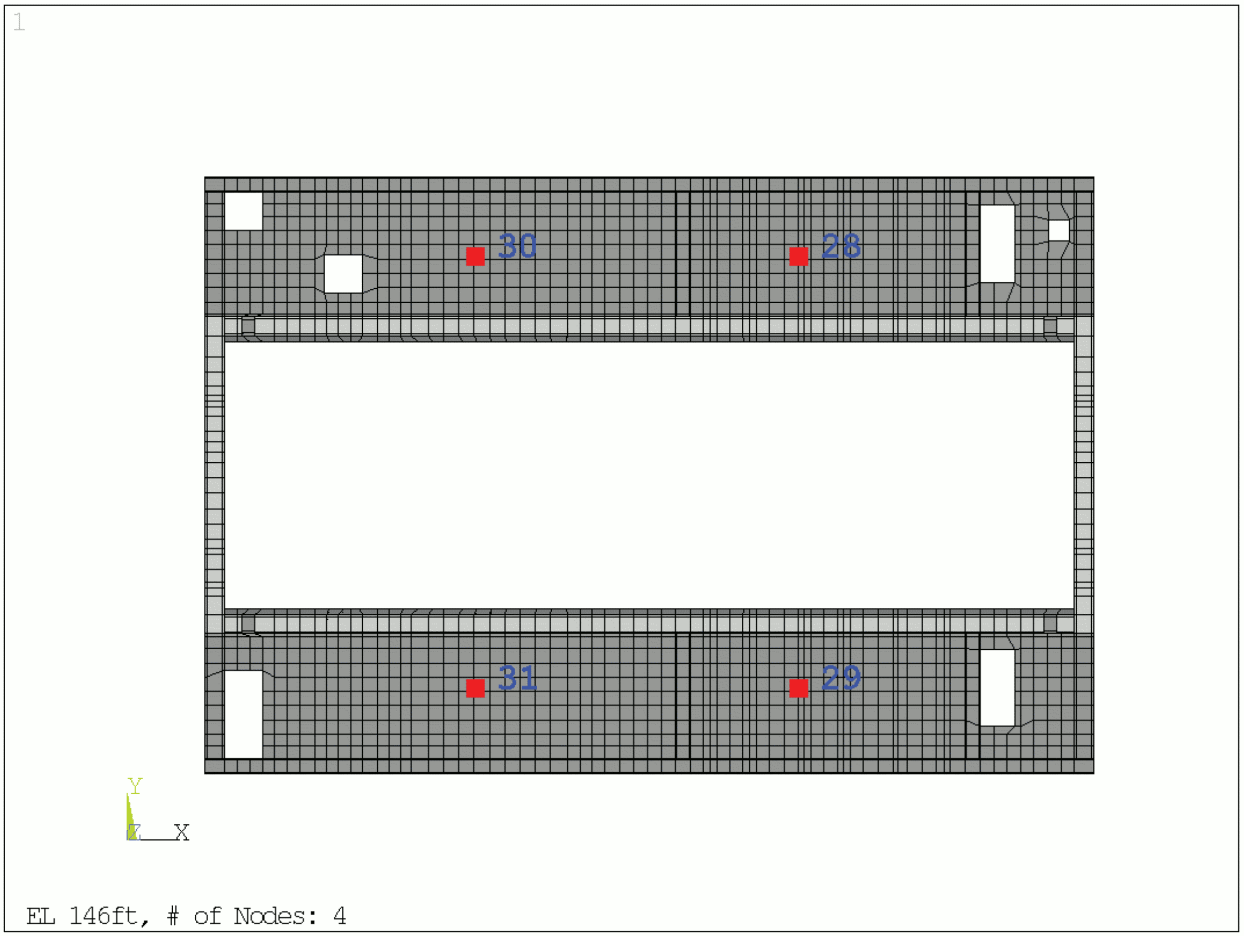
Note: The ISRS calculated at these nodes are enveloped.

Figure 3.7.2-25: Selected Nodes at Elevation 100 ft for Reactor Building Sensitivity Studies



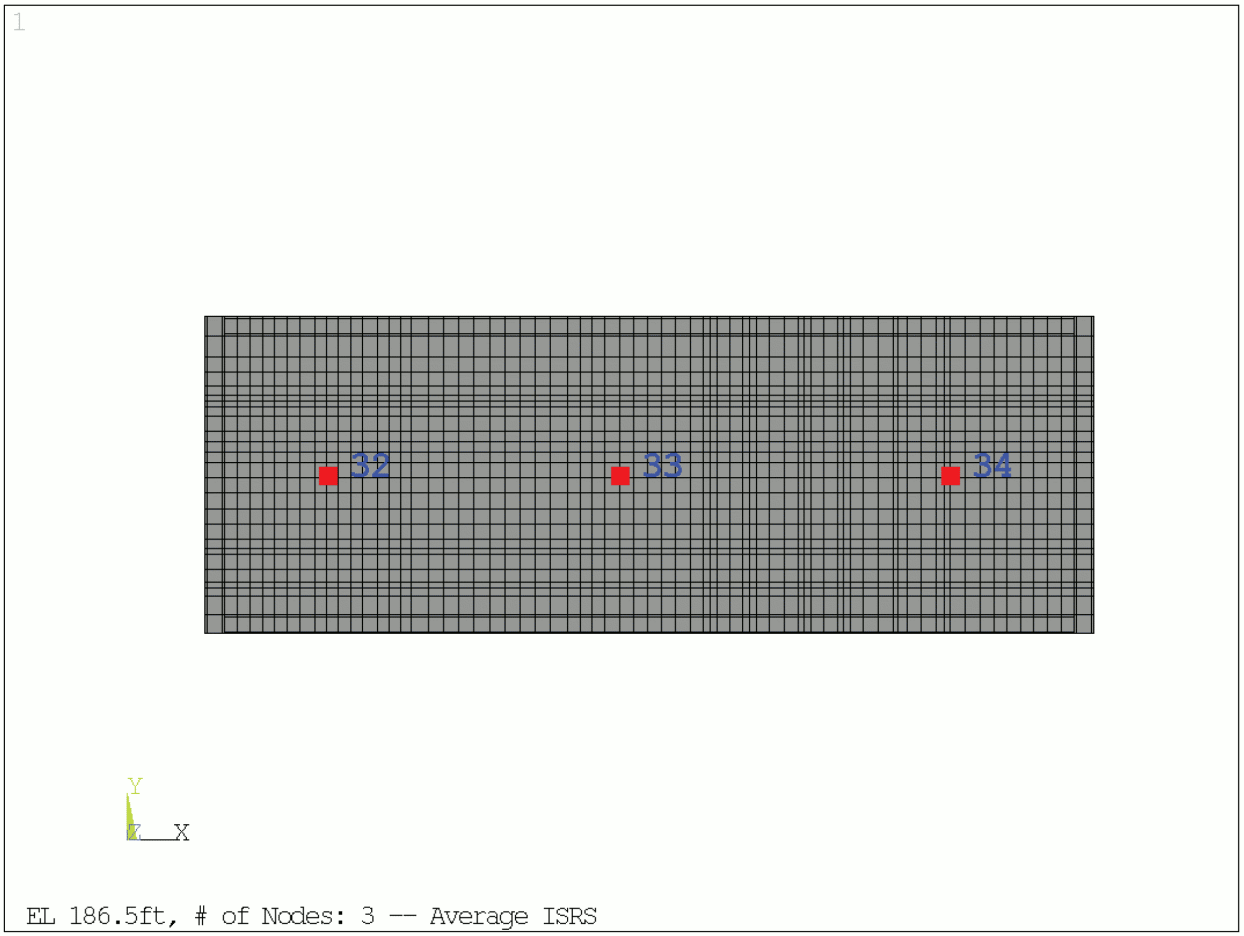
Note: The ISRS calculated at these nodes are enveloped.

Figure 3.7.2-26: Selected Nodes at Elevation 146 ft for Reactor Building Sensitivity Studies



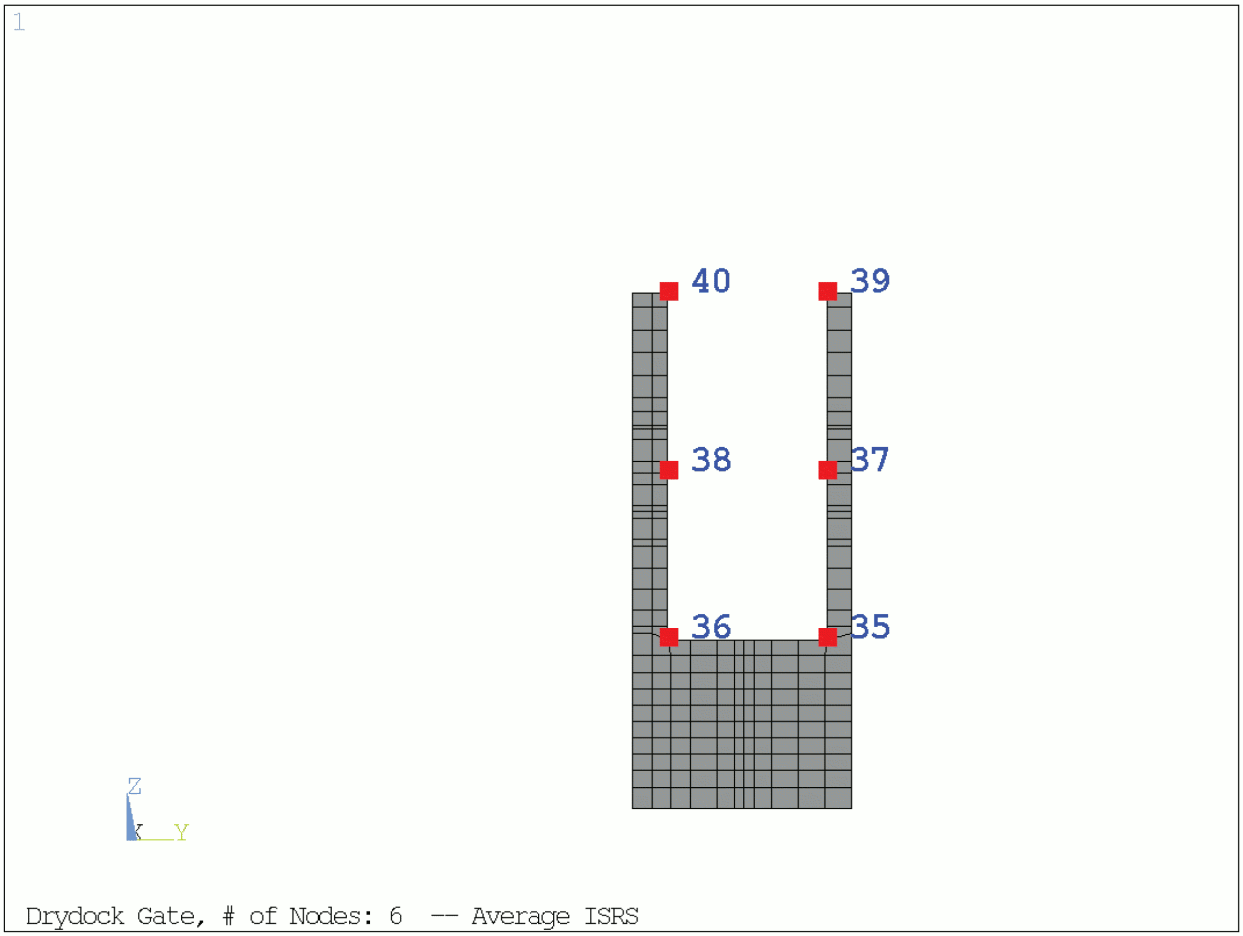
Note: The ISRS calculated at these nodes are enveloped.

Figure 3.7.2-27: Selected Nodes at Elevation 187 ft for Reactor Building Sensitivity Studies



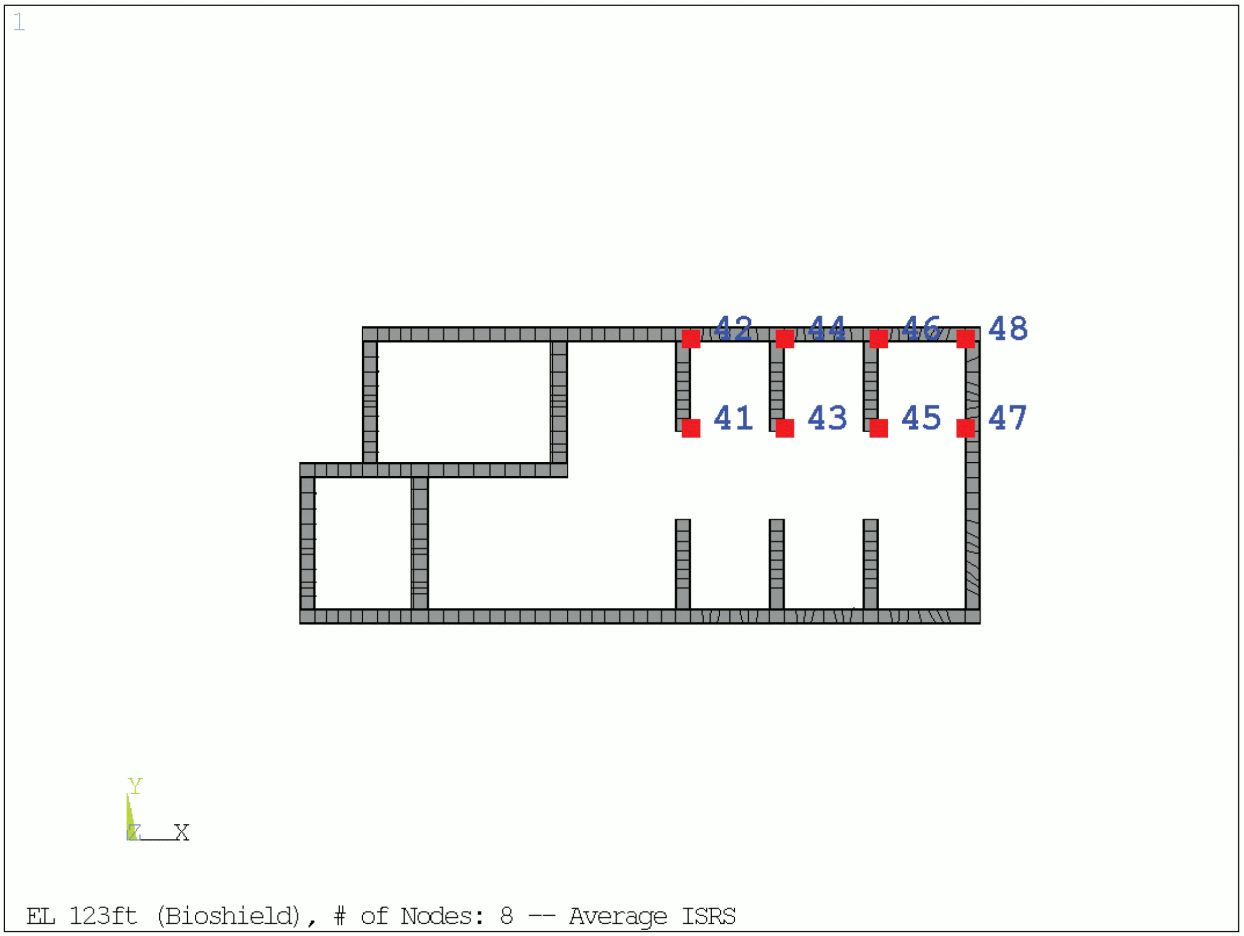
Note: The ISRS calculated at these nodes are averaged.

Figure 3.7.2-28: Selected Nodes Around the Dry Dock Gate for Reactor Building Sensitivity Studies



Note: The ISRS calculated at these nodes are averaged.

Figure 3.7.2-29: Selected Nodes at Elevation 123 ft to Represent the Bioshield for Reactor Building Sensitivity Studies



Note: The ISRS calculated at these nodes are averaged.

Figure 3.7.2-30: Node 33 from Figure 3.7.2-28 Acceleration X-Component Time Histories and Fourier Spectra Calculated with Capitola Input Motion for Baseline Soil-7 Case

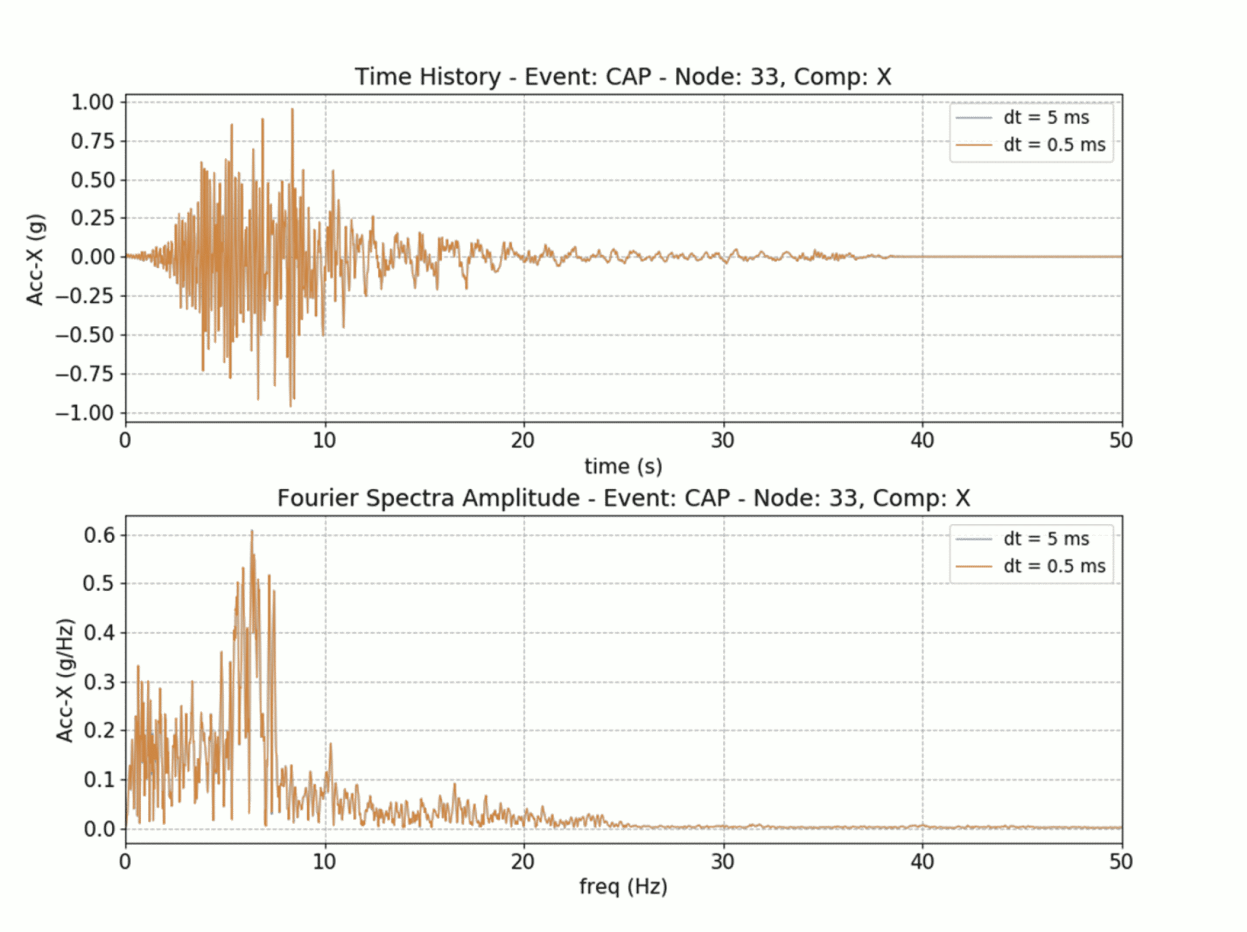


Figure 3.7.2-31: Node 33 from Figure 3.7.2-28 Acceleration Y-Component Time Histories and Fourier Spectra Calculated with Capitola Input Motion for Baseline Soil-7 Case

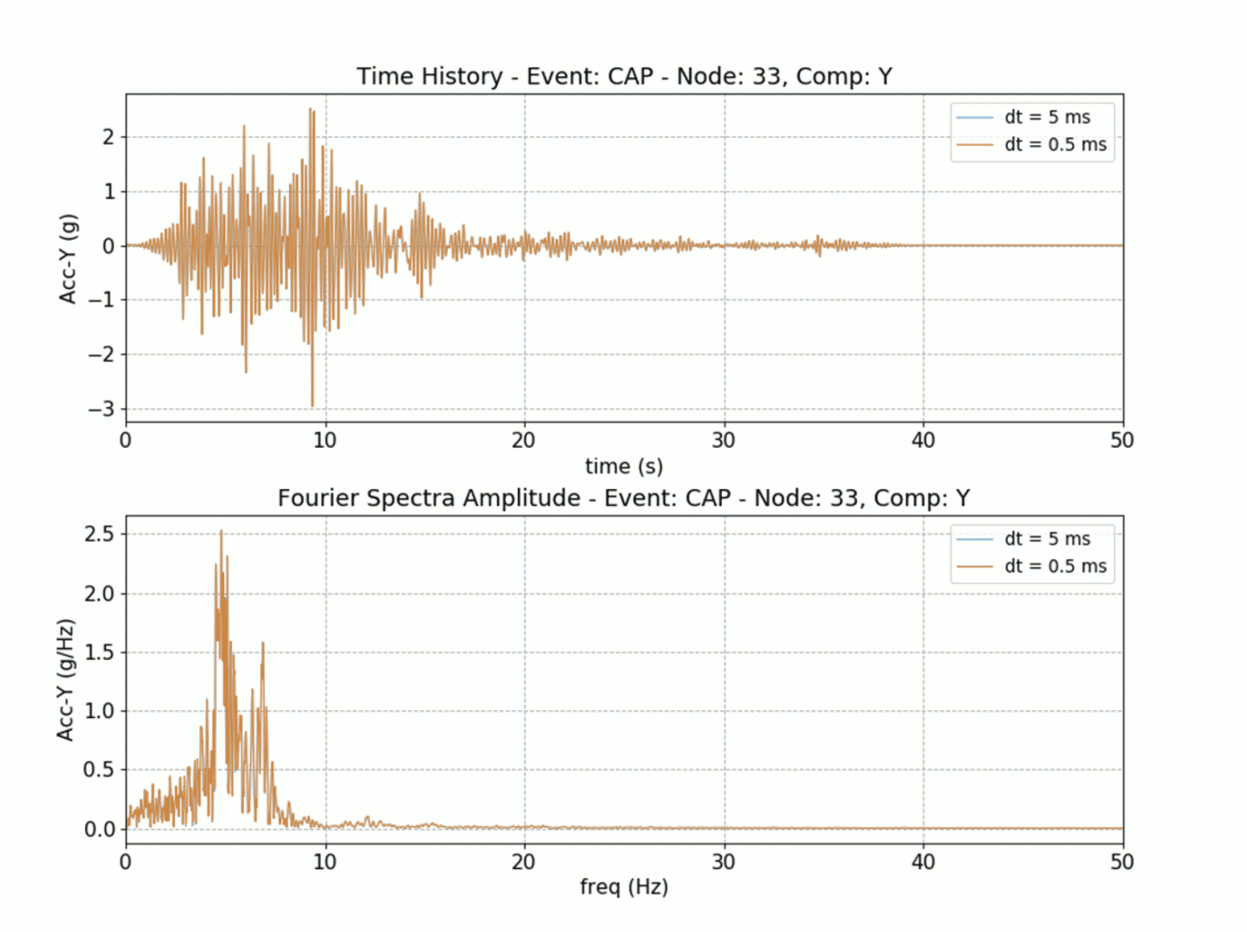


Figure 3.7.2-32: Node 33 from Figure 3.7.2-28 Acceleration Z-Component Time Histories and Fourier Spectra Calculated with Capitola Input Motion for Baseline Soil-7 Case

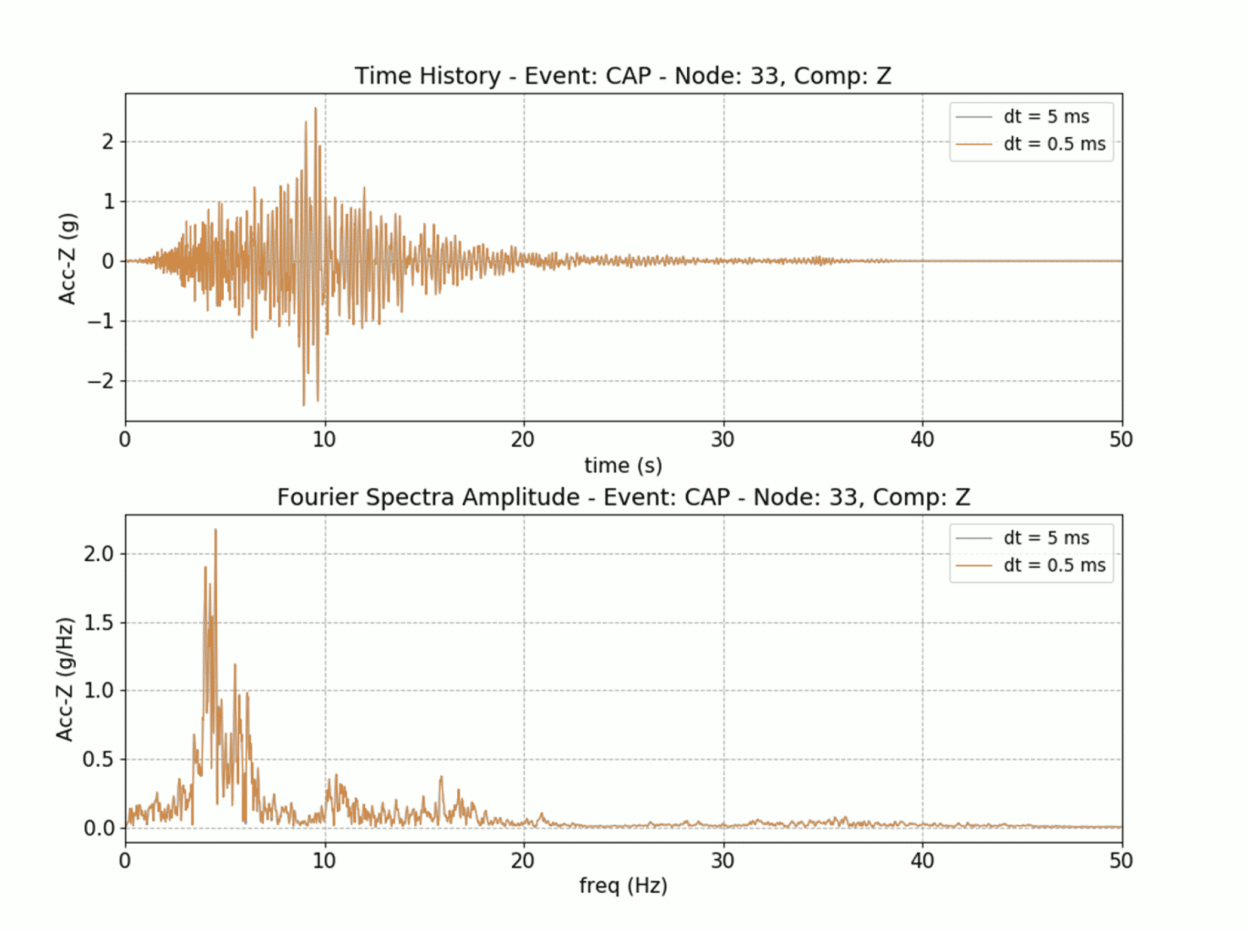
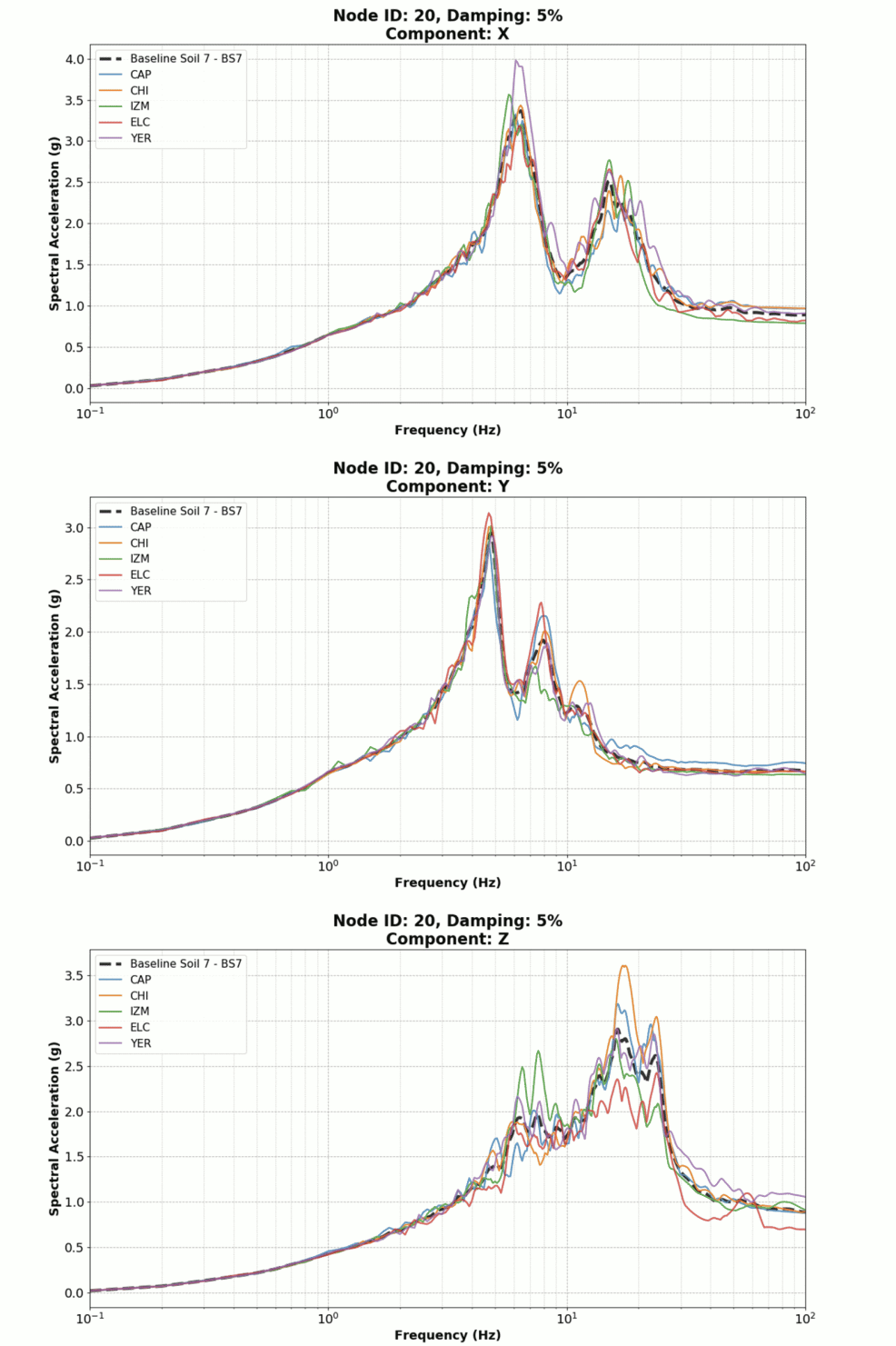
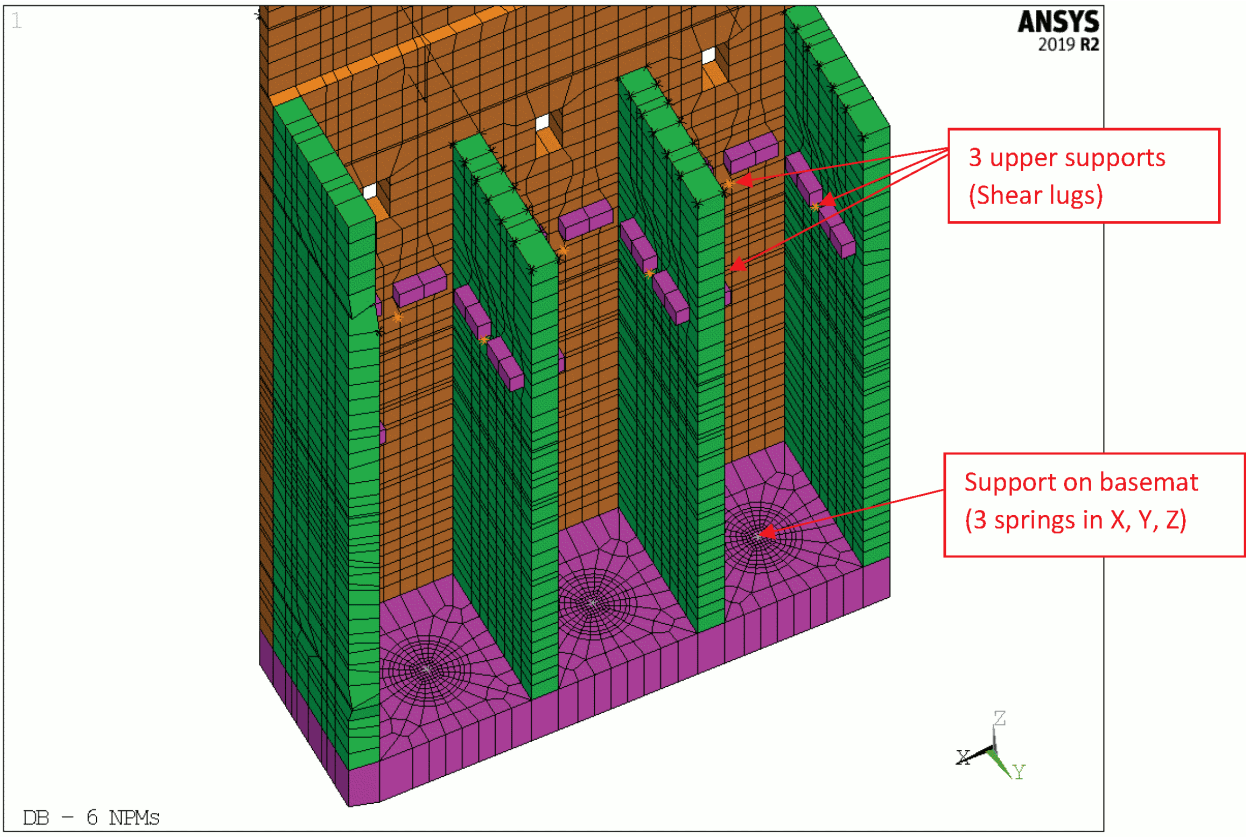


Figure 3.7.2-33: In-Structure Response Spectra Generated for Node 20 at Elevation 100 ft (Figure 3.7.2-26) for the Sensitivity Study Baseline with Soil-7 Case



Note: The dashed, black line is the average ISRS from the five ISRS (colored curves) generated for the events in CSDRS group.

Figure 3.7.2-34: Location of Lateral and Vertical Supports at each NuScale Power Module



Notes: 3 springs for the upper supports and 3 springs for the supports on the basemat

Figure 3.7.2-35: Spring Elements Selected for Calculation of NuScale Power Module Reactions

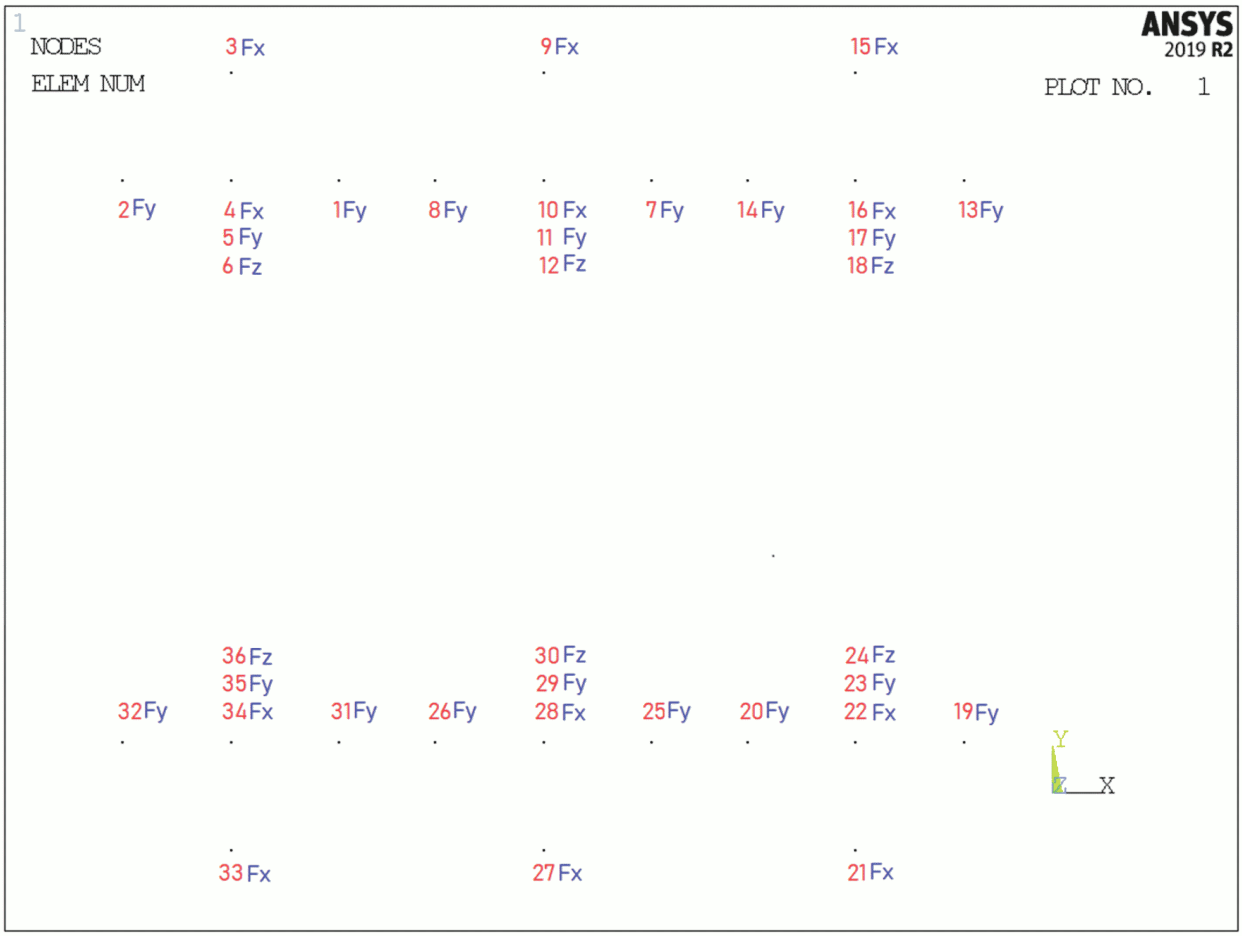
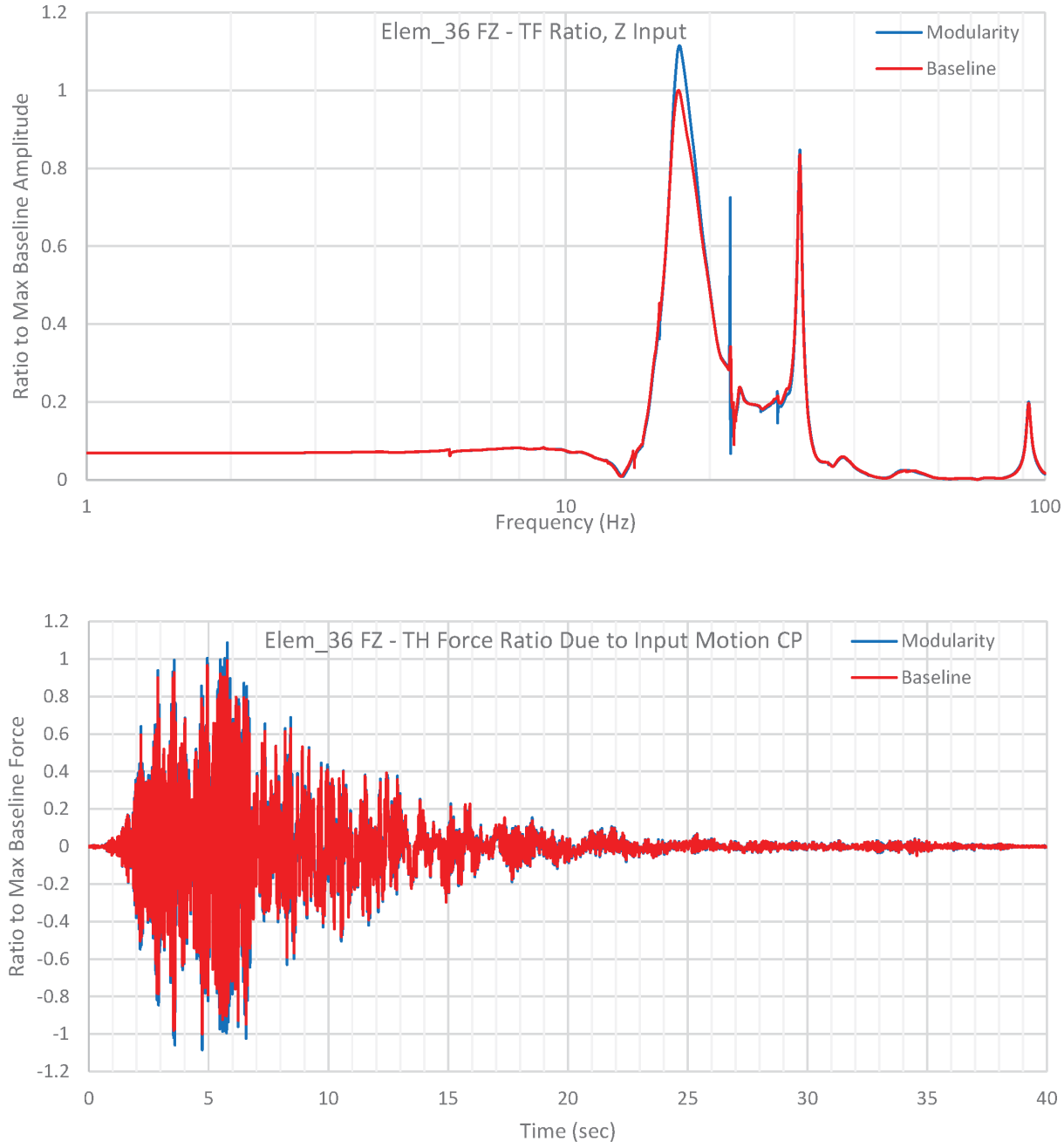


Figure 3.7.2-36: Transfer Functions and the Time-History Response Comparison for the Spring Force of Element 36 for the Baseline and Modularity Cases with Soil 7 Soil Properties and Input Motion Based on Capitola Record



Note: Results are divided by the maximum baseline result to achieve the presented ratios.

Figure 3.7.2-37: Location and Identification of Section Cuts at Basemat Elevation for Reactor Building Sensitivity Studies

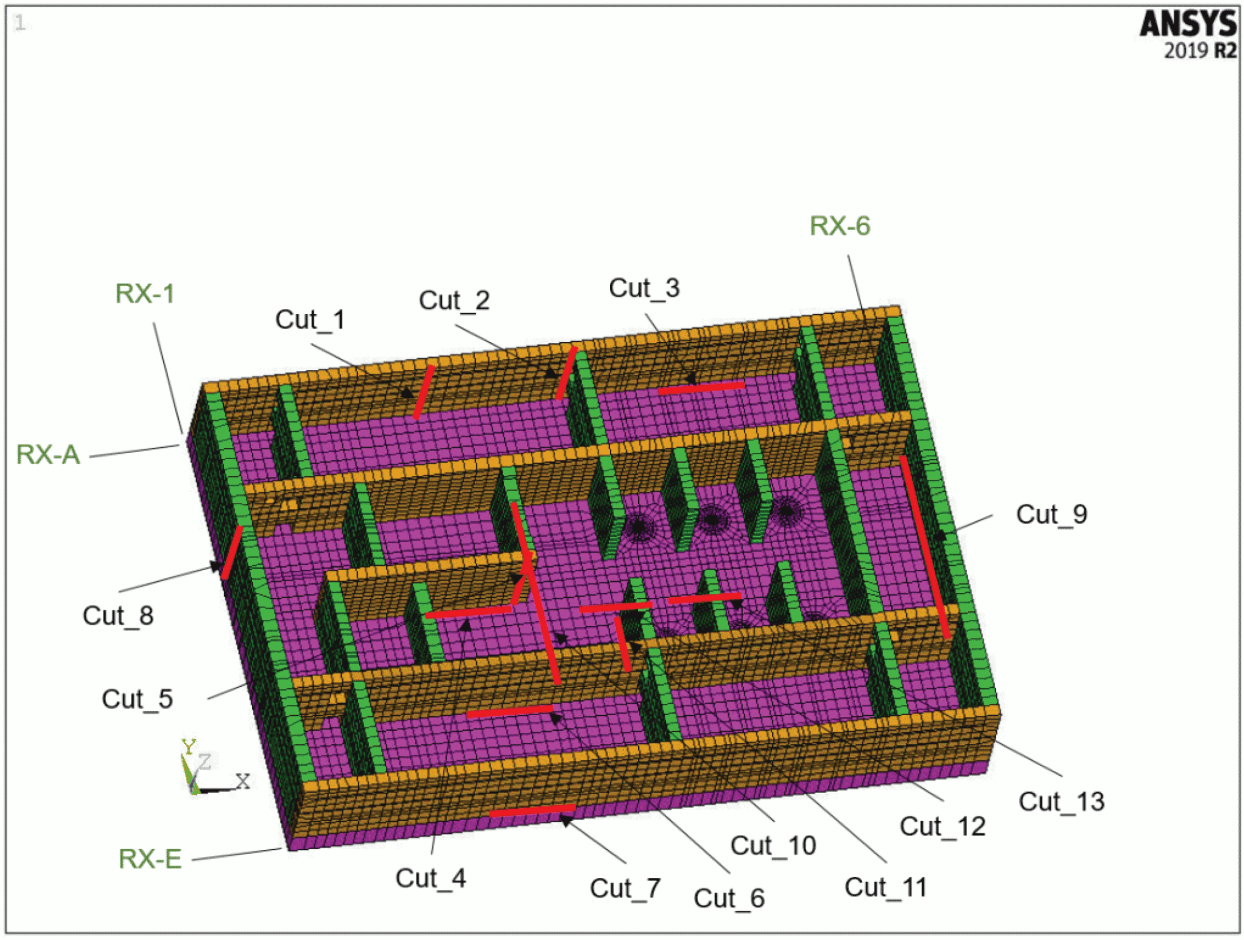


Figure 3.7.2-38: Location and Identification of Section Cuts at Floor at Elevation 55 ft for Reactor Building Sensitivity Studies

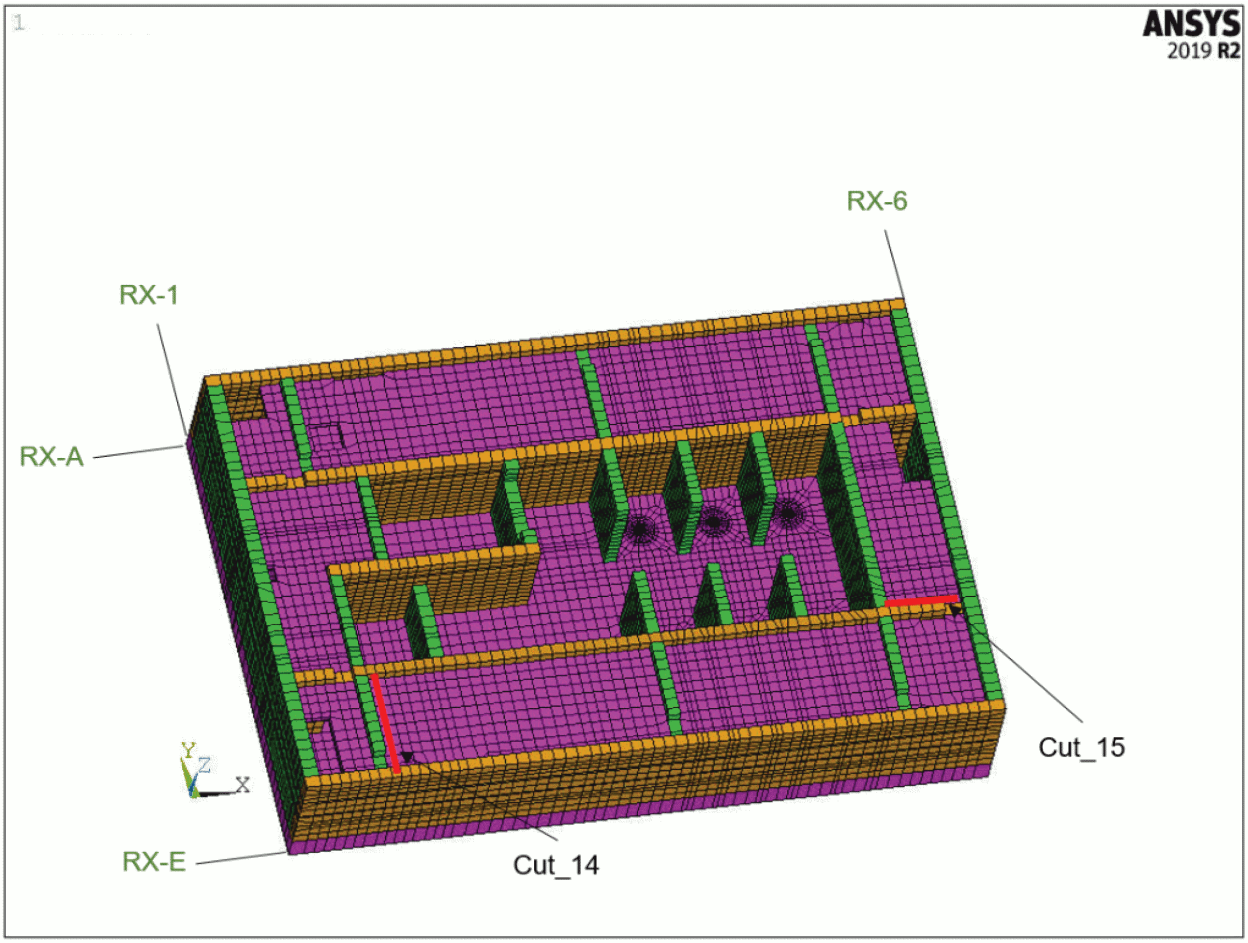


Figure 3.7.2-39: Location and Identification of Section Cuts at Floor at Elevation 70 ft for Reactor Building Sensitivity Studies

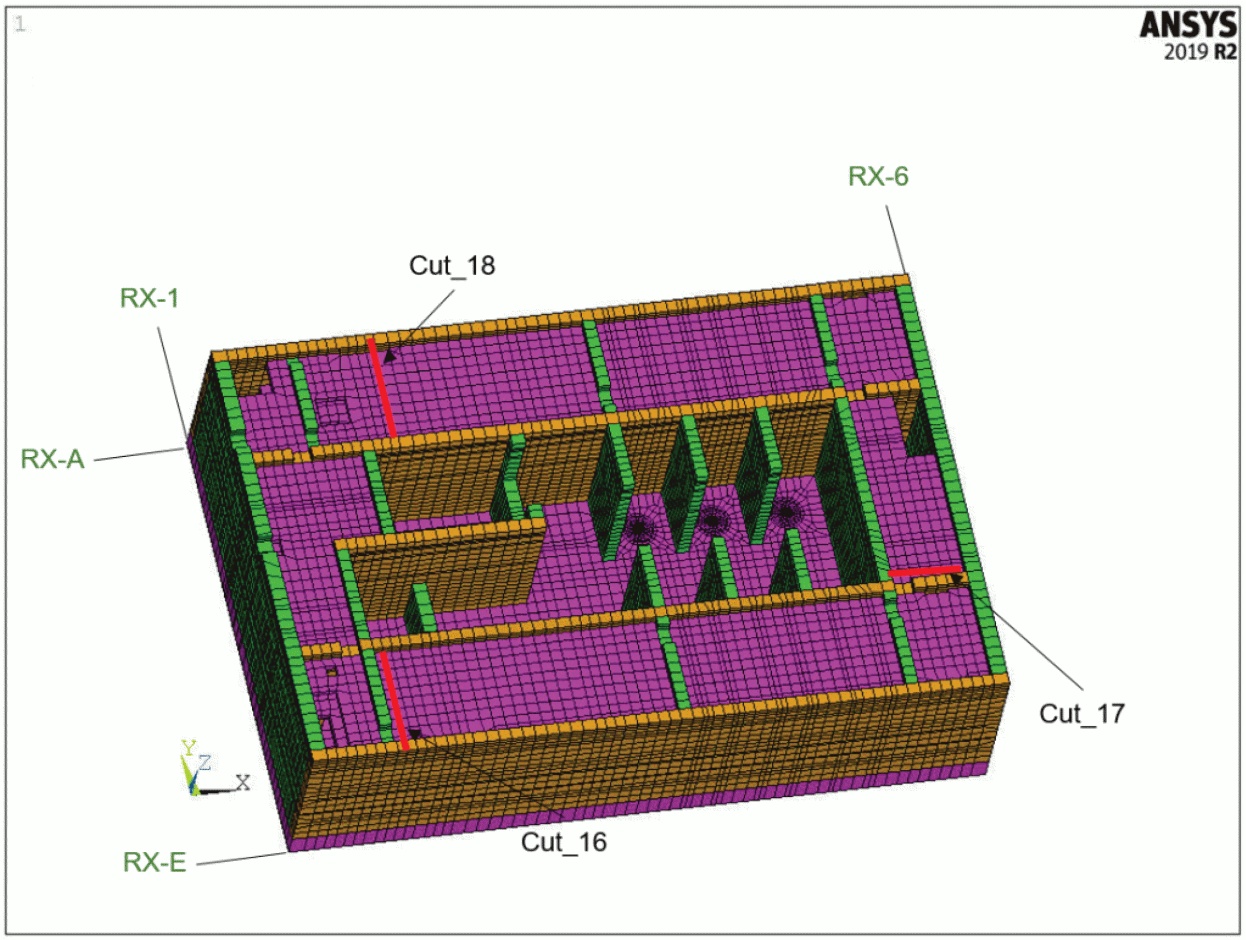


Figure 3.7.2-40: Location and Identification of Section Cuts at Floor at Elevation 85 ft for Reactor Building Sensitivity Studies

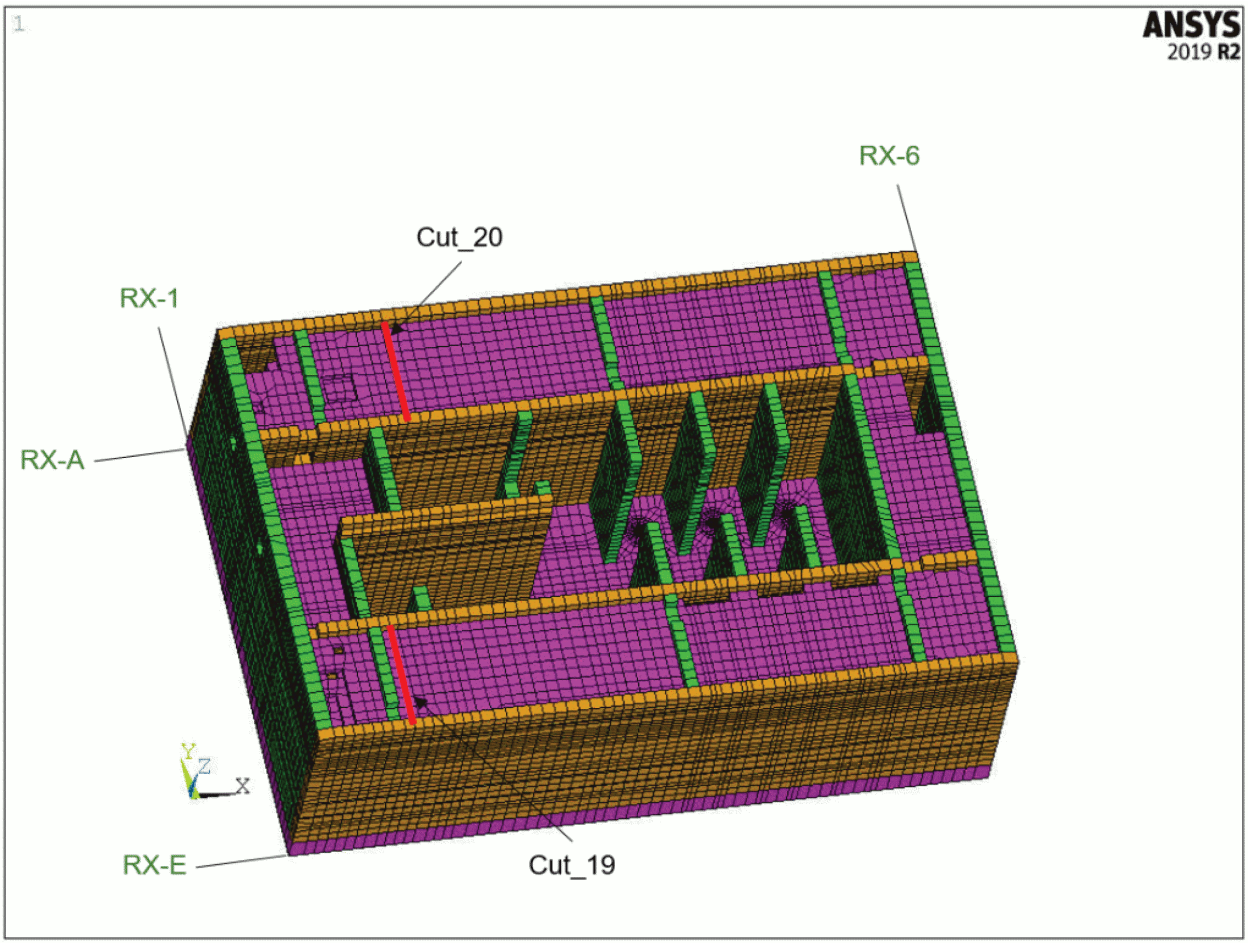


Figure 3.7.2-41: Location and Identification of Section Cuts at Floor at Elevation 100 ft for Reactor Building Sensitivity Studies

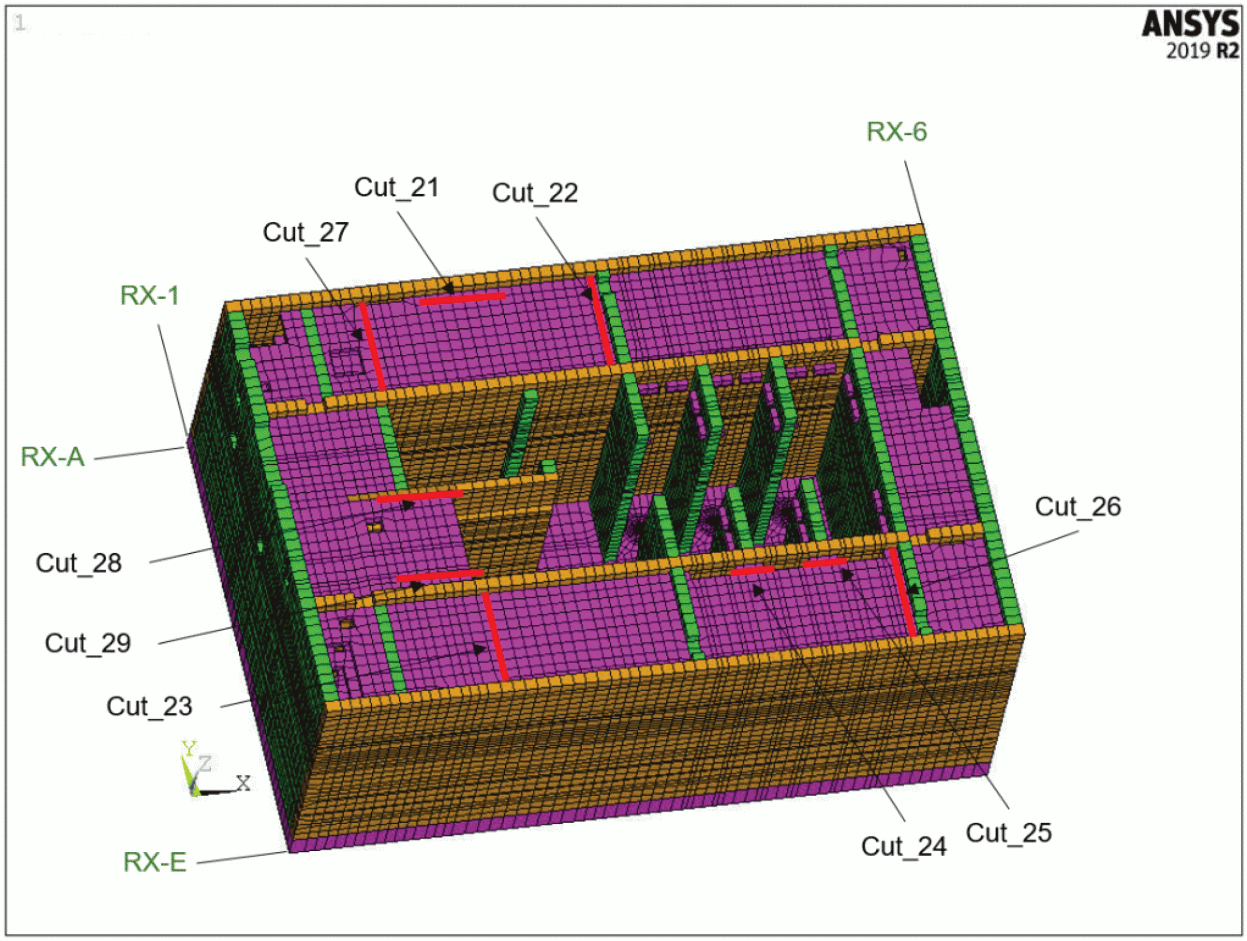


Figure 3.7.2-42: Location and Identification of Section Cuts at Roof at Elevation 146.5 ft for Reactor Building Sensitivity Studies

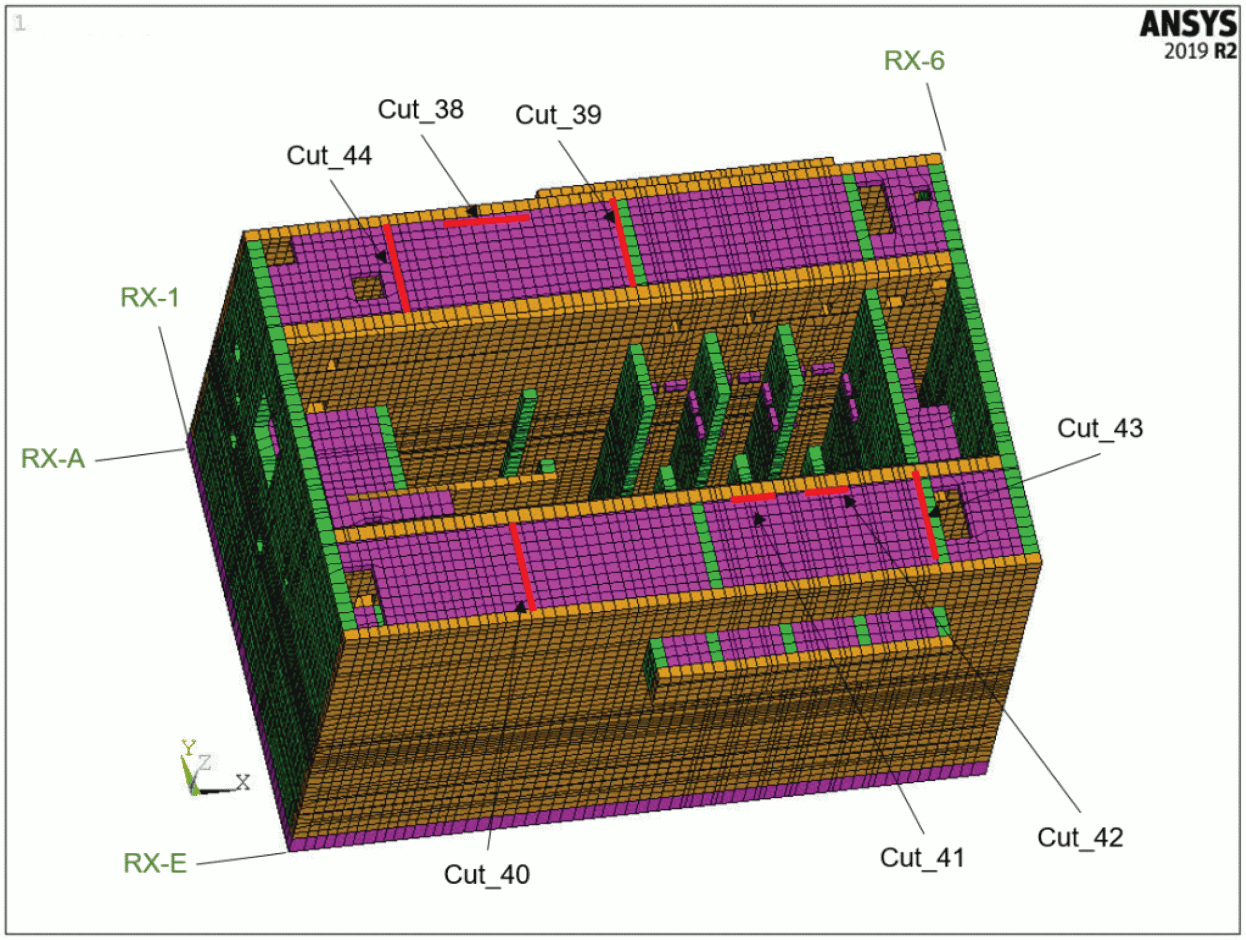


Figure 3.7.2-43: Location and Identification of Section Cuts at Roof Elevation for Reactor Building Sensitivity Studies

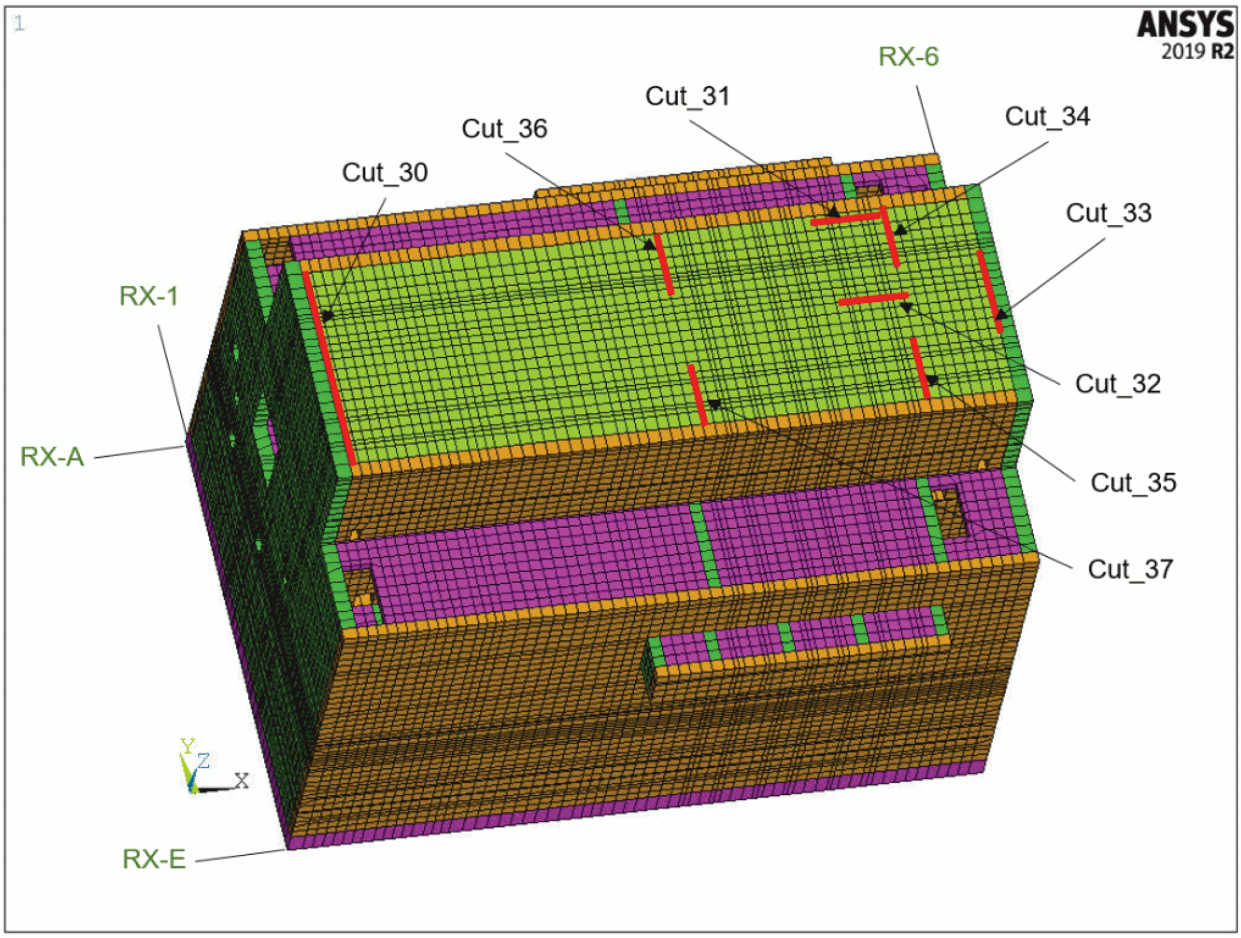
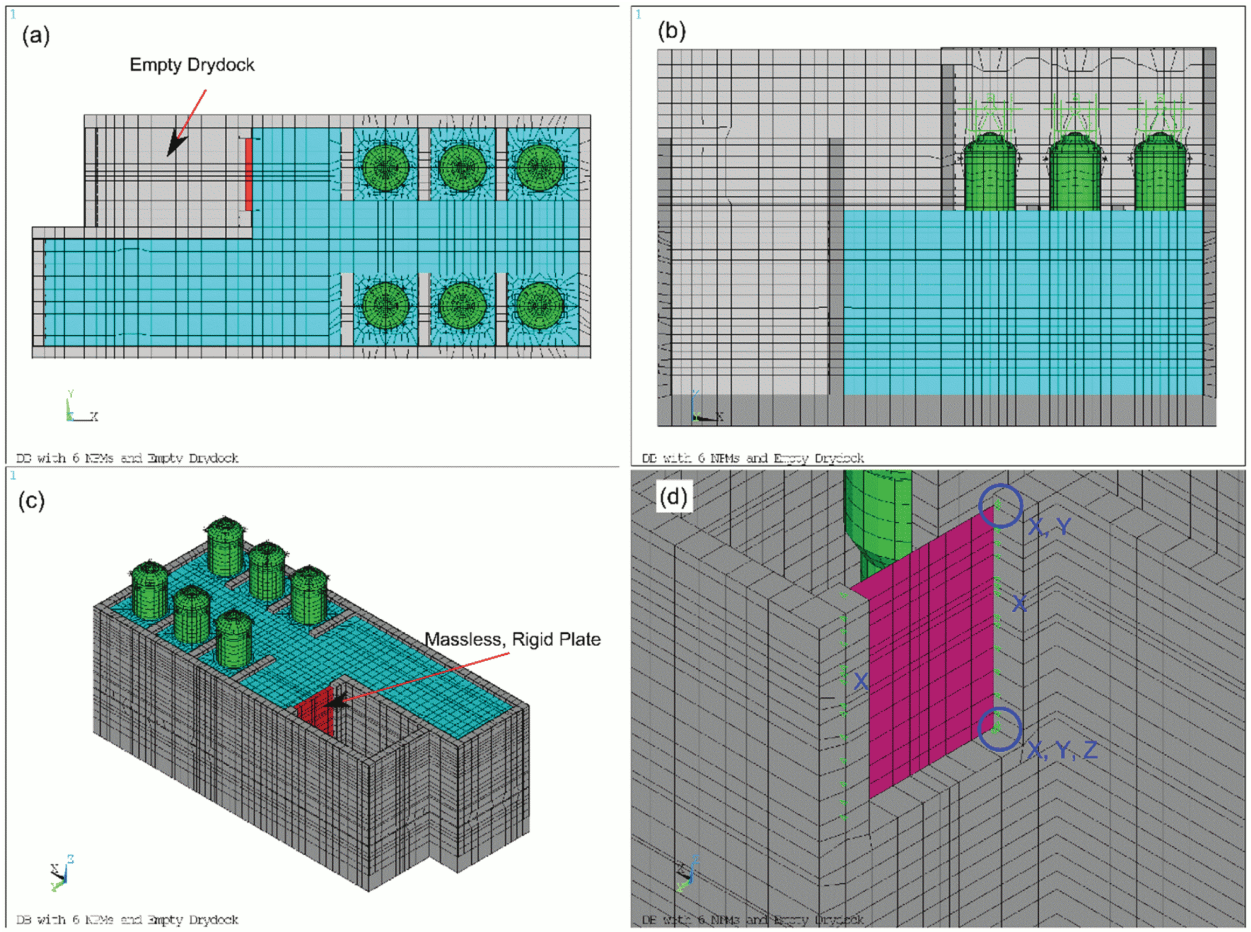


Figure 3.7.2-44: For the Reactor Building Sensitivity Study, (a) Top View of the Pool with Empty Dry Dock, (b) Elevation View of the Pool with Empty Dry Dock, (c) Massless, Rigid Plate Separating Reactor Pool from the Empty Dry Dock and the Location of the MASS21 Element, Representing the Gate Mass (d) Transition Degree-of-Freedoms Coupled in the Indicated Directions to the Neighboring Walls



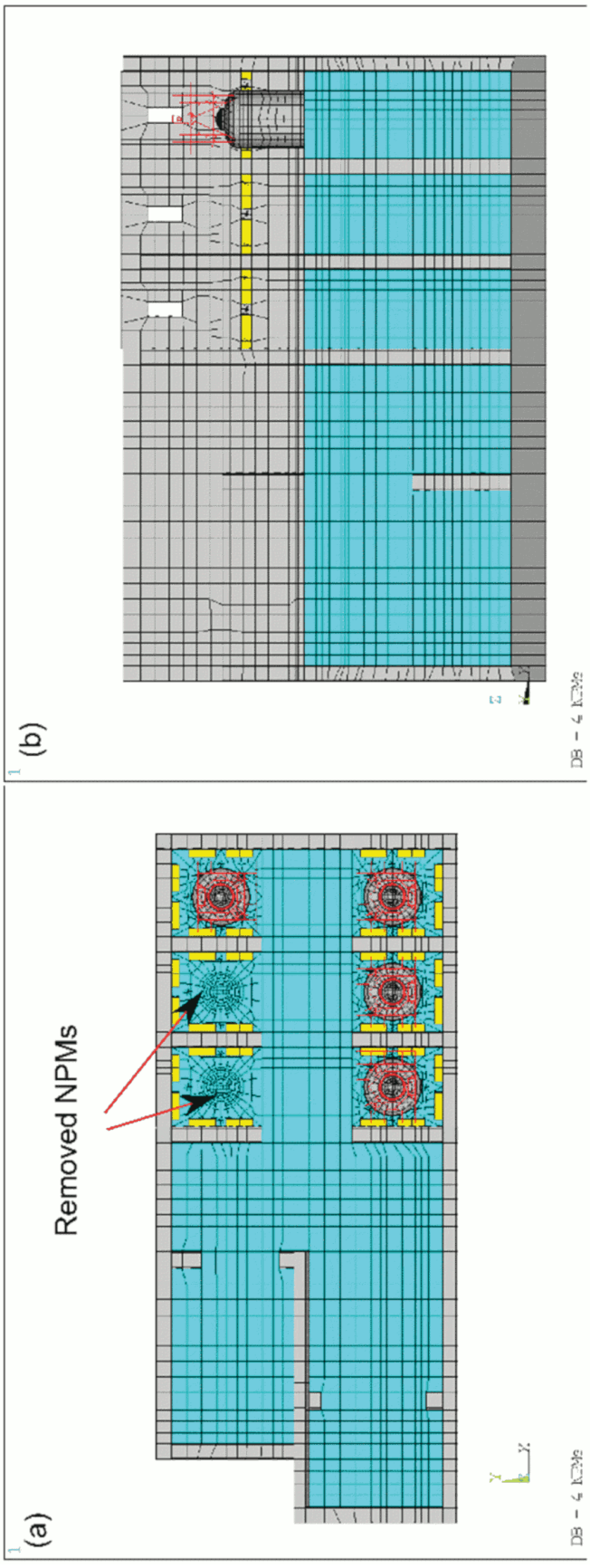


Figure 3.7.2-45: Top (a) and Elevation (b) Views of the Pool, NuScale Power Modules and Neighboring Walls for the Model with Reduced Number of NuScale Power Modules for the Reactor Building Sensitivity Study

Figure 3.7.2-46: Backfill of the Soil Separation Model, where the Top Four Soil Layers in the Top 25 ft with Reduced Stiffness are Colored

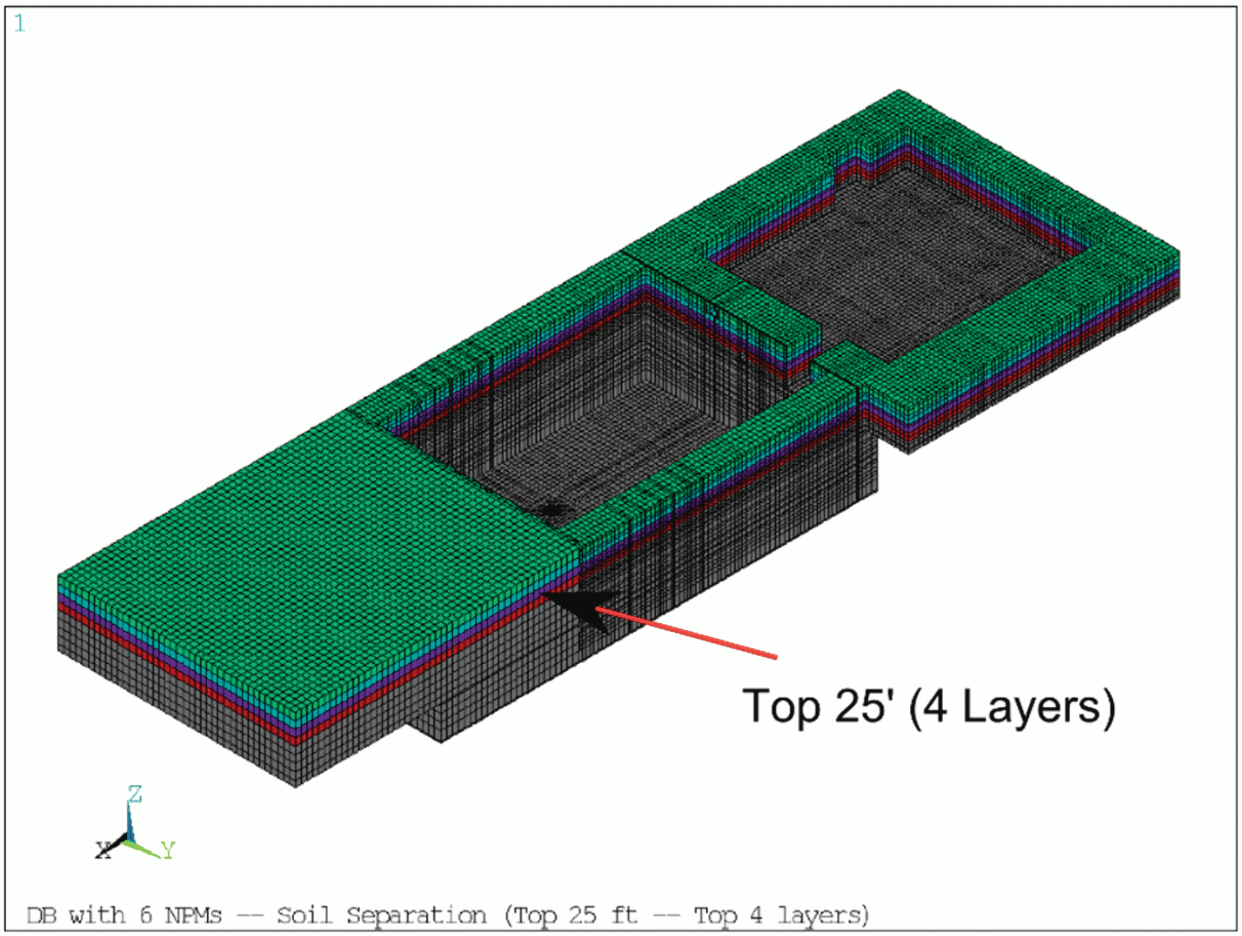
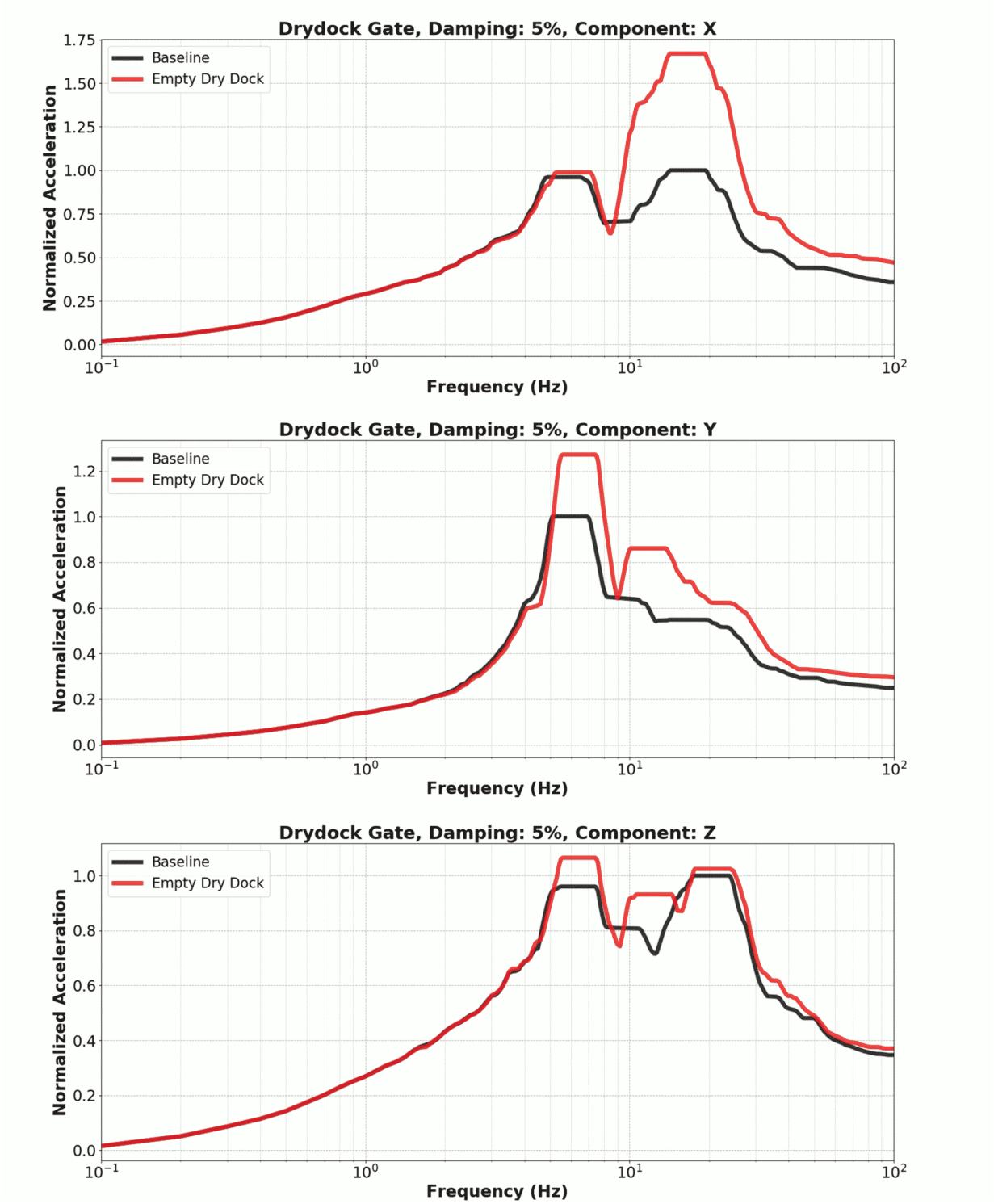
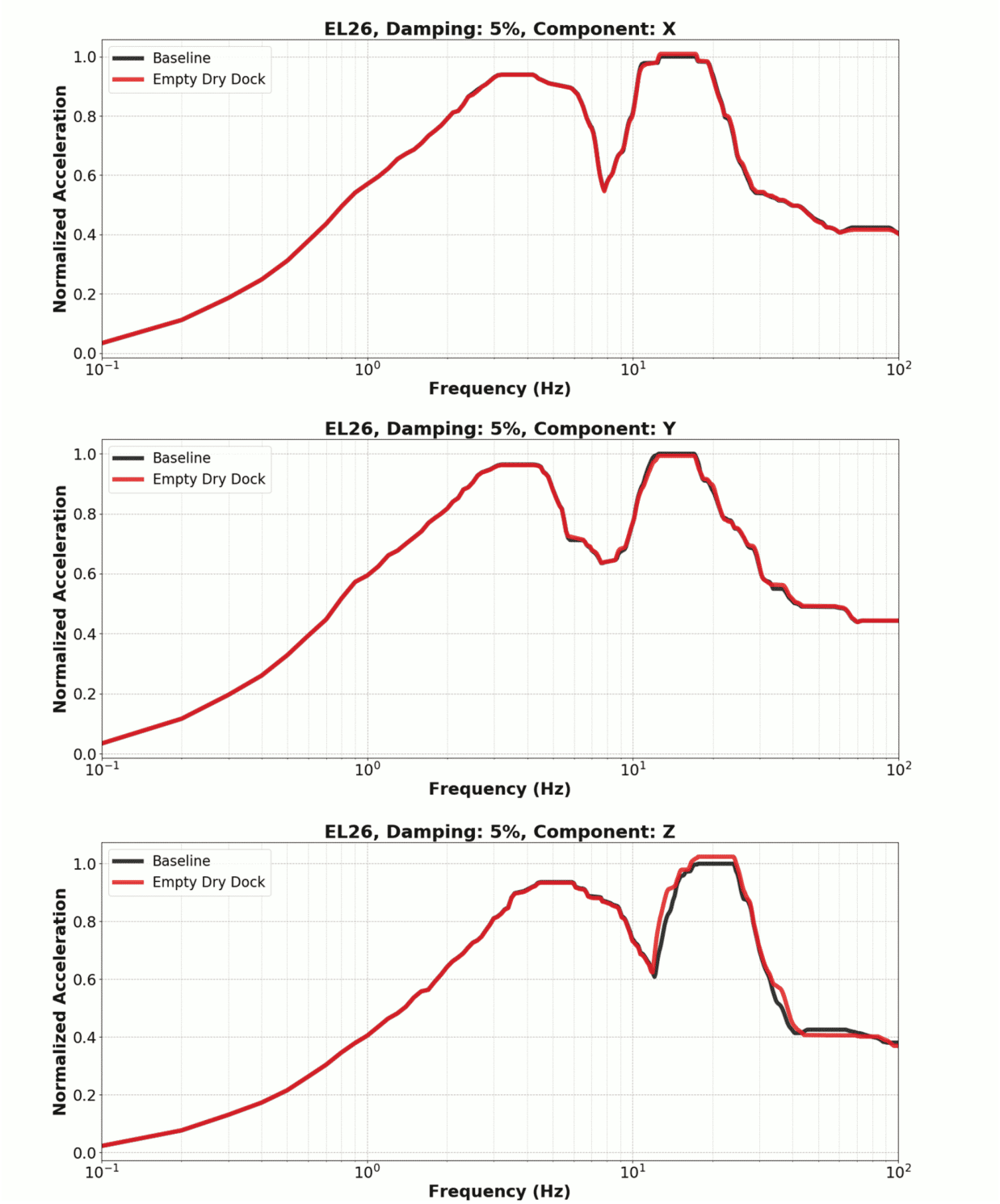


Figure 3.7.2-47: Empty Dry Dock vs Baseline - Dry Dock Gate Node Group



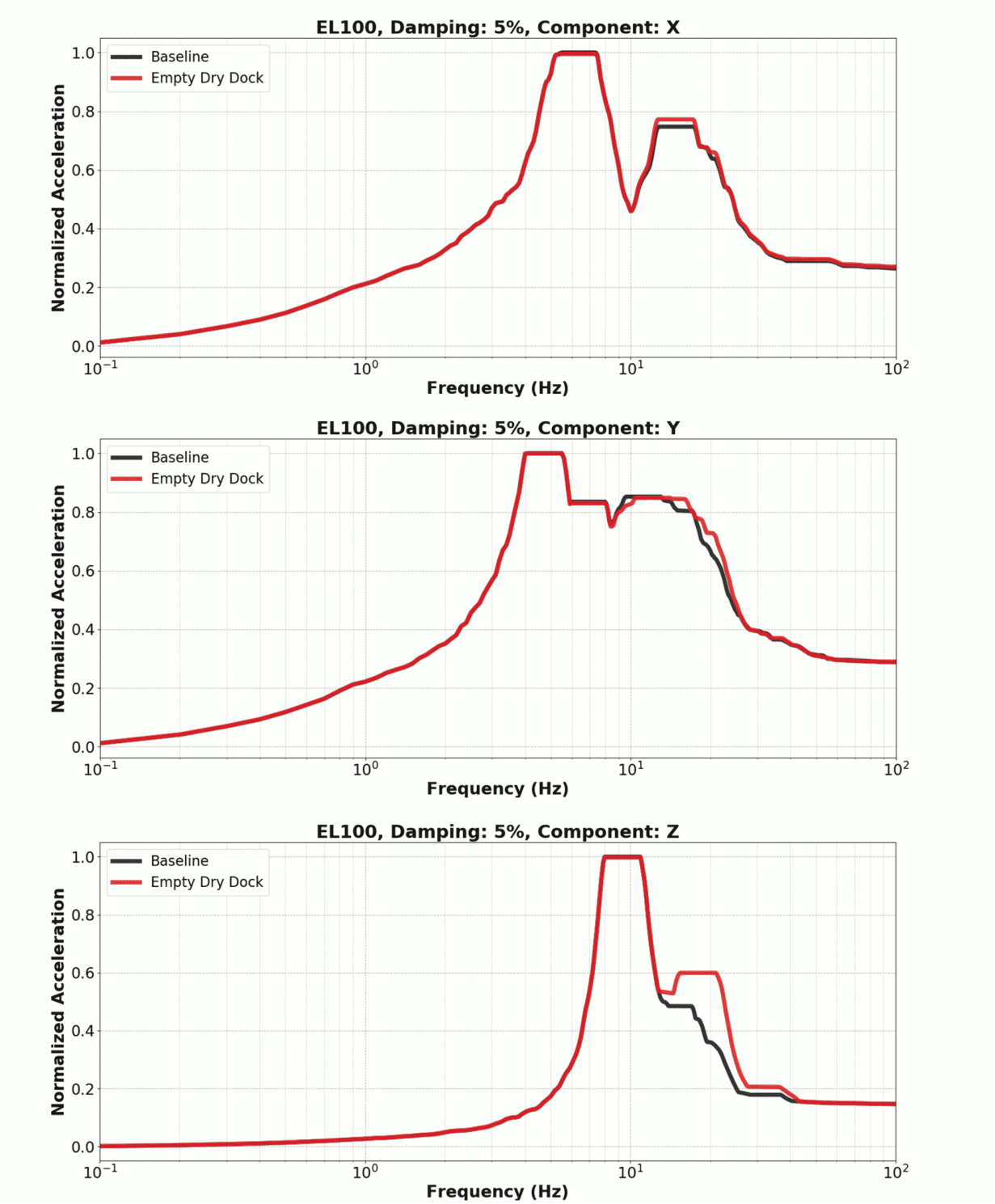
Note: Averaged, broadened and normalized absolute acceleration ISRS comparison for dry dock gate node group with 5% damping.

Figure 3.7.2-48: Empty Dry Dock vs Baseline- Node at El. 26 ft



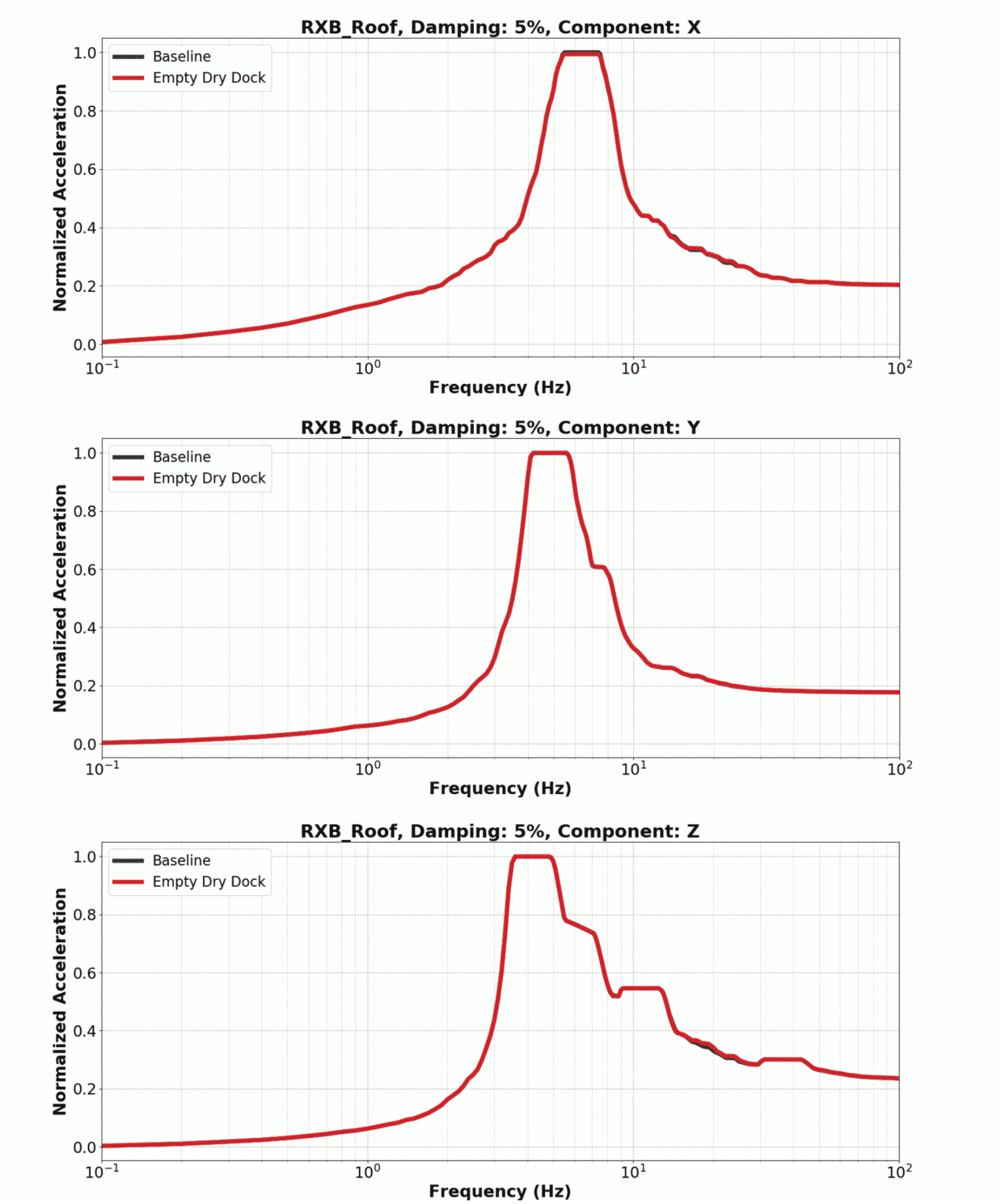
Note: Broadened and normalized absolute acceleration ISRS comparison for the node at the elevation of 26 ft (above the basemat) with 5% damping.

Figure 3.7.2-49: Empty Dry Dock vs Baseline - Nodes at El. 100 ft



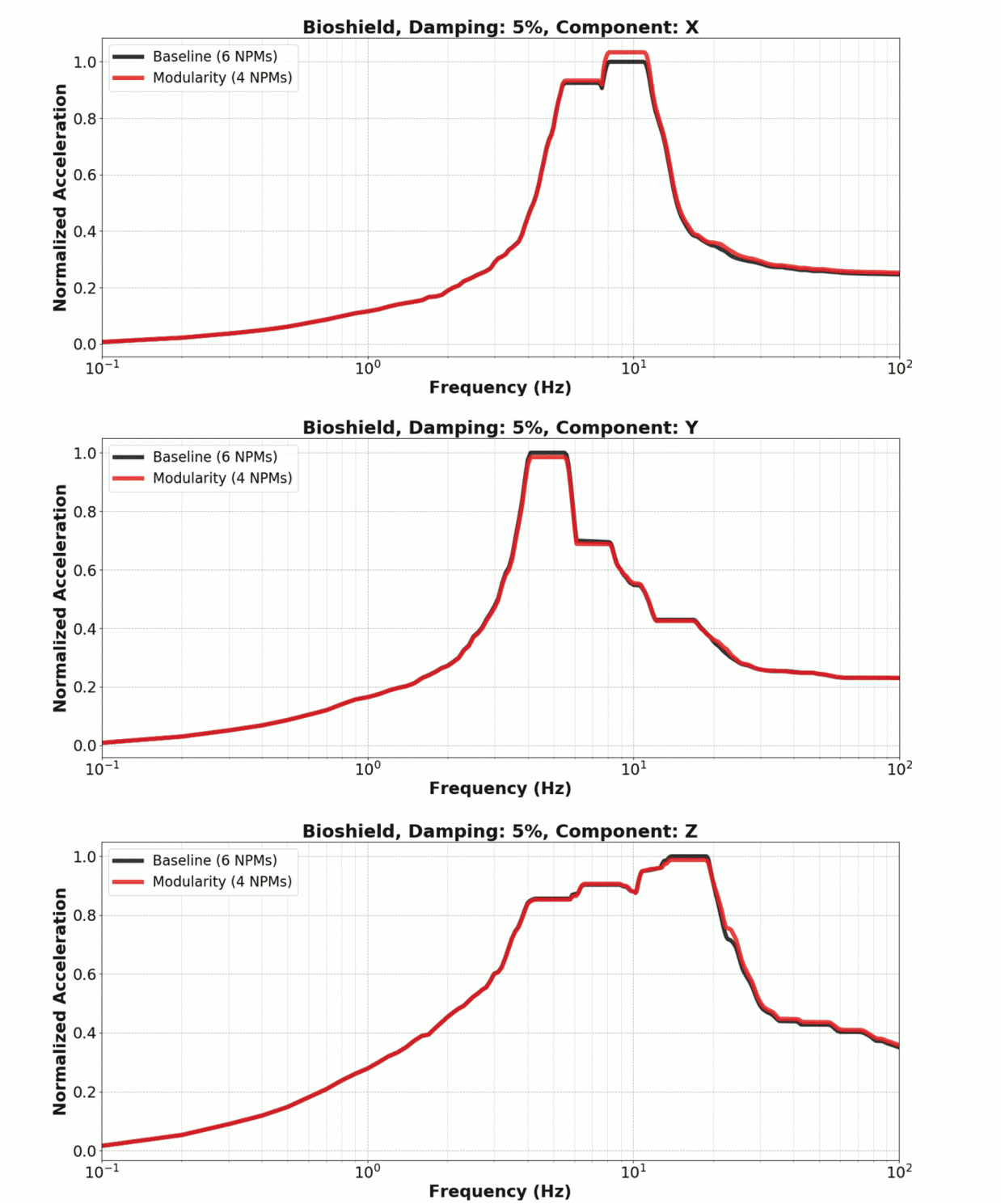
Note: Enveloped, broadened and normalized absolute acceleration ISRS comparison for the group of nodes at the elevation of 100 ft with 5% damping.

Figure 3.7.2-50: Empty Dry Dock vs Baseline - Nodes at Roof



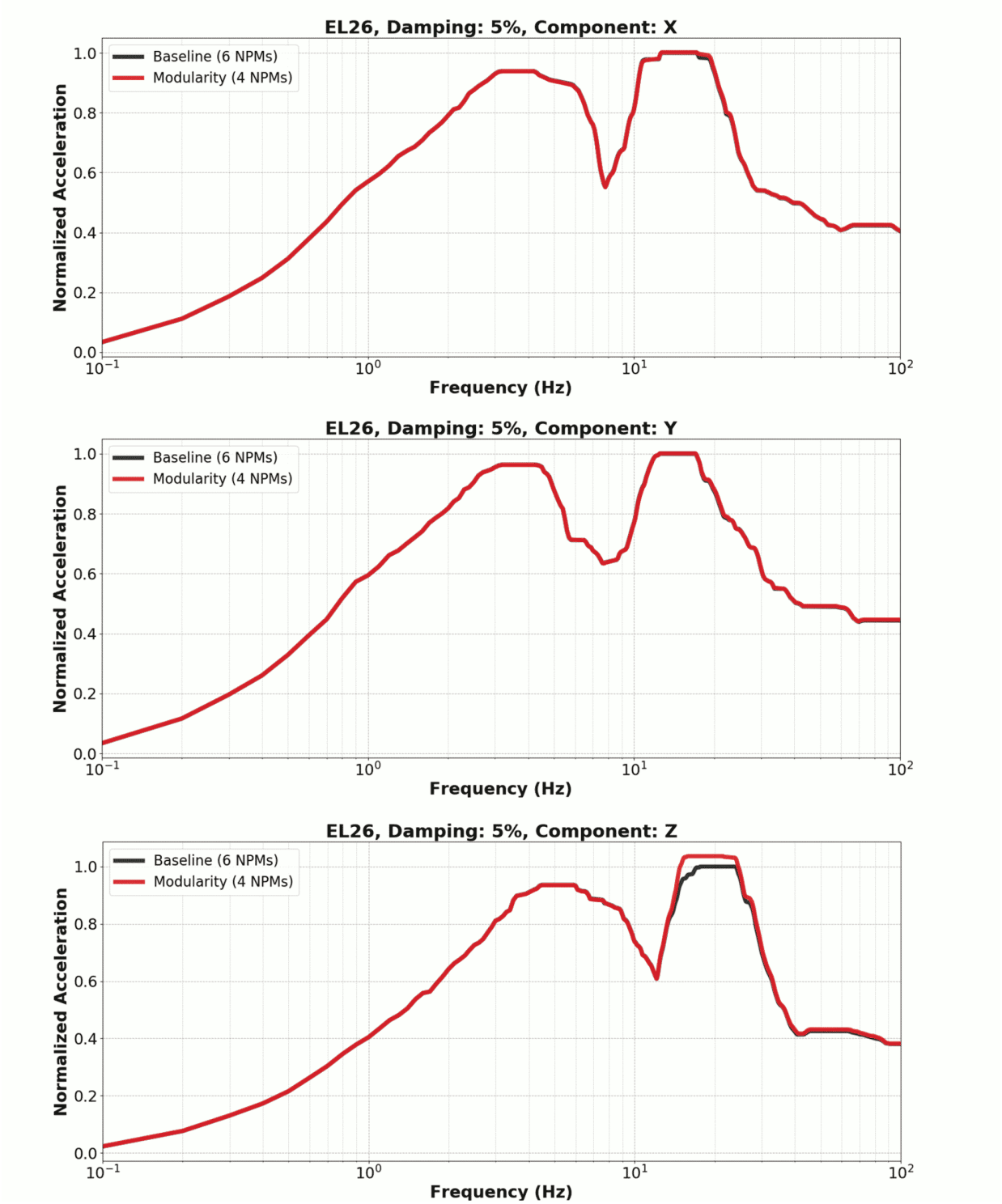
Note: Averaged, broadened and normalized absolute acceleration ISRS comparison for the group of nodes RXB roof with 5% damping.

Figure 3.7.2-51: Modularity vs Baseline - Bioshield node group



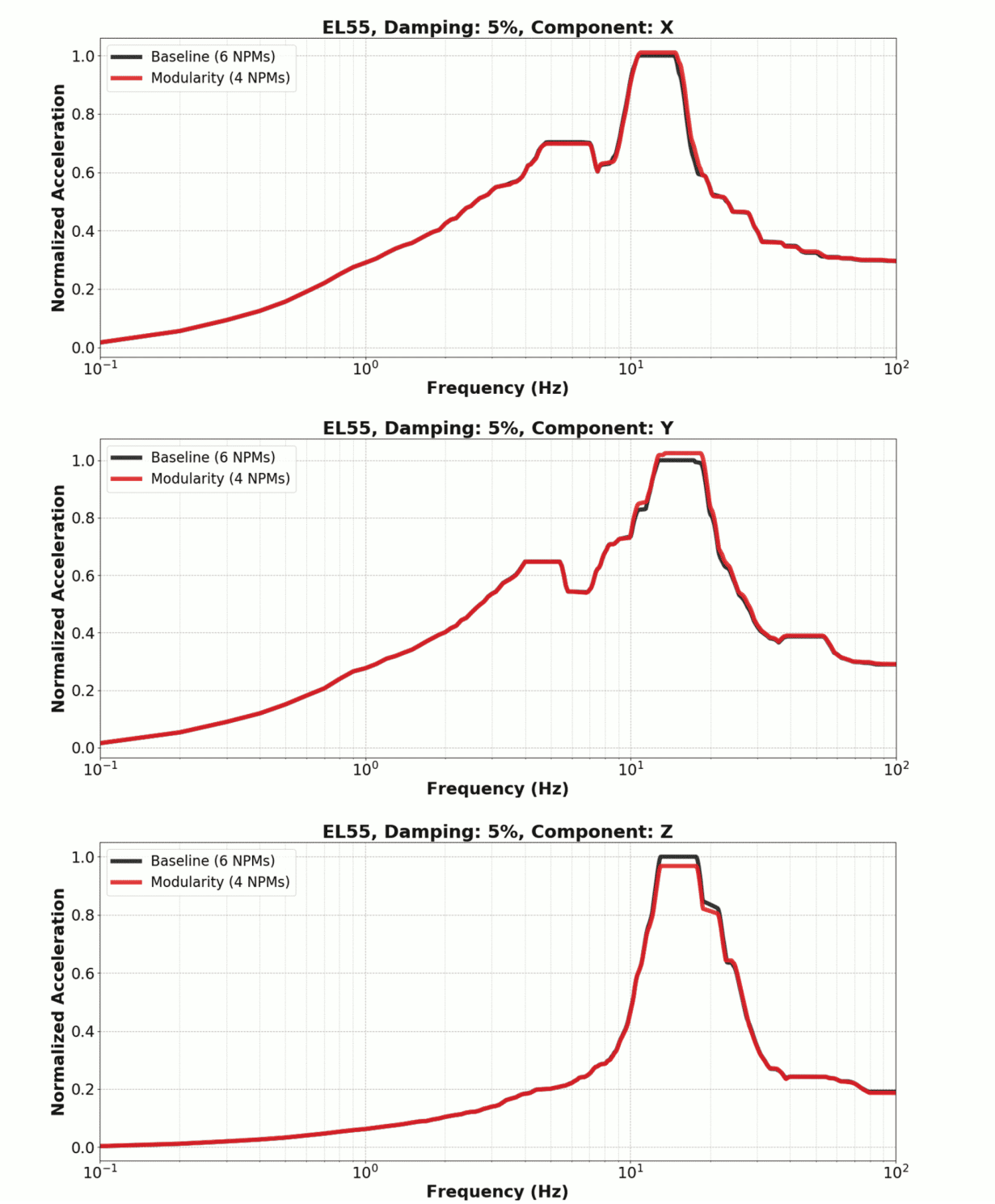
Note: Averaged, broadened and normalized absolute acceleration ISRS comparison for Bioshield node group with 5% damping.

Figure 3.7.2-52: Modularity vs Baseline - Node at El. 26 ft



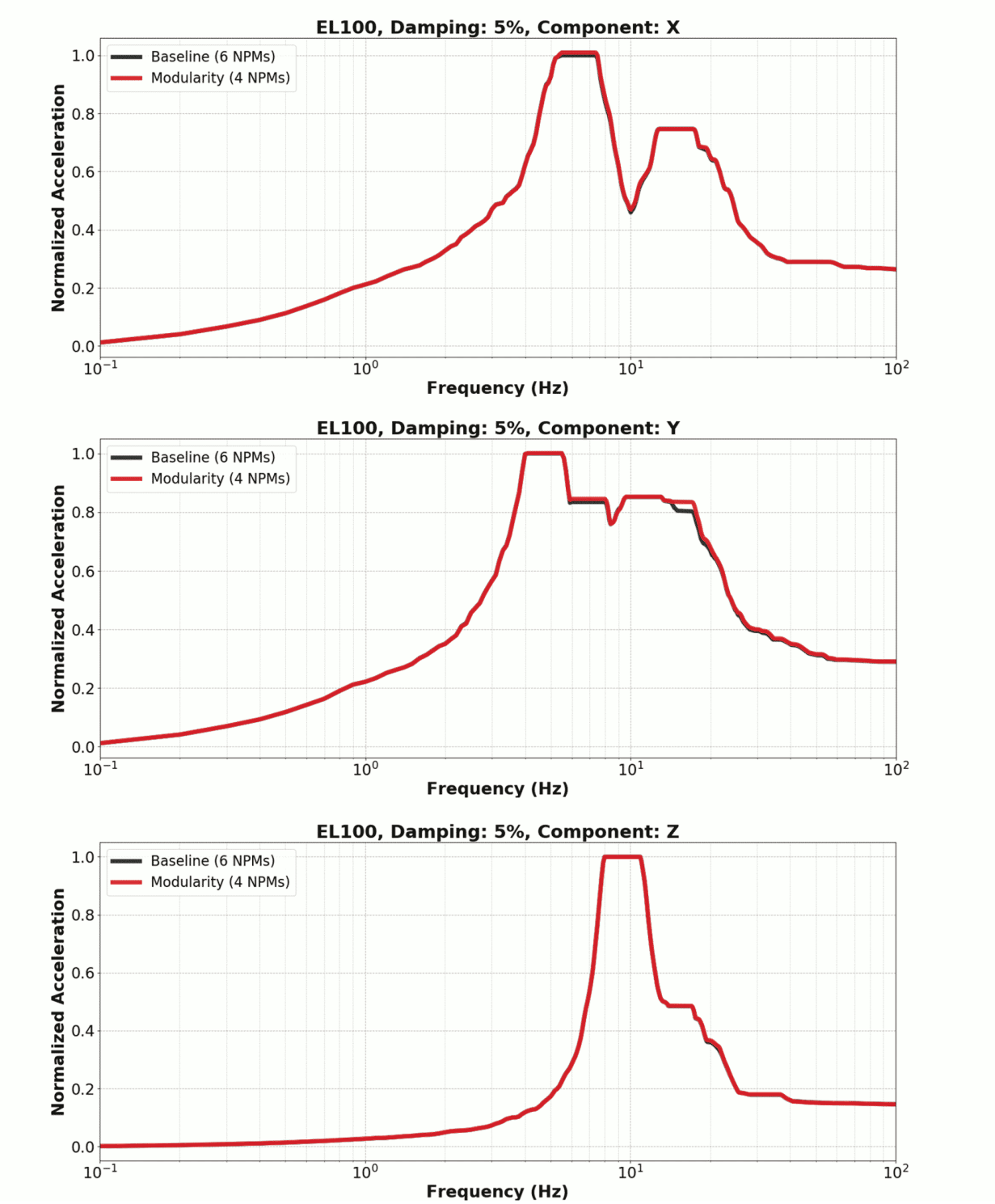
Note: Broadened and normalized absolute acceleration ISRS comparison for the node at the elevation of 26 ft (above the basemat) with 5% damping.

Figure 3.7.2-53: Modularity vs Baseline Nodes at El. 55 ft



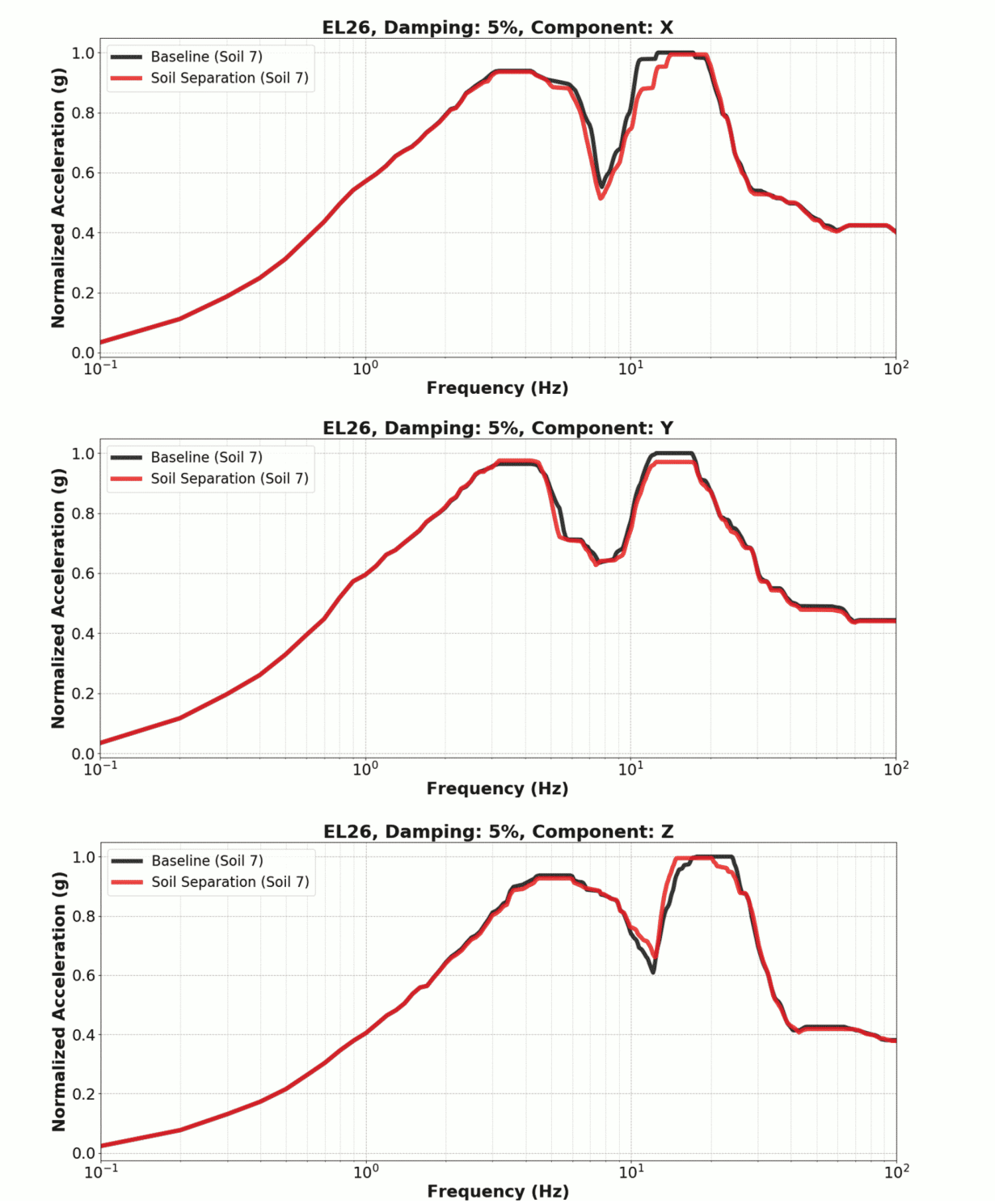
Note: Enveloped, broadened and normalized absolute acceleration ISRS comparison for the group of nodes at the elevation of 55 ft with 5% damping.

Figure 3.7.2-54: Modularity vs Baseline - Nodes at El. 100 ft



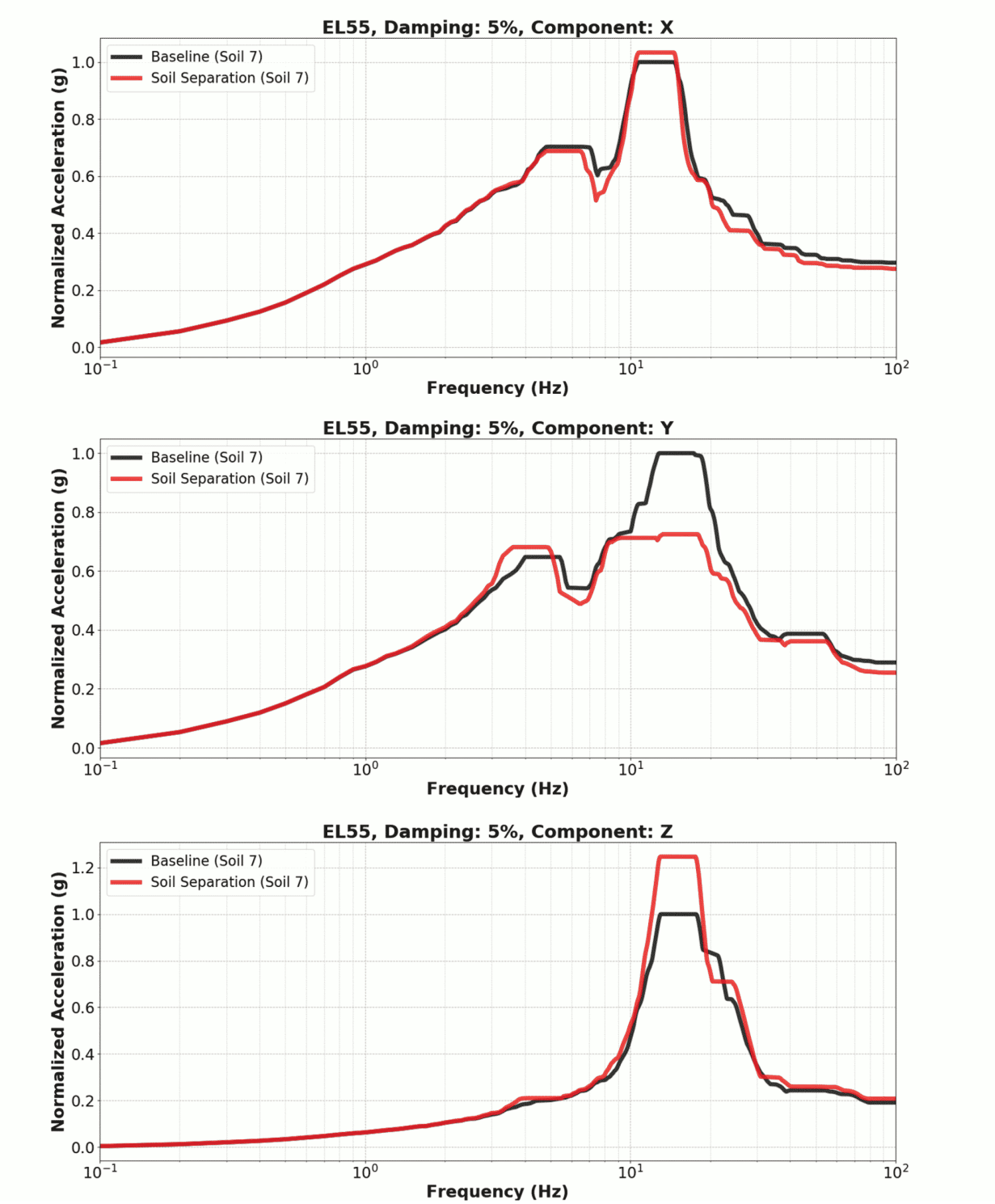
Notes: Enveloped, broadened and normalized absolute acceleration ISRS comparison for the group of nodes at the elevation of 100 ft with 5% damping.

Figure 3.7.2-55: Soil Separation vs Baseline - Node at El. 26 ft



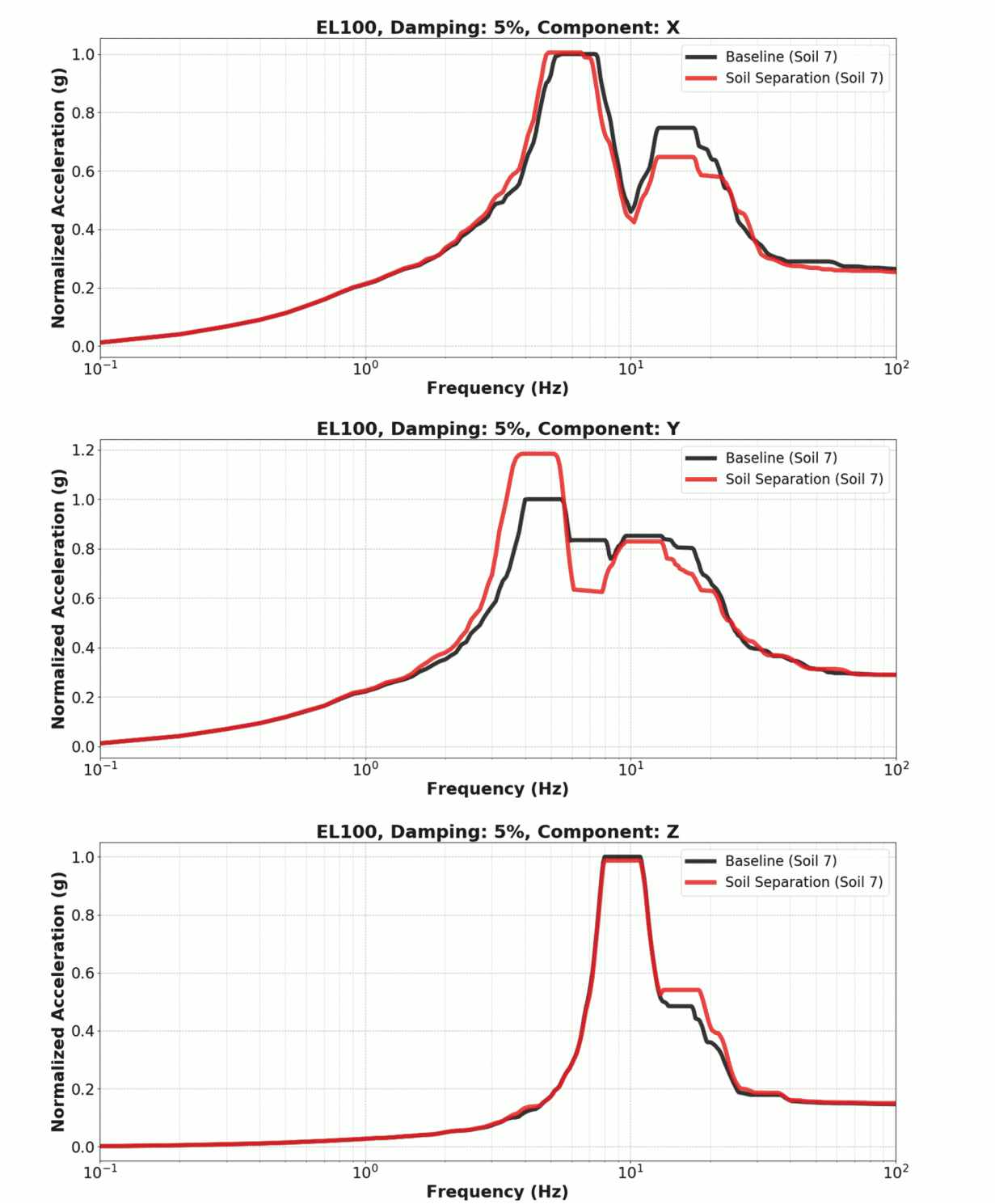
Notes: Broadened absolute acceleration ISRS comparison for the node at the elevation of 26 ft (above the basemat) with 5% damping.

Figure 3.7.2-56: Soil Separation vs Baseline - Nodes at El. 55 ft



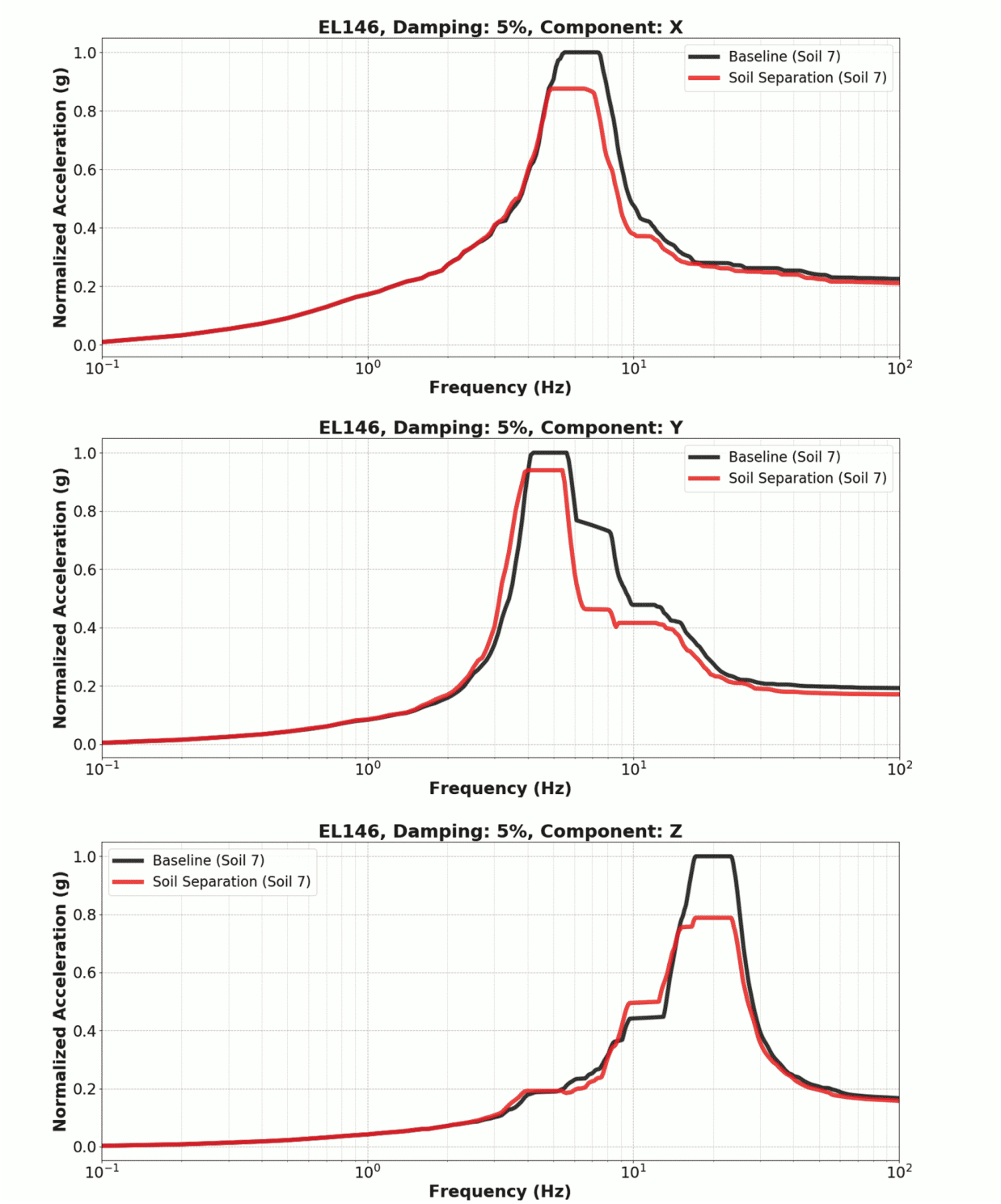
Note: Enveloped, broadened and normalized absolute acceleration ISRS comparison for the group of nodes at the elevation of 55 ft with 5% damping.

Figure 3.7.2-57: Soil Separation vs Baseline - Nodes at El. 100 ft



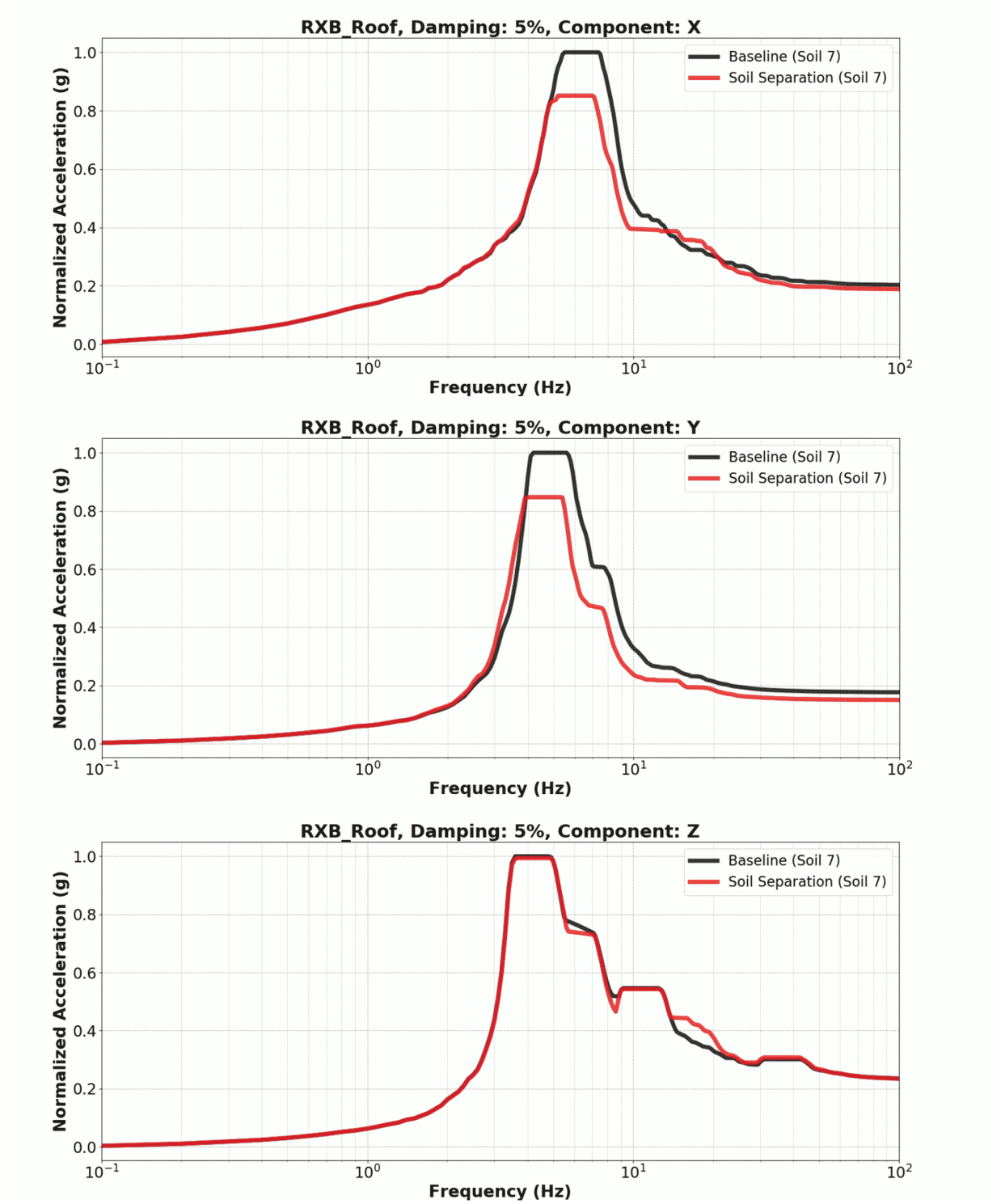
Note: Enveloped, broadened and normalized absolute acceleration ISRS comparison for the group of nodes at the elevation of 100 ft with 5% damping.

Figure 3.7.2-58: Soil Separation vs Baseline Nodes at El. 146 ft



Note: Enveloped, broadened and normalized absolute acceleration ISRS comparison for the group of nodes at the elevation of 146 ft with 5% damping.

Figure 3.7.2-59: Soil Separation vs Baseline Nodes at RXB roof



Note: Averaged, broadened and normalized absolute acceleration ISRS comparison for the group of nodes RXB roof with 5% damping.

Figure 3.7.2-60: Reactor Building ANSYS Model

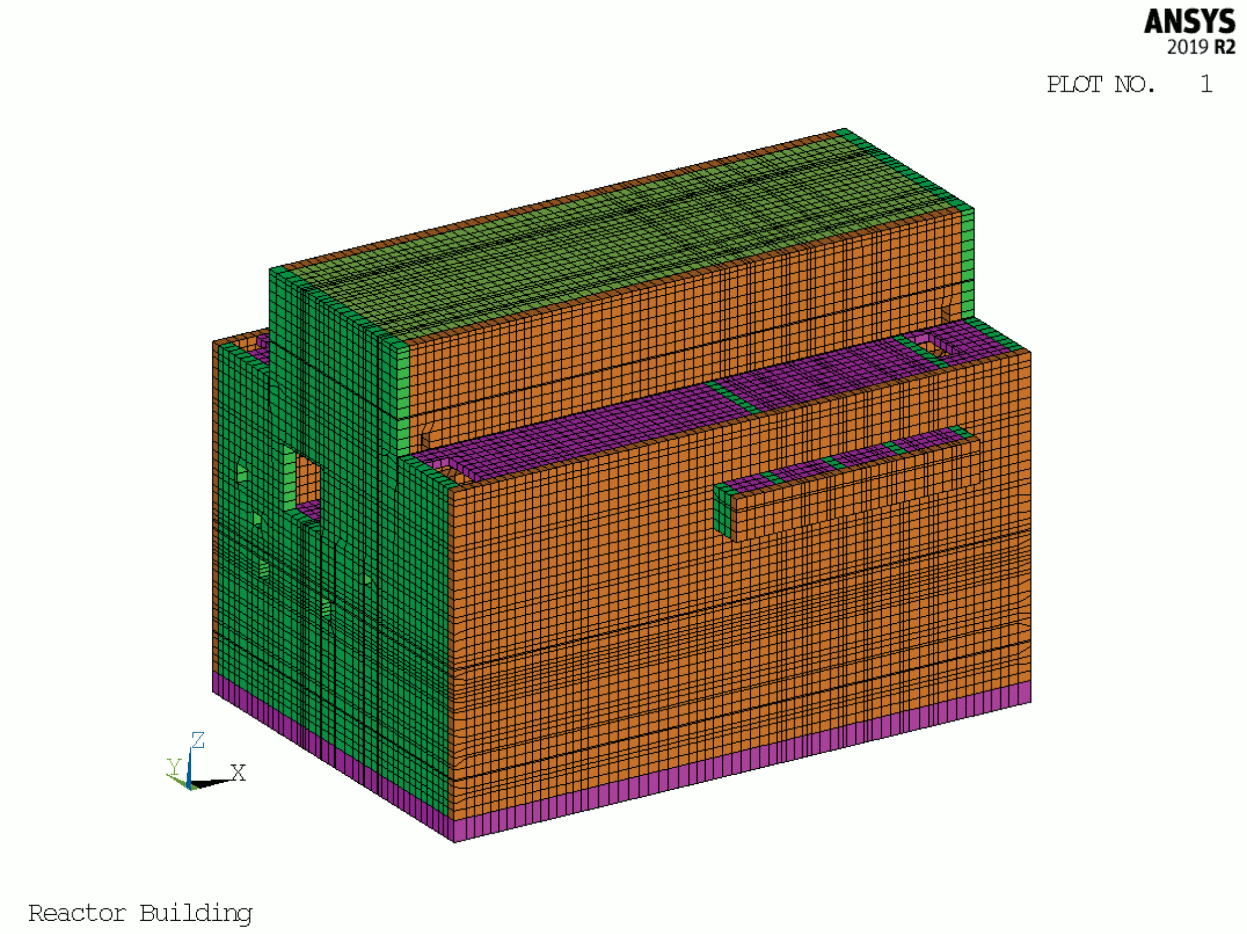


Figure 3.7.2-61: Reactor Building: (a) XZ Plane Cross-Section (b) YZ Plane Cross-Section (c) XY Plane Cross-Section (d) East-West Walls (e) North-South Walls (f) Basemat, Floors, Roof

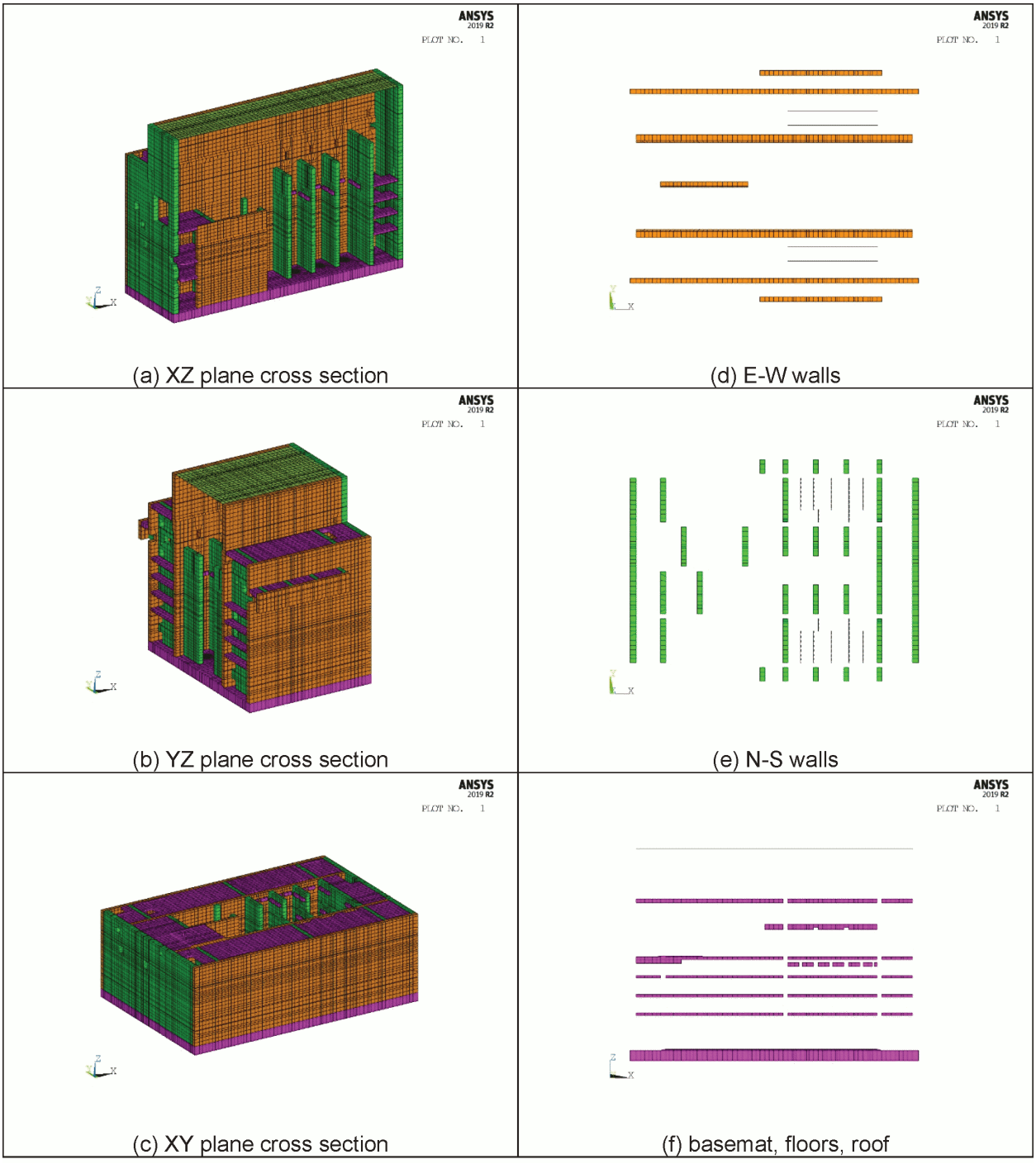


Figure 3.7.2-62: Reactor Building Basemat at Elevation 25 ft

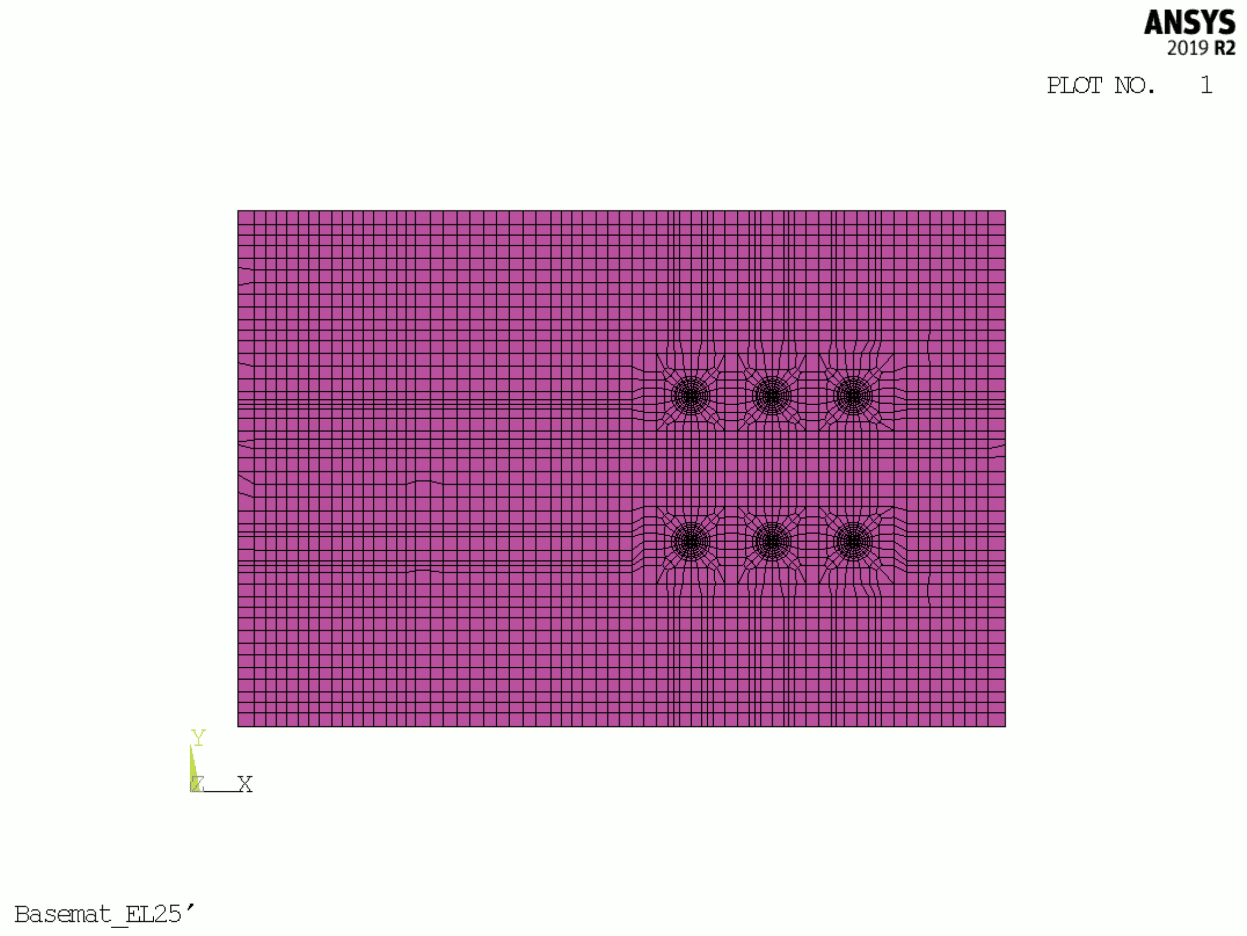


Figure 3.7.2-63: Reactor Building Floor Elevation 40 ft

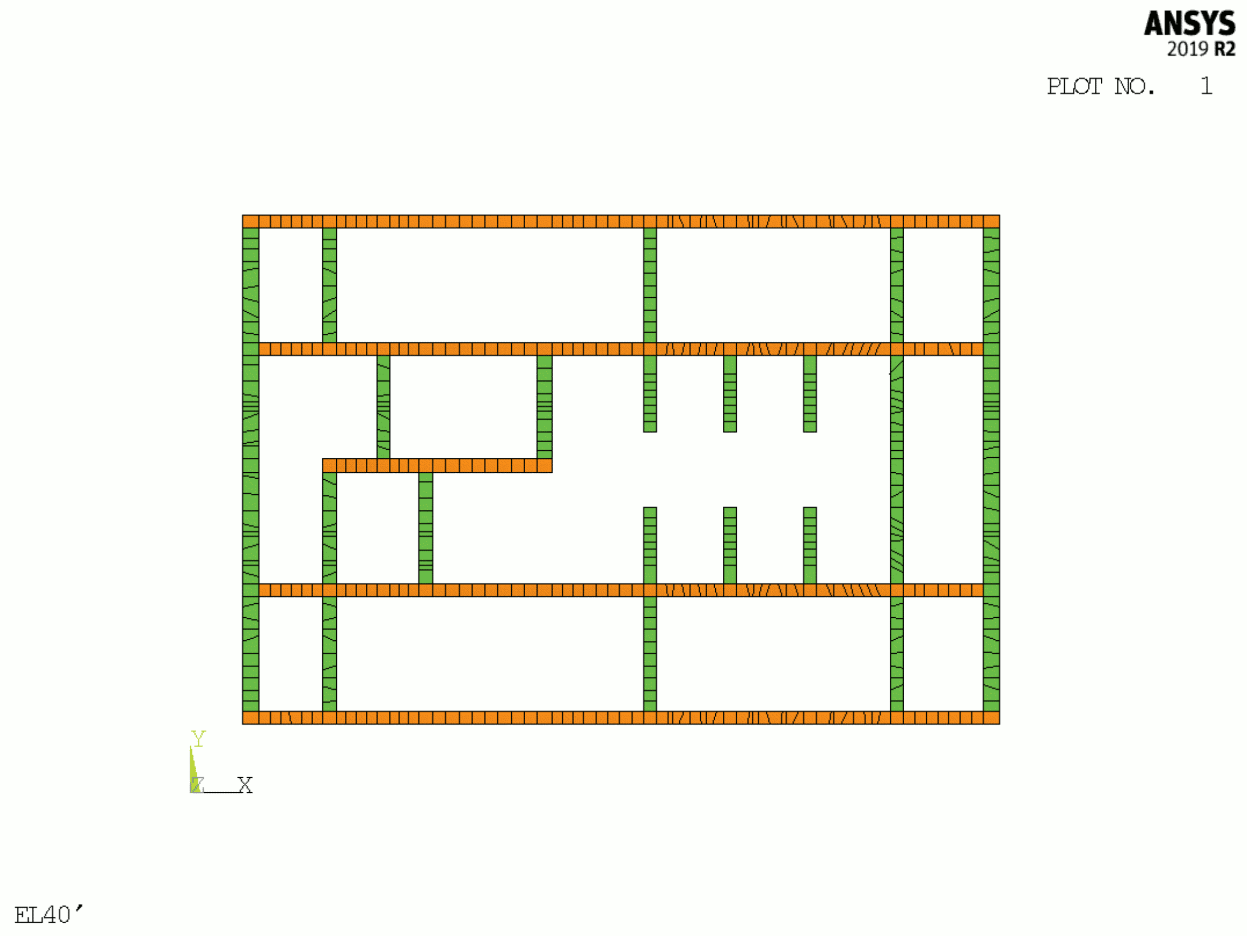


Figure 3.7.2-64: Reactor Building Floor Elevation 55 ft

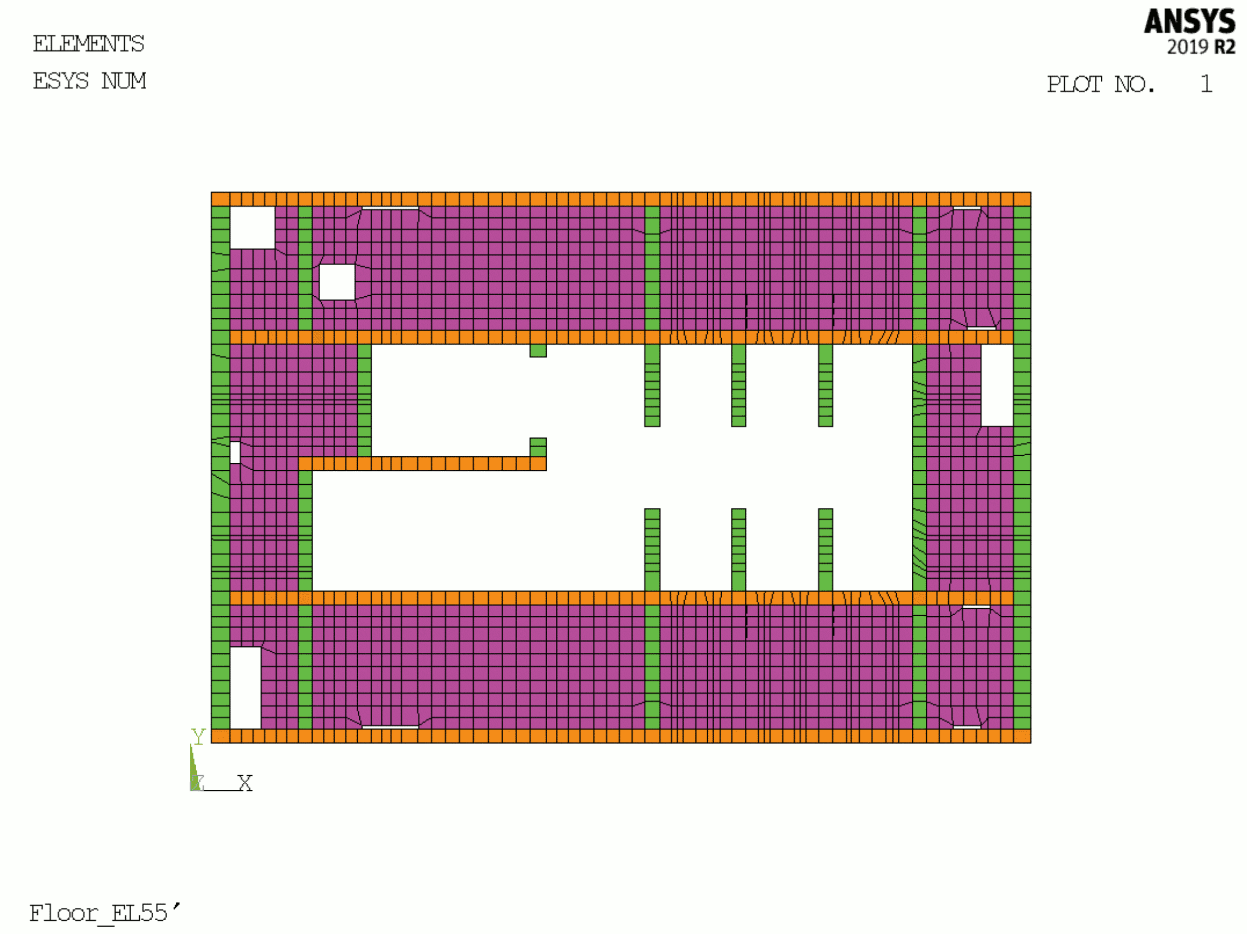


Figure 3.7.2-65: Reactor Building Floor Elevation 70 ft

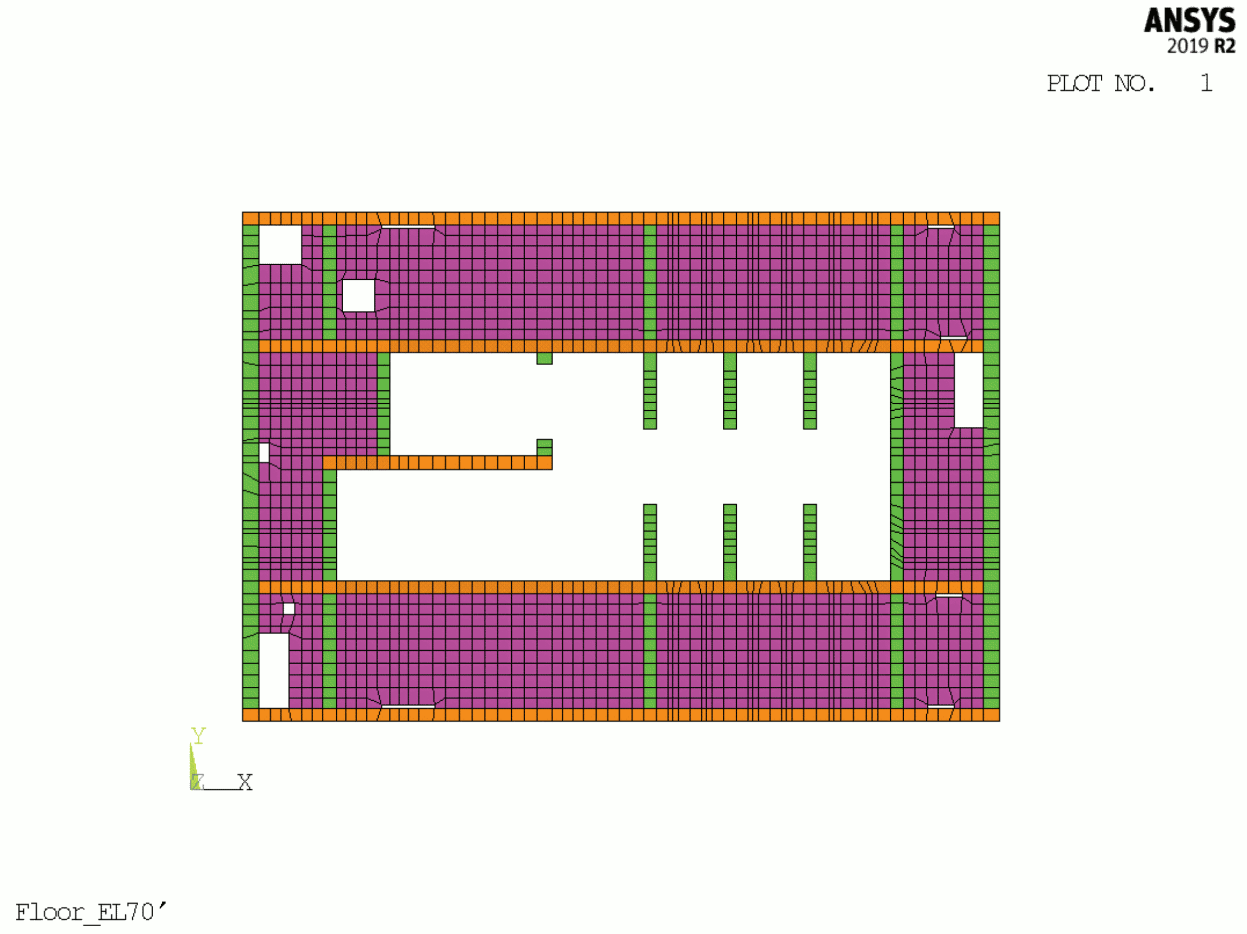


Figure 3.7.2-66: Reactor Building Floor Elevation 81 ft

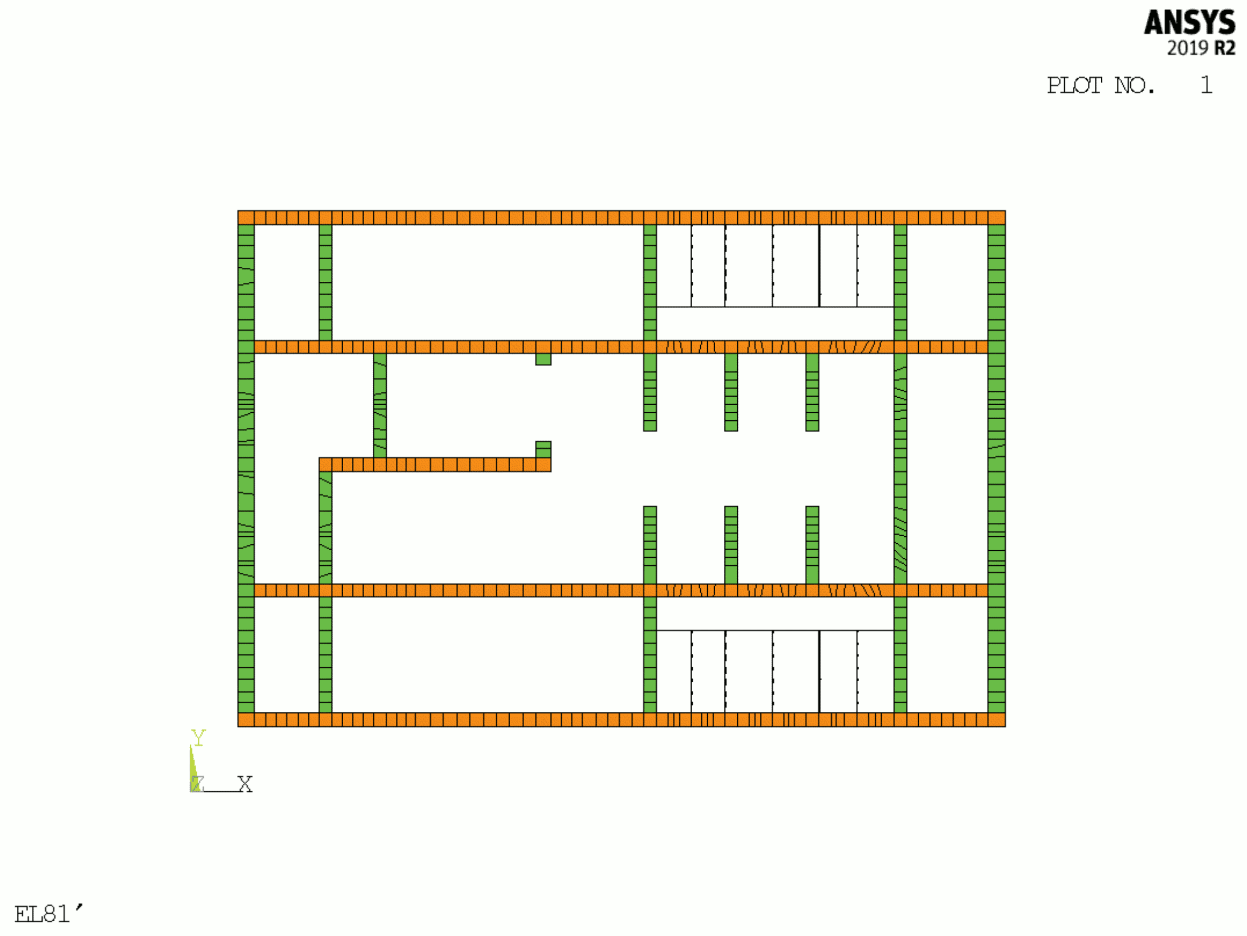


Figure 3.7.2-67: Reactor Building Floor Elevation 85 ft



Figure 3.7.2-68: Reactor Building Floor Elevation 100 ft

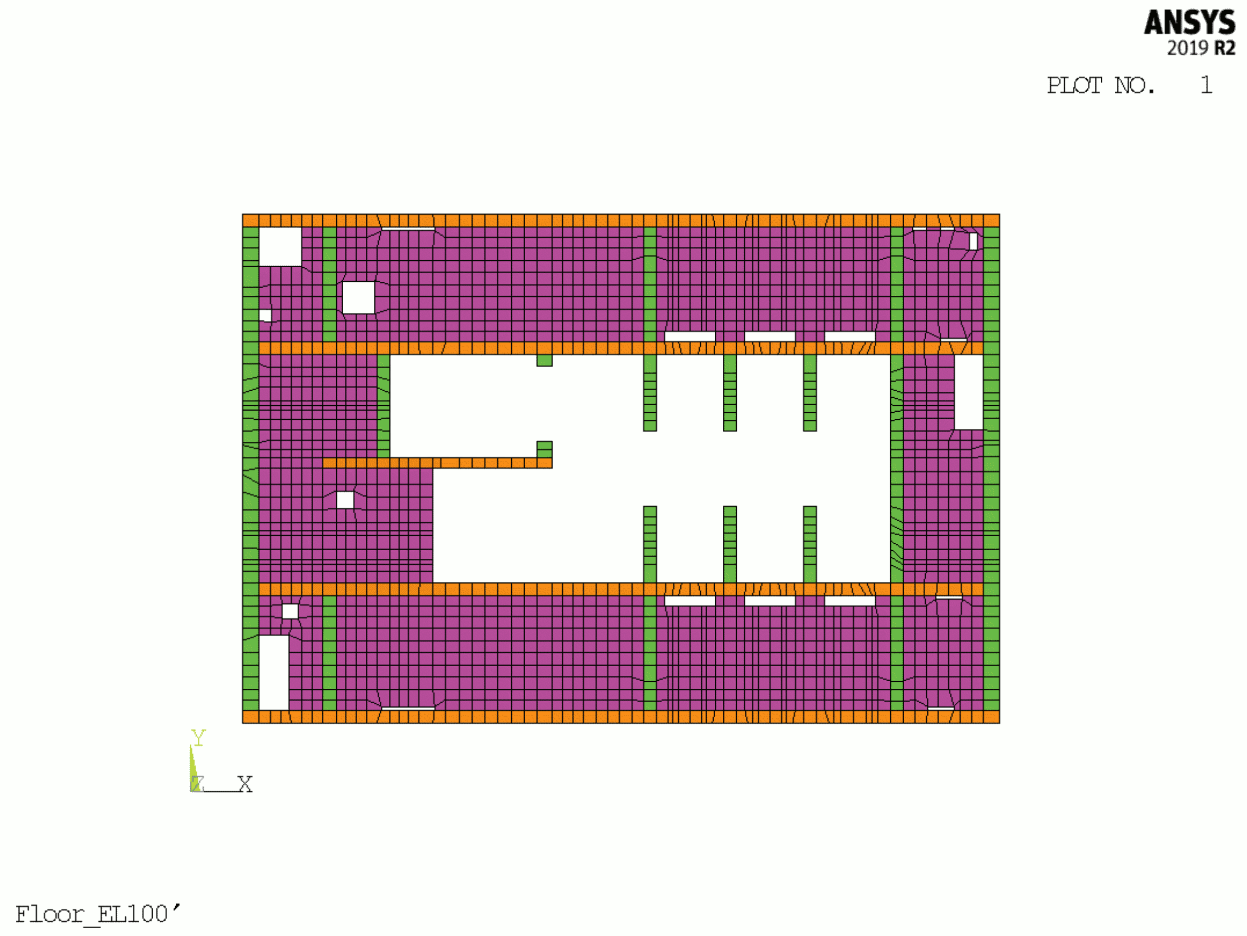


Figure 3.7.2-69: Reactor Building Floor Elevation 126 ft

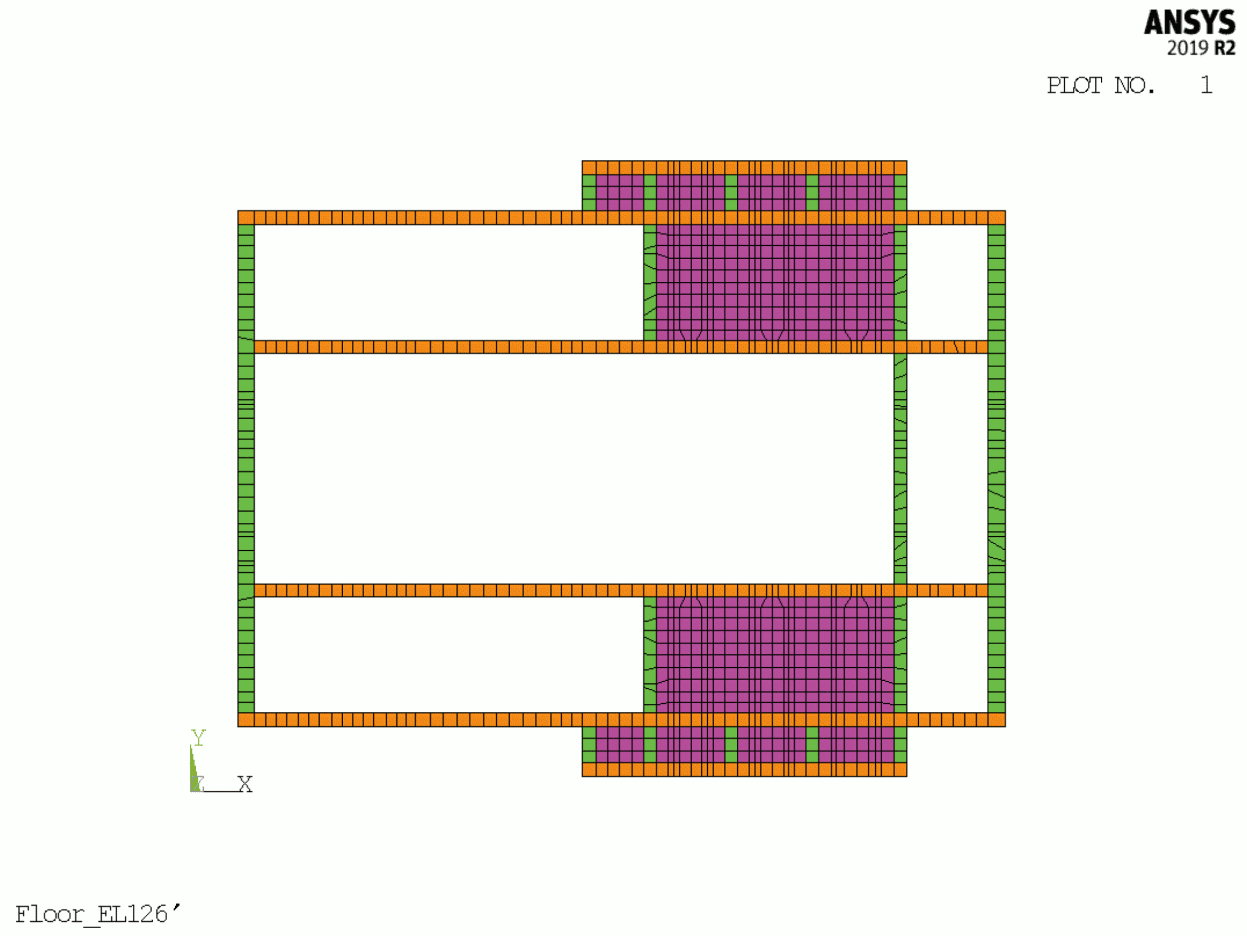


Figure 3.7.2-70: Reactor Building Floor Elevation 146.5 ft

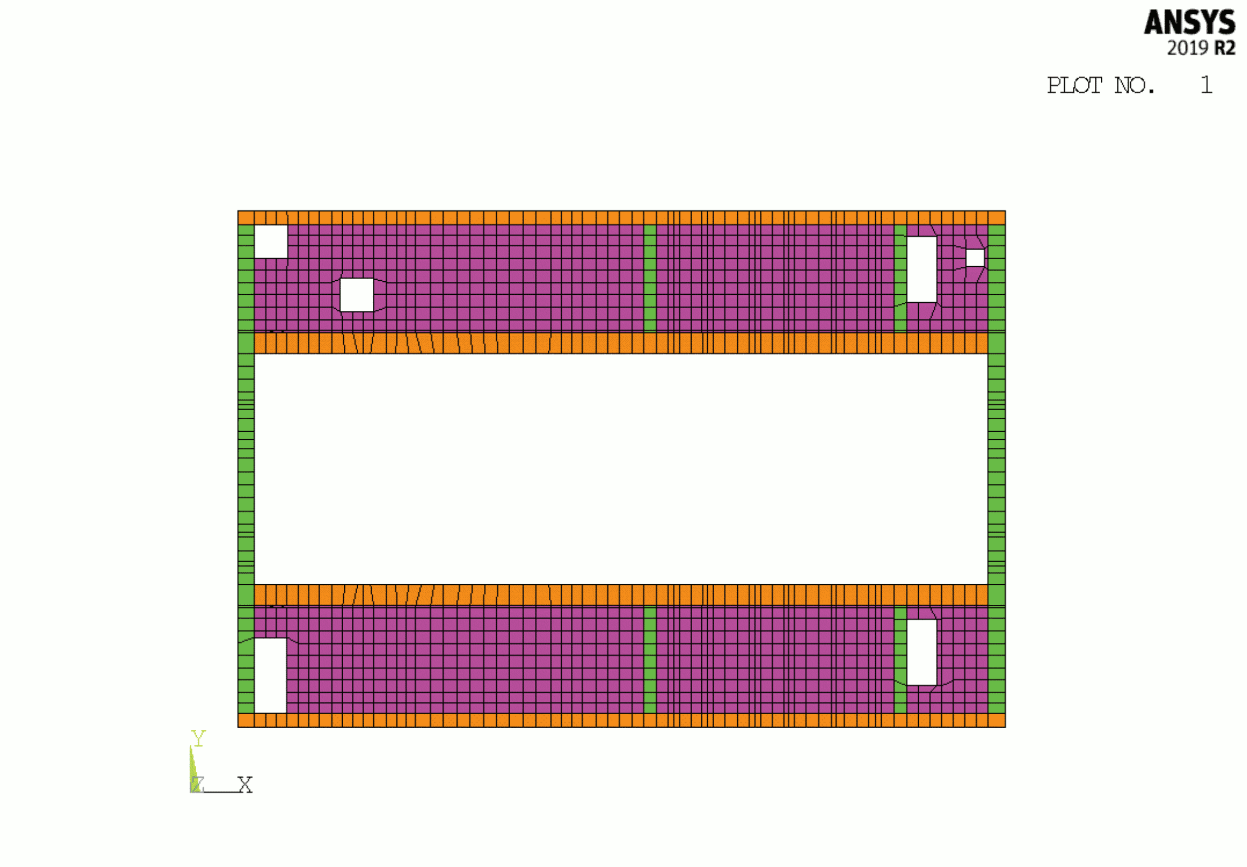


Figure 3.7.2-71: Reactor Building Roof at Elevation 187.5 ft

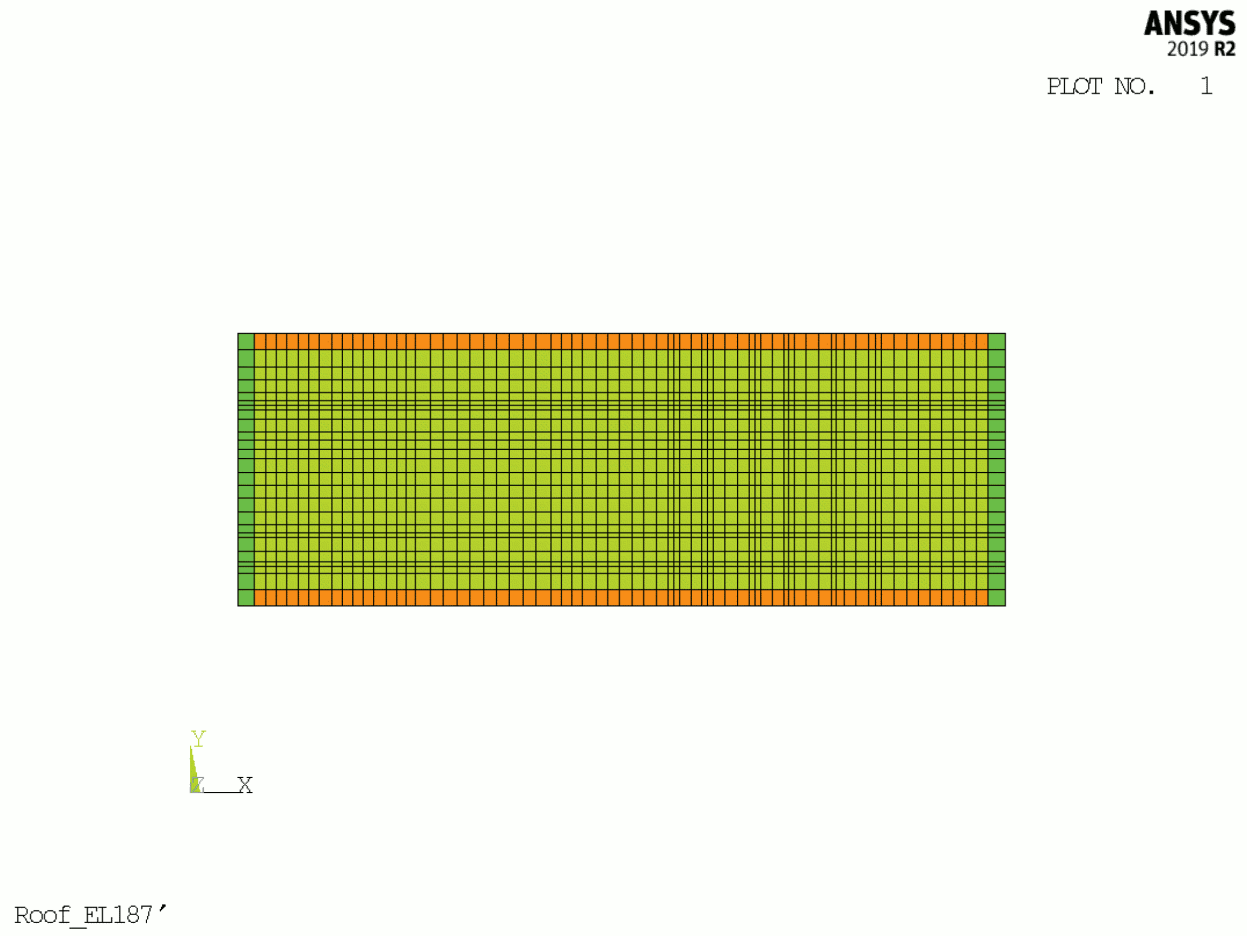


Figure 3.7.2-72: Reactor Building East-West Wall at Gridline A, Elevation View

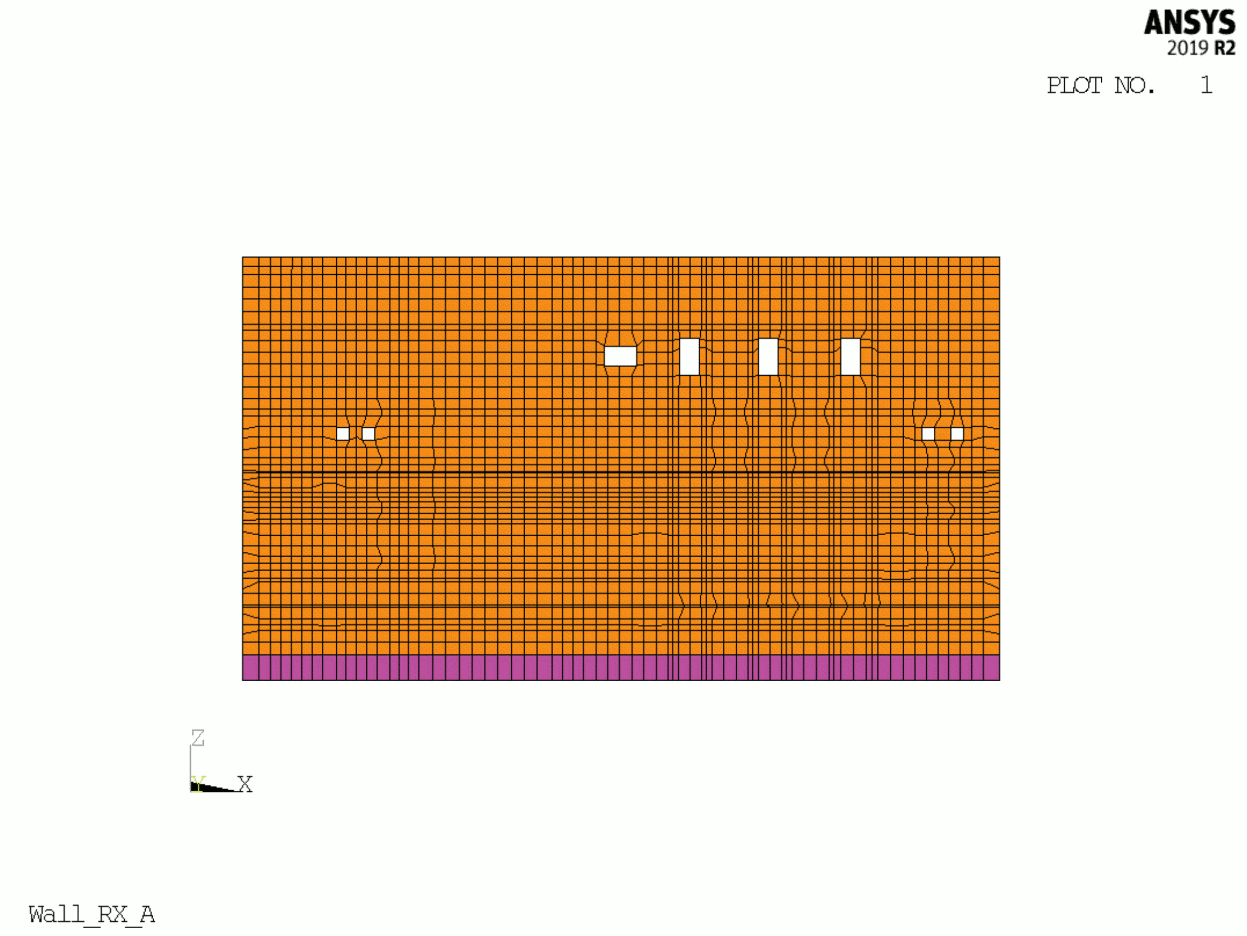


Figure 3.7.2-73: Reactor Building East-West Wall at Gridline B, Elevation View

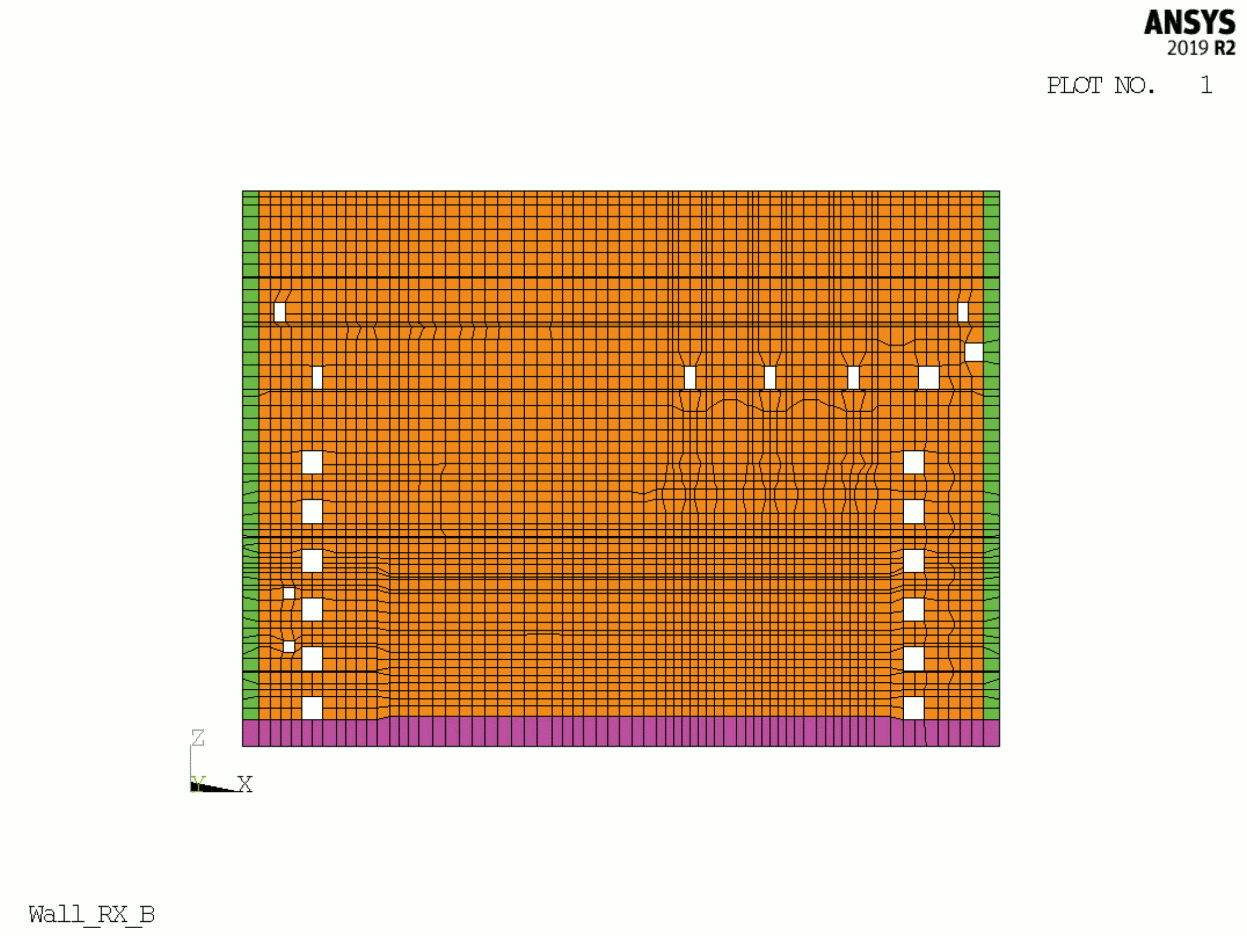


Figure 3.7.2-74: Reactor Building East-West Wall at Gridline C, Elevation View

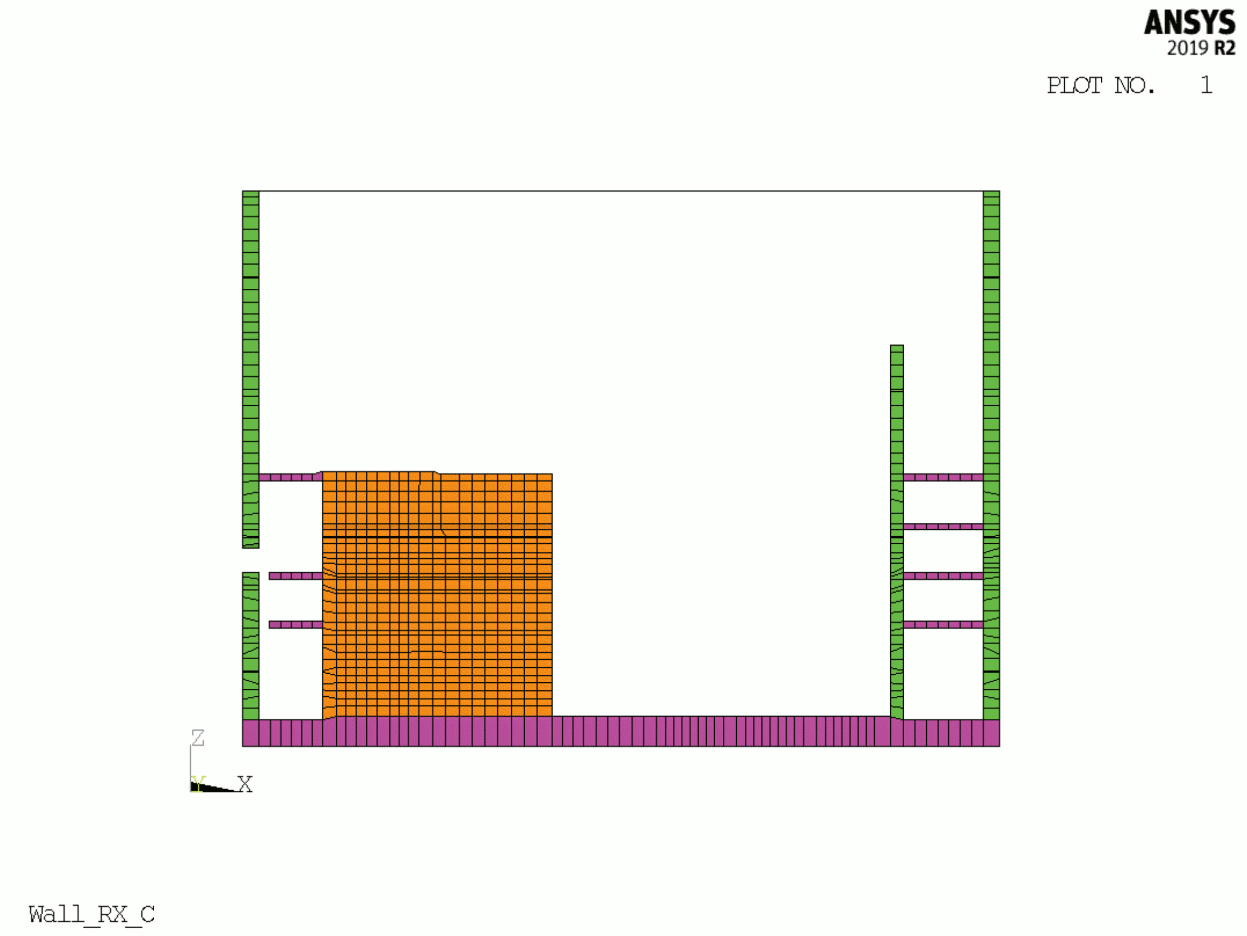


Figure 3.7.2-75: Reactor Building East-West Wall at Gridline D, Elevation View

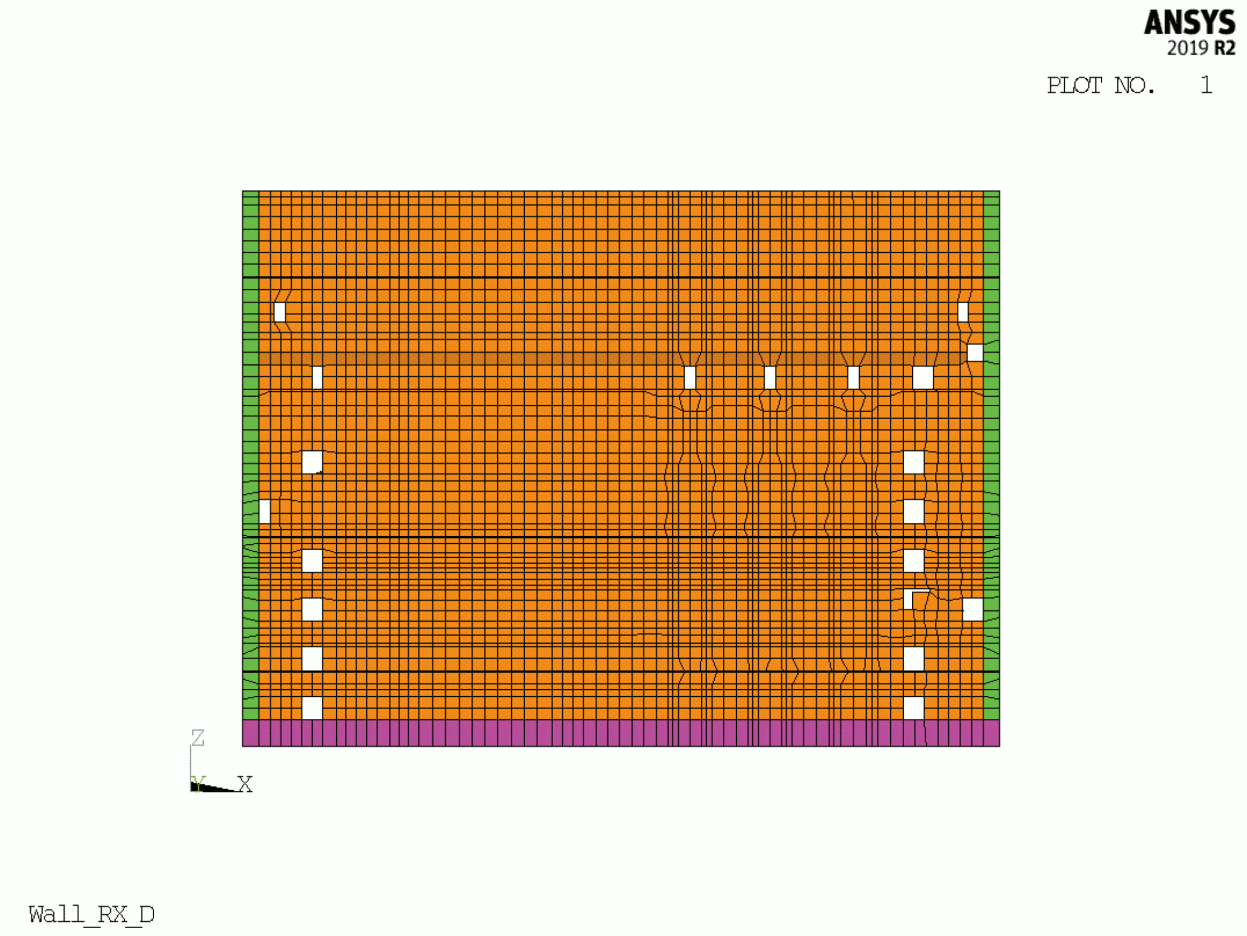


Figure 3.7.2-76: Reactor Building East-West Wall at Gridline E, Elevation View

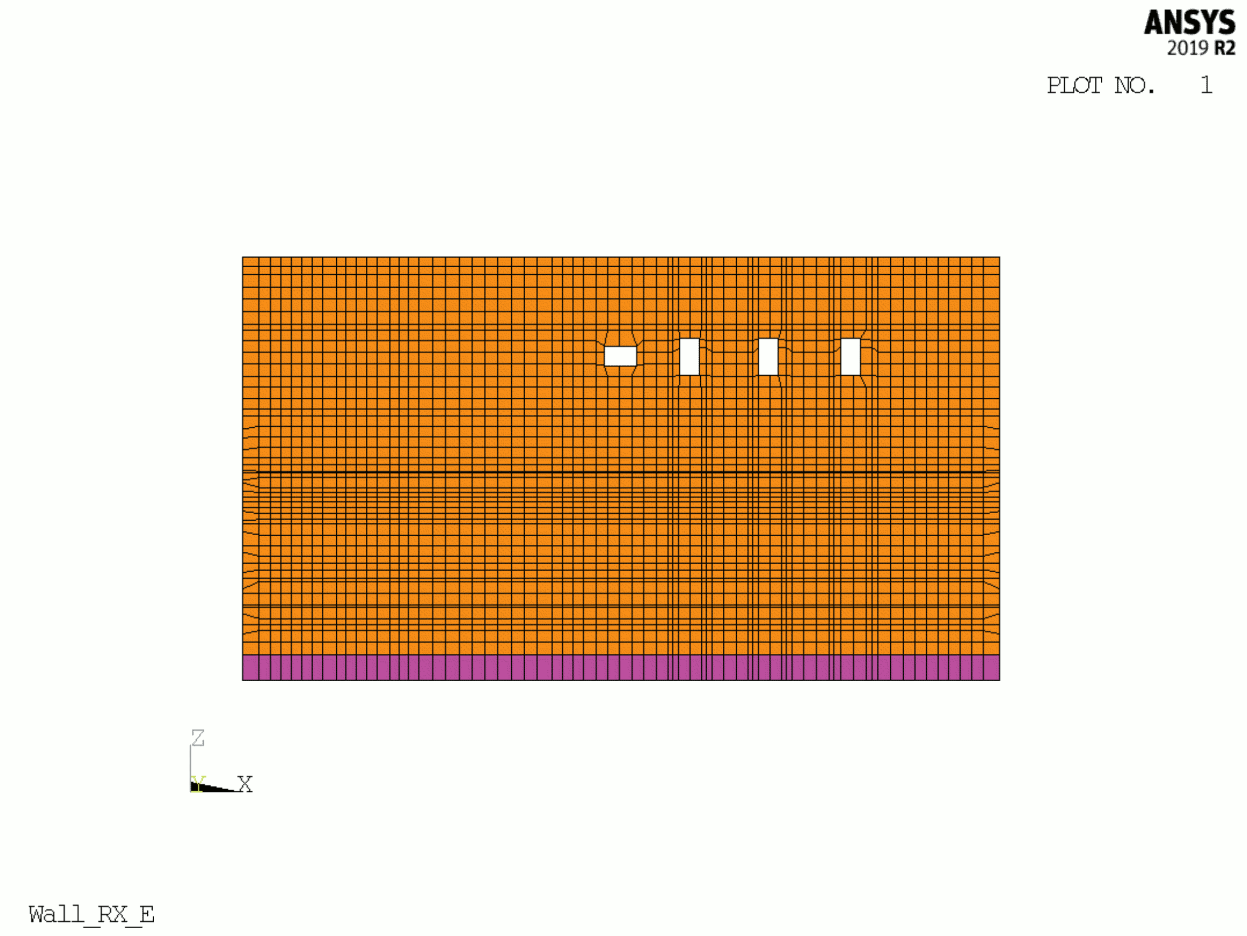


Figure 3.7.2-77: Reactor Building North-South Wall at Gridline 1, Elevation View

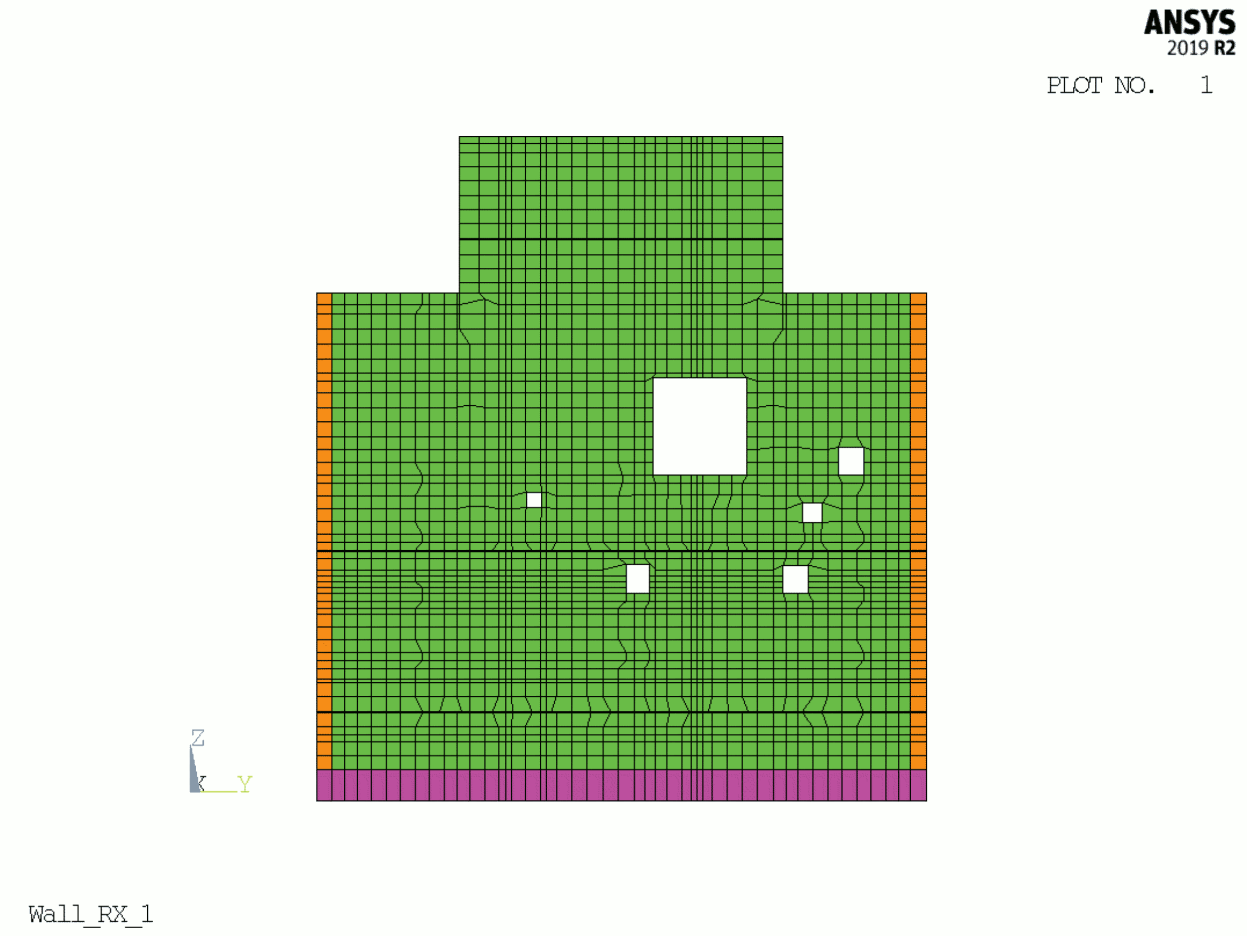


Figure 3.7.2-78: Reactor Building North-South Wall at Gridline 2, Elevation View

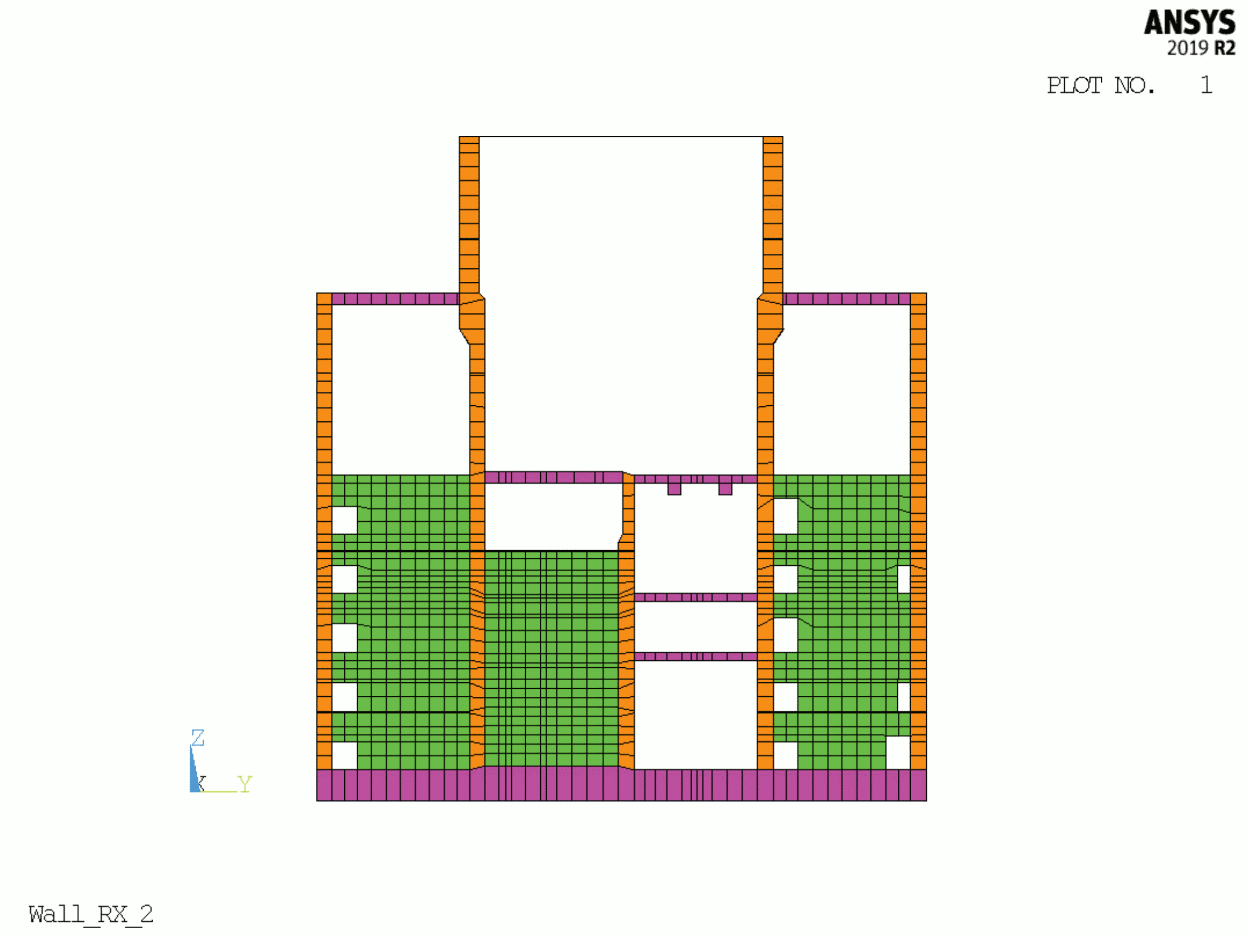


Figure 3.7.2-79: Reactor Building North-South Wall at Gridline 2.2, Elevation View

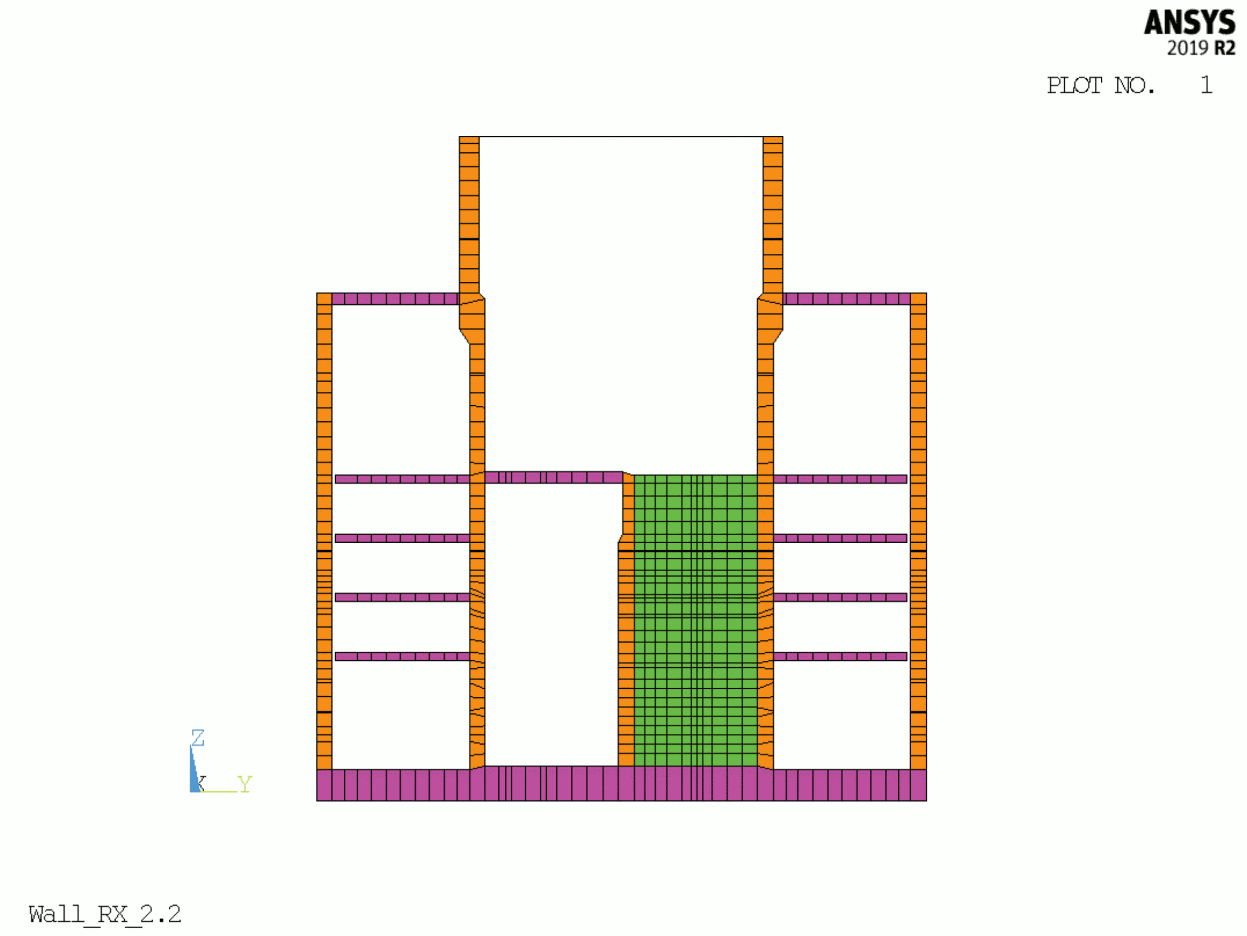


Figure 3.7.2-80: Reactor Building North-South Wall at Gridline 2.4, Elevation View

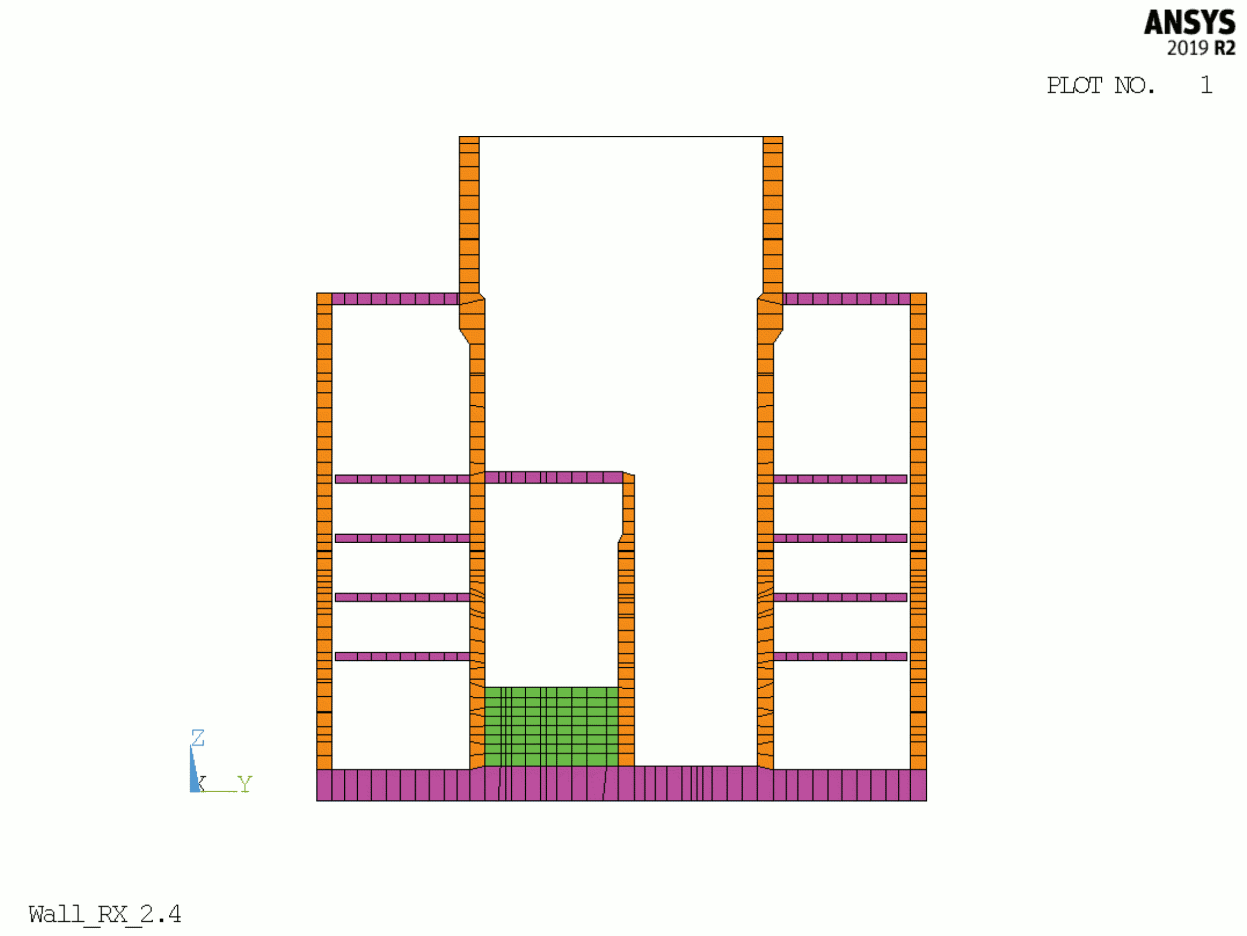


Figure 3.7.2-81: Reactor Building North-South Wall at Gridline 3, Elevation View

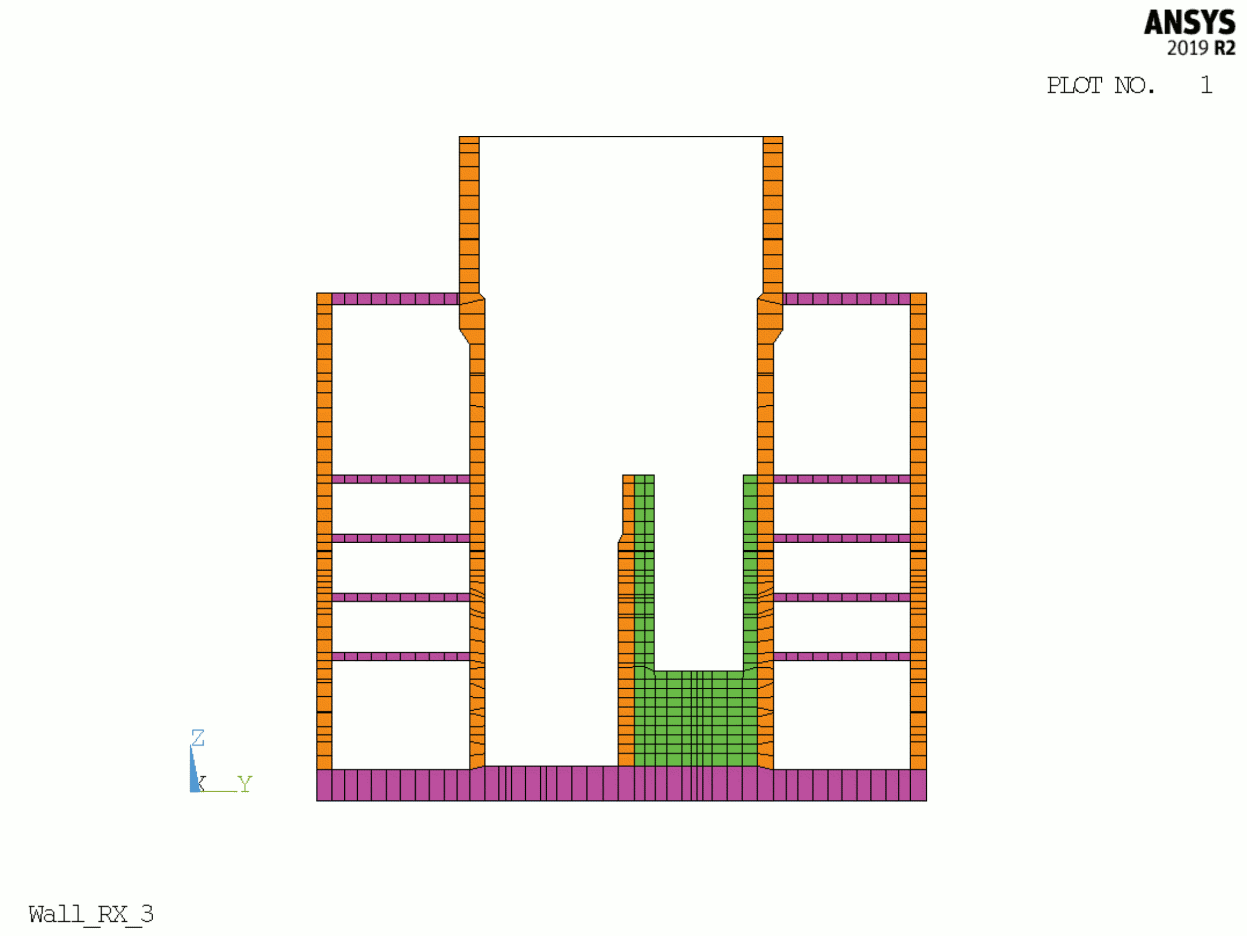


Figure 3.7.2-82: Reactor Building North-South Wall at Gridline 4, Elevation View

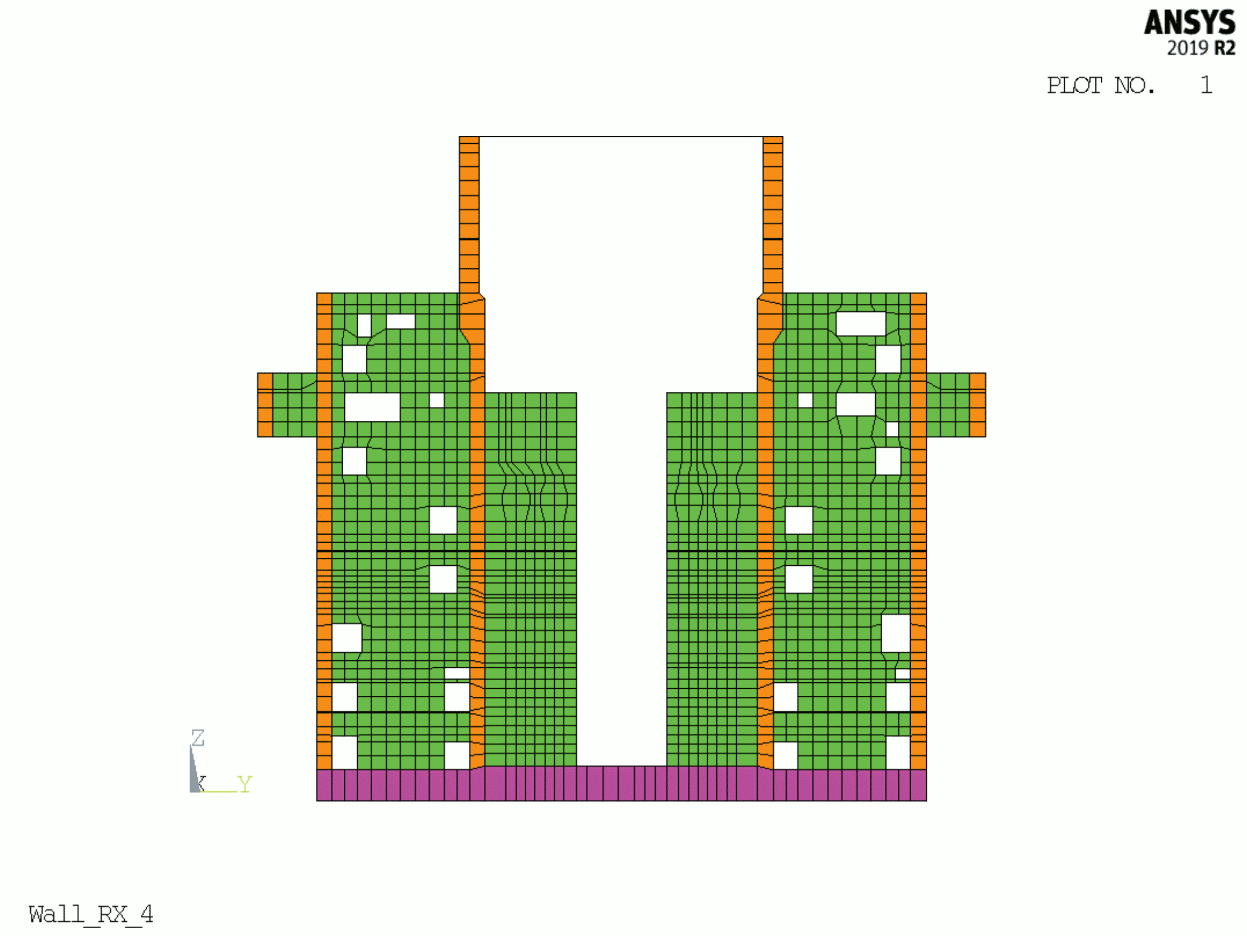


Figure 3.7.2-83: Reactor Building North-South Wall at Gridline 4.3, Elevation View

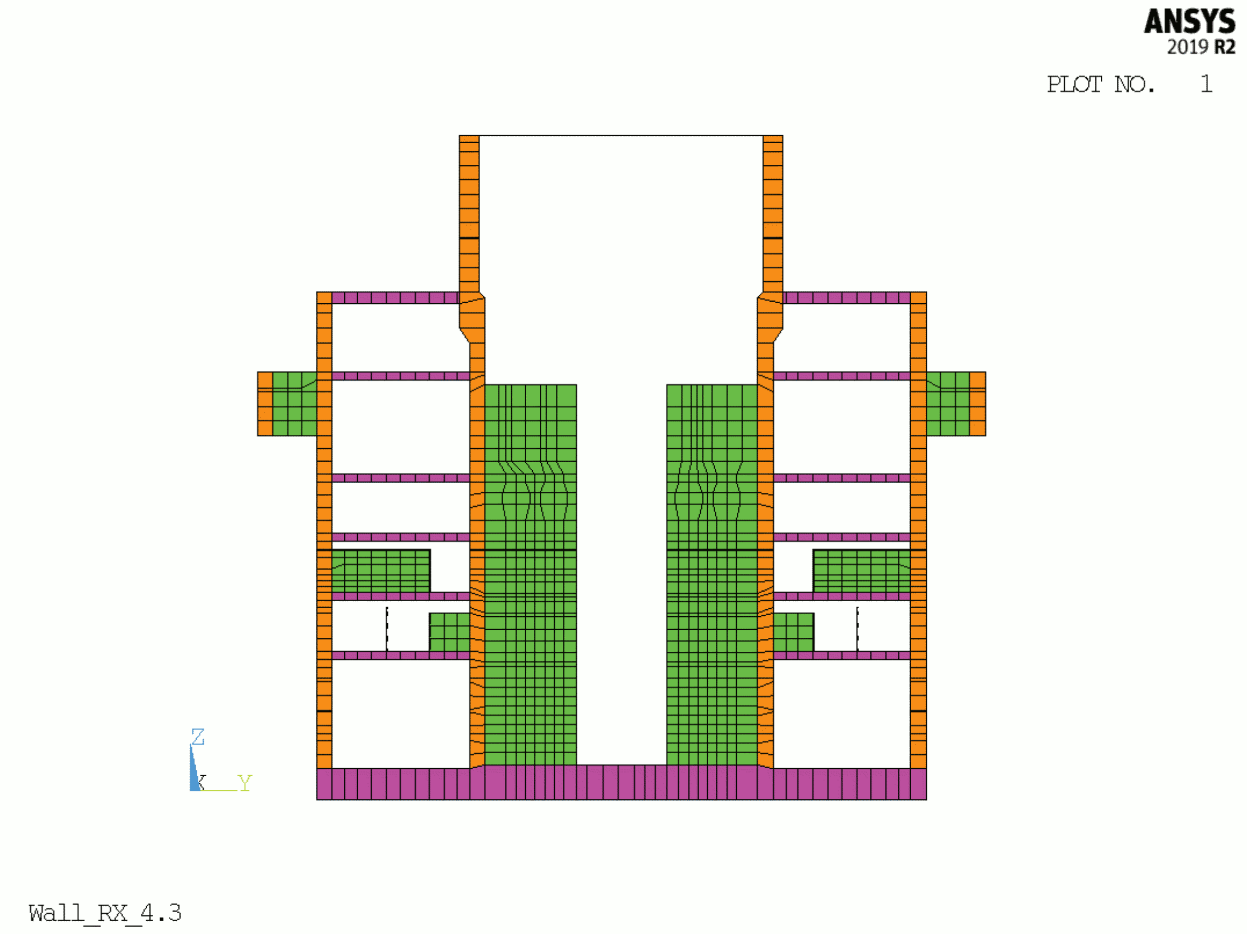


Figure 3.7.2-84: Reactor Building North-South Wall at Gridline 4.6, Elevation View

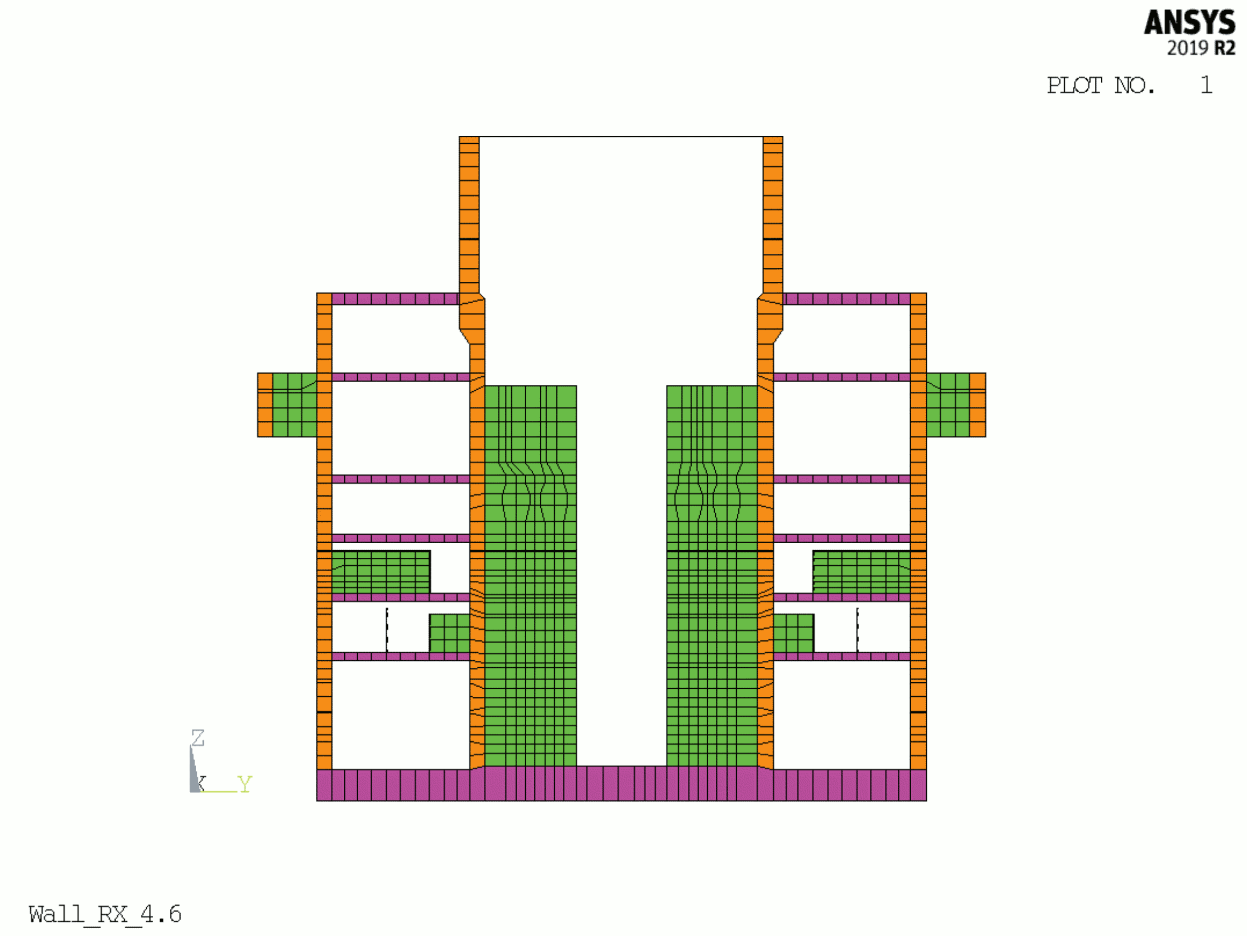


Figure 3.7.2-85: Reactor Building North-South Wall at Gridline 5, Elevation View

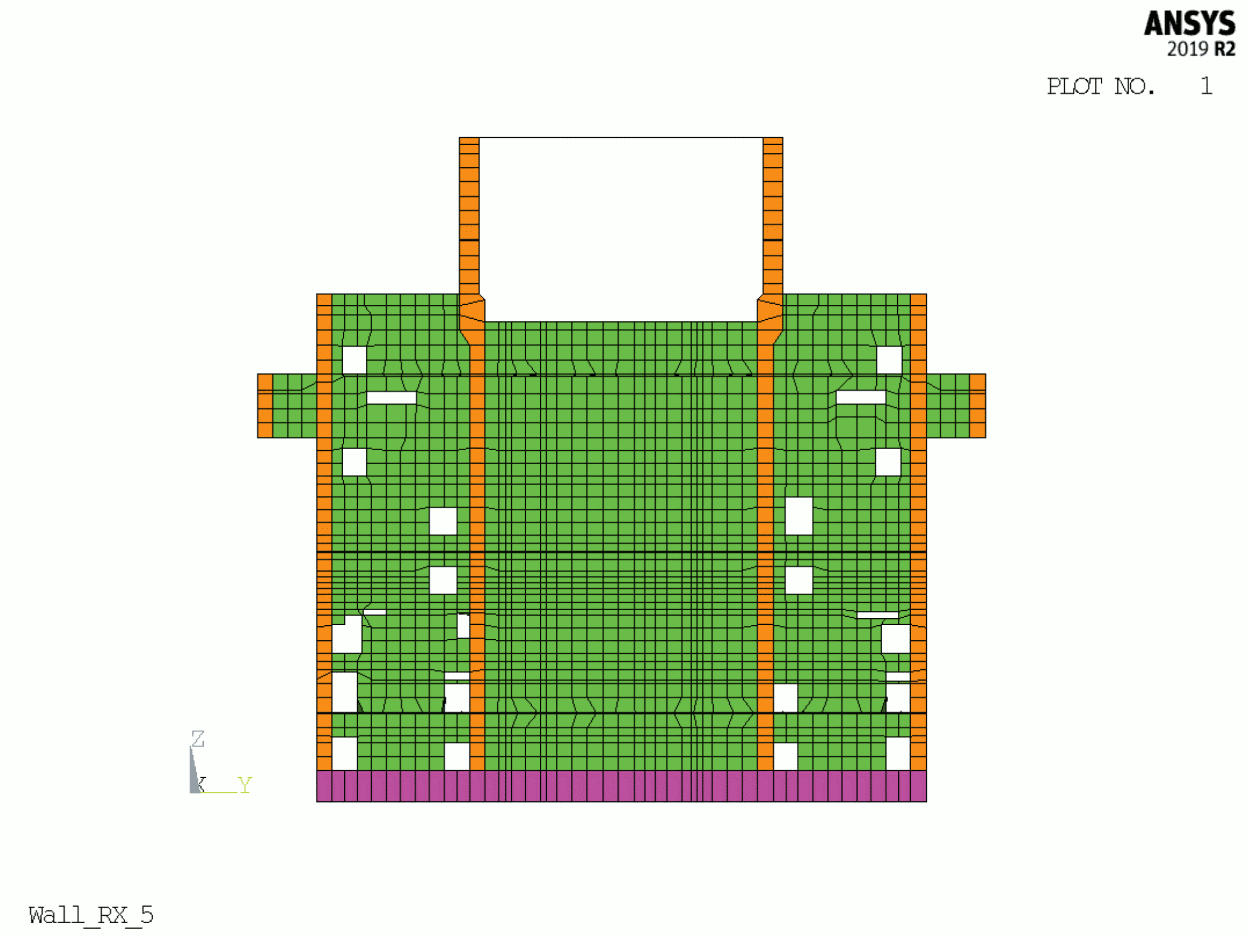


Figure 3.7.2-86: Reactor Building North-South Wall at Gridline 6, Elevation View

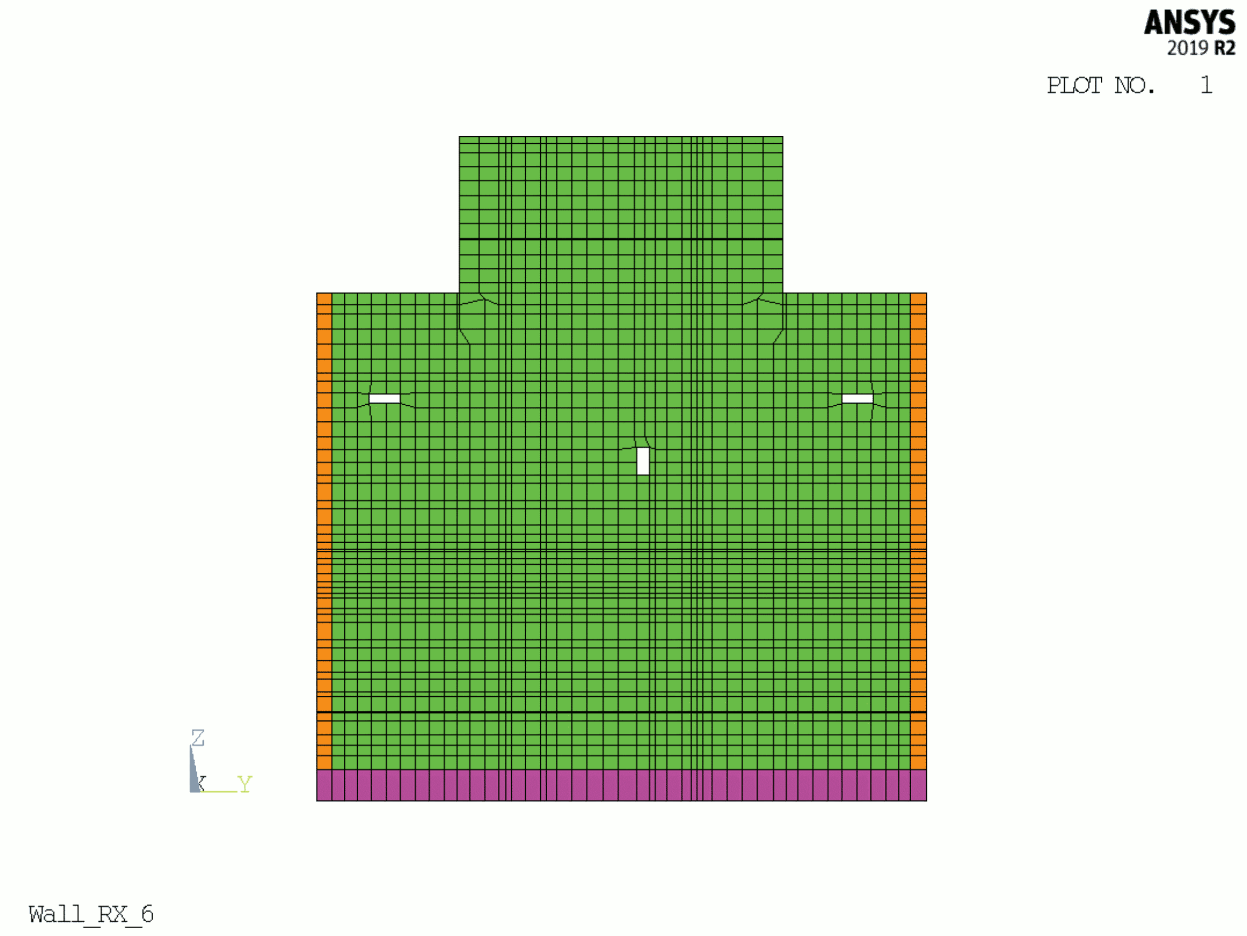


Figure 3.7.2-87: Reactor Building Penetrations Shrouds Component

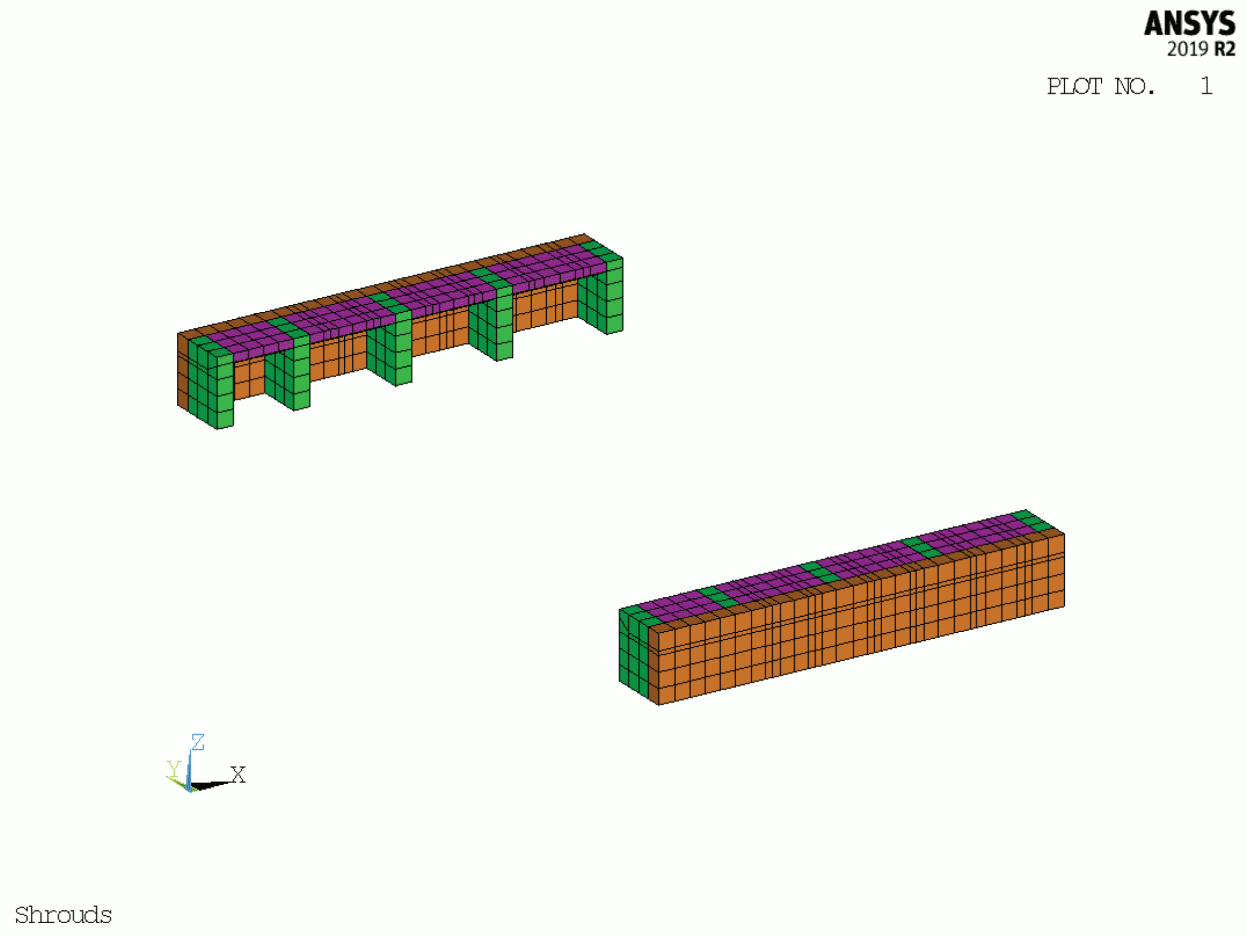


Figure 3.7.2-88: Selected Nodes on the Reactor Building Basemat at Elevation 25' for In-Structure Response Spectra and Relative Displacement Calculation

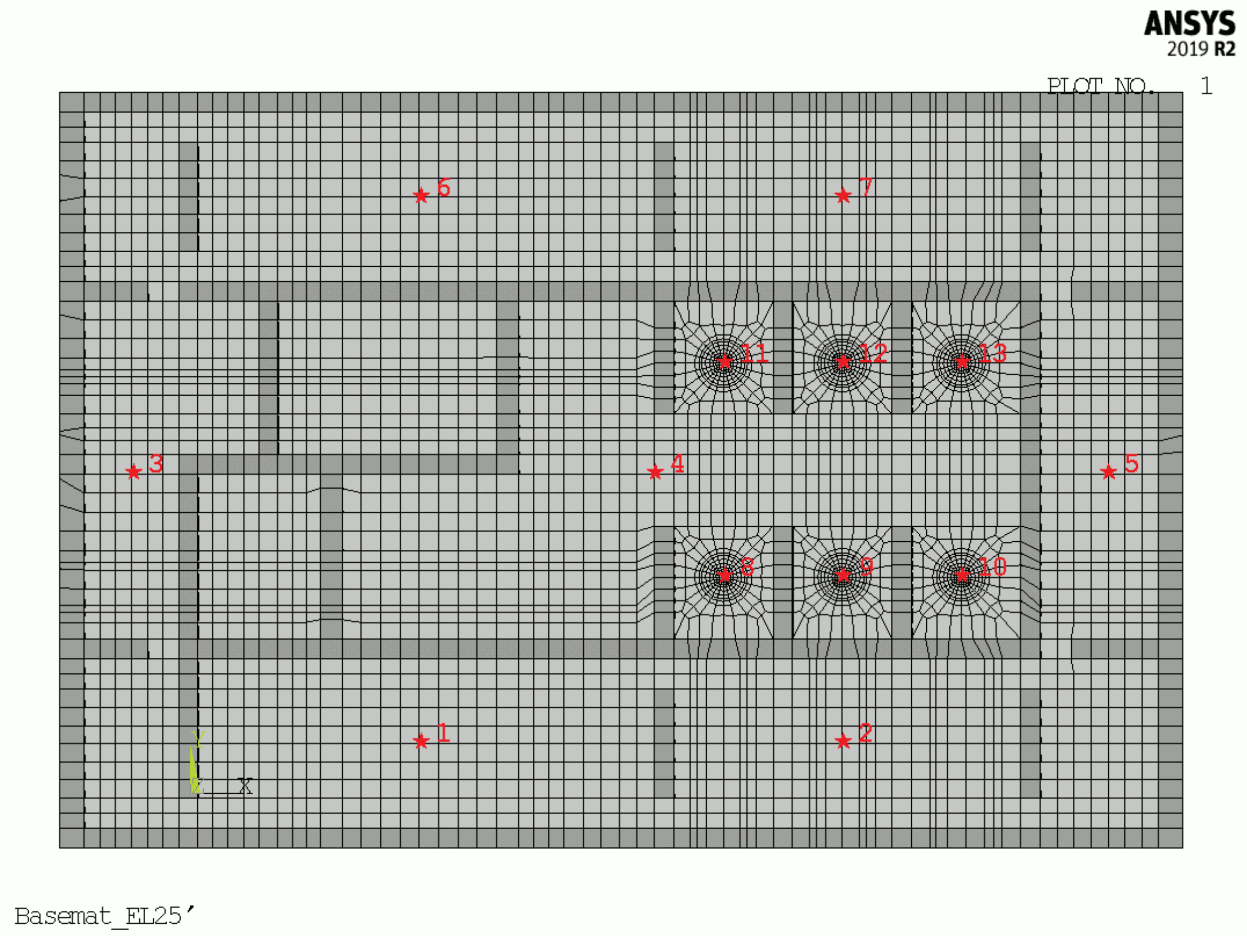


Figure 3.7.2-89: Selected Nodes on the Reactor Building for NuScale Power Module Lug Supports for In-Structure Response Spectra and Relative Displacement Calculation

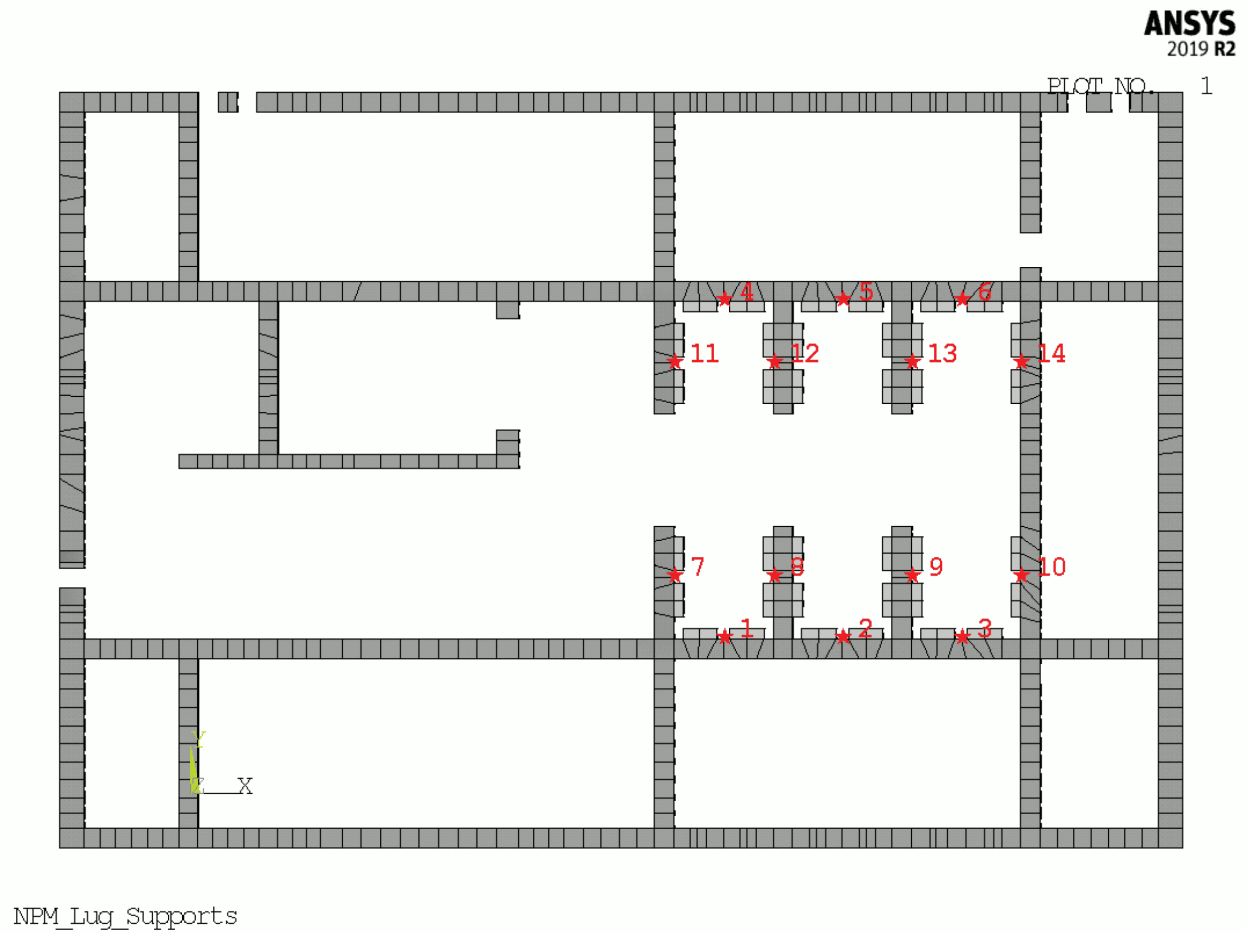


Figure 3.7.2-90: East-West, North-South, and Vertical In-Structure Response Spectra due to Certified Seismic Design Response Spectra for Reactor Building Basemat at EL 25'

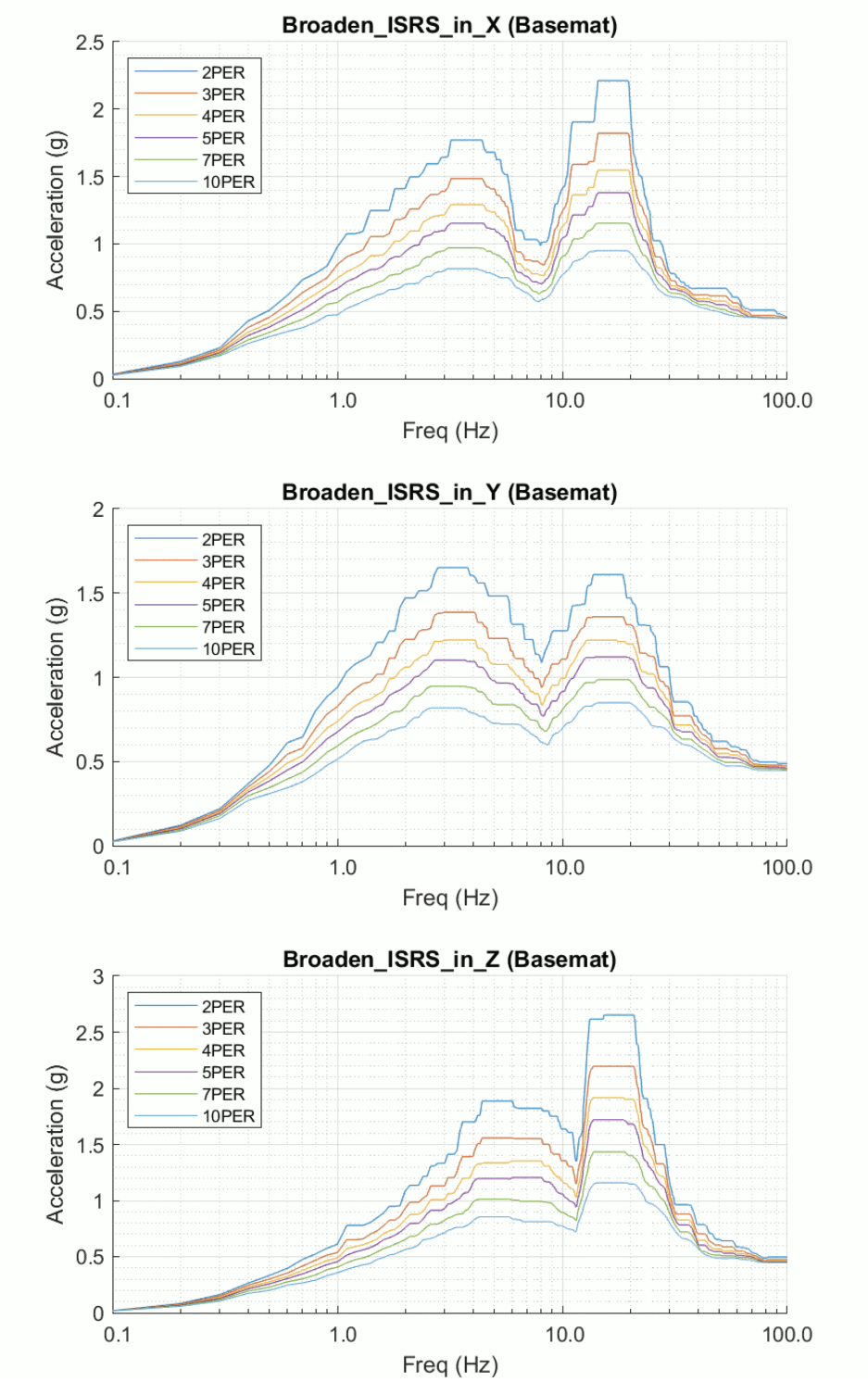


Figure 3.7.2-91: East-West, North-South, and Vertical In-Structure Response Spectra due to Certified Seismic Design Response Spectra for NuScale Power Module Base at EL 25'

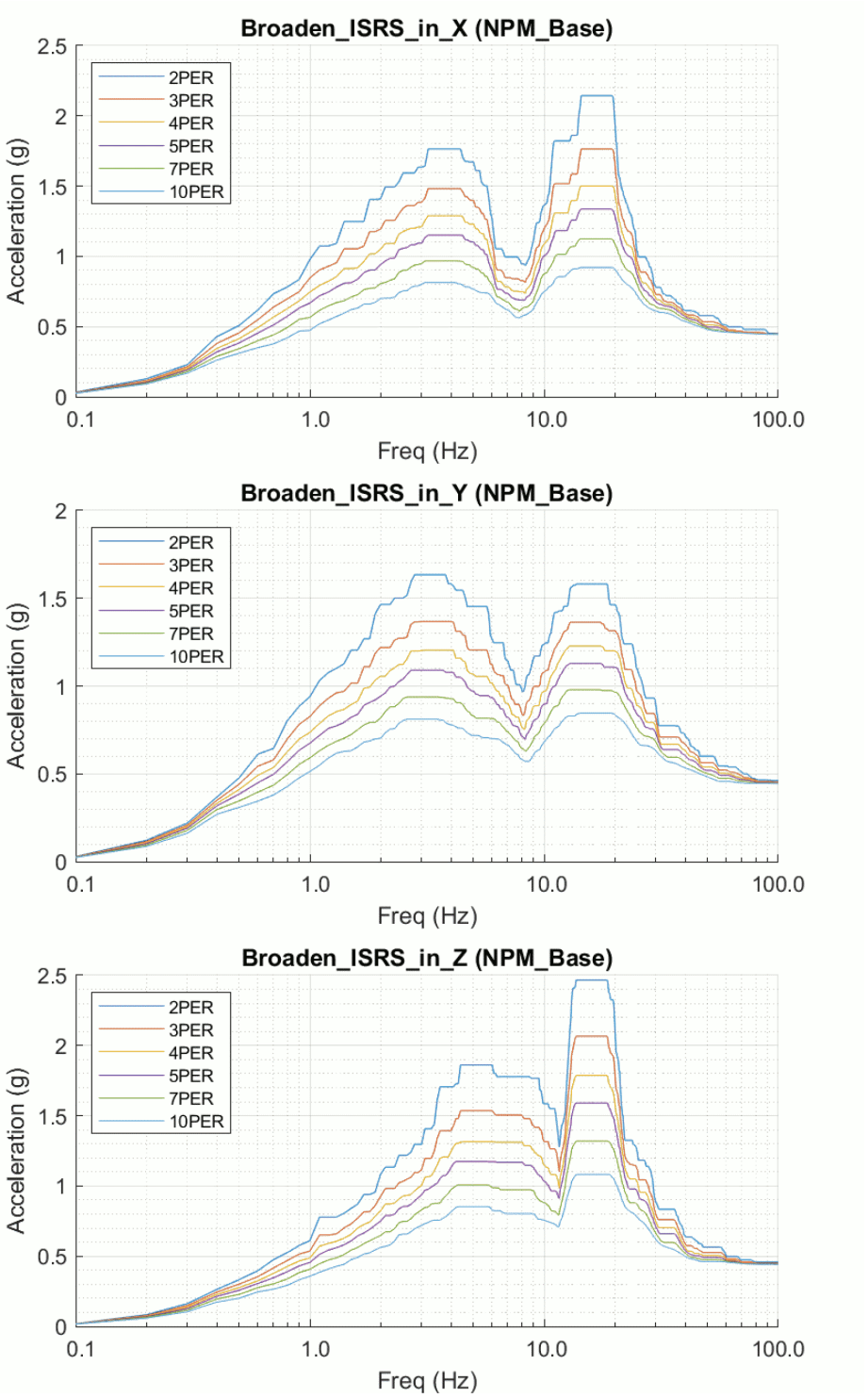


Figure 3.7.2-92: East-West, North-South, and Vertical In-Structure Response Spectra due to Certified Seismic Design Response Spectra - High Frequency for Reactor Building Basemat at Elevation 25'

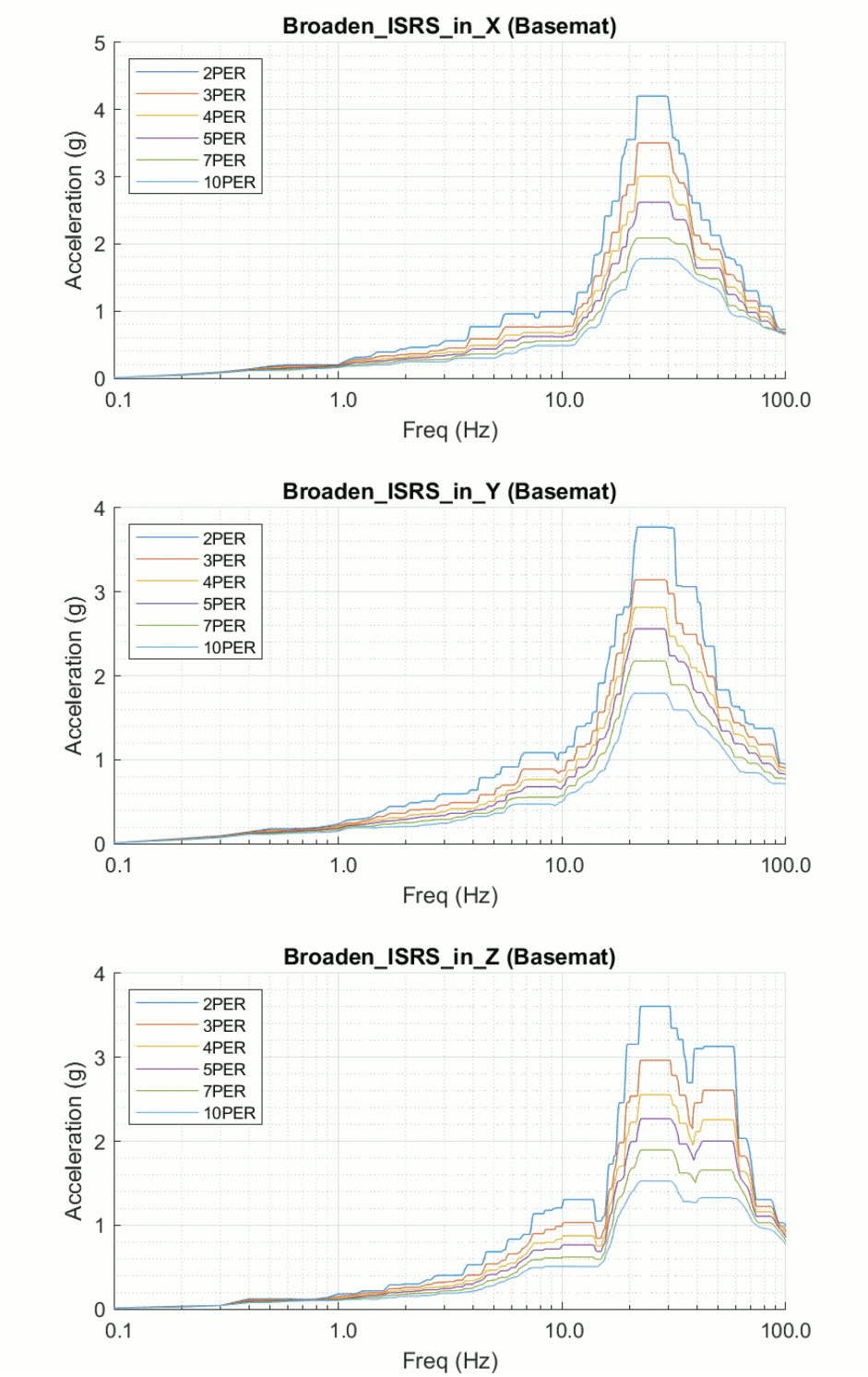


Figure 3.7.2-93: East-West, North-South, and vertical In-Structure Response Spectra due to Certified Seismic Design Response Spectra - High Frequency for NuScale Power Module Base at Elevation 25'

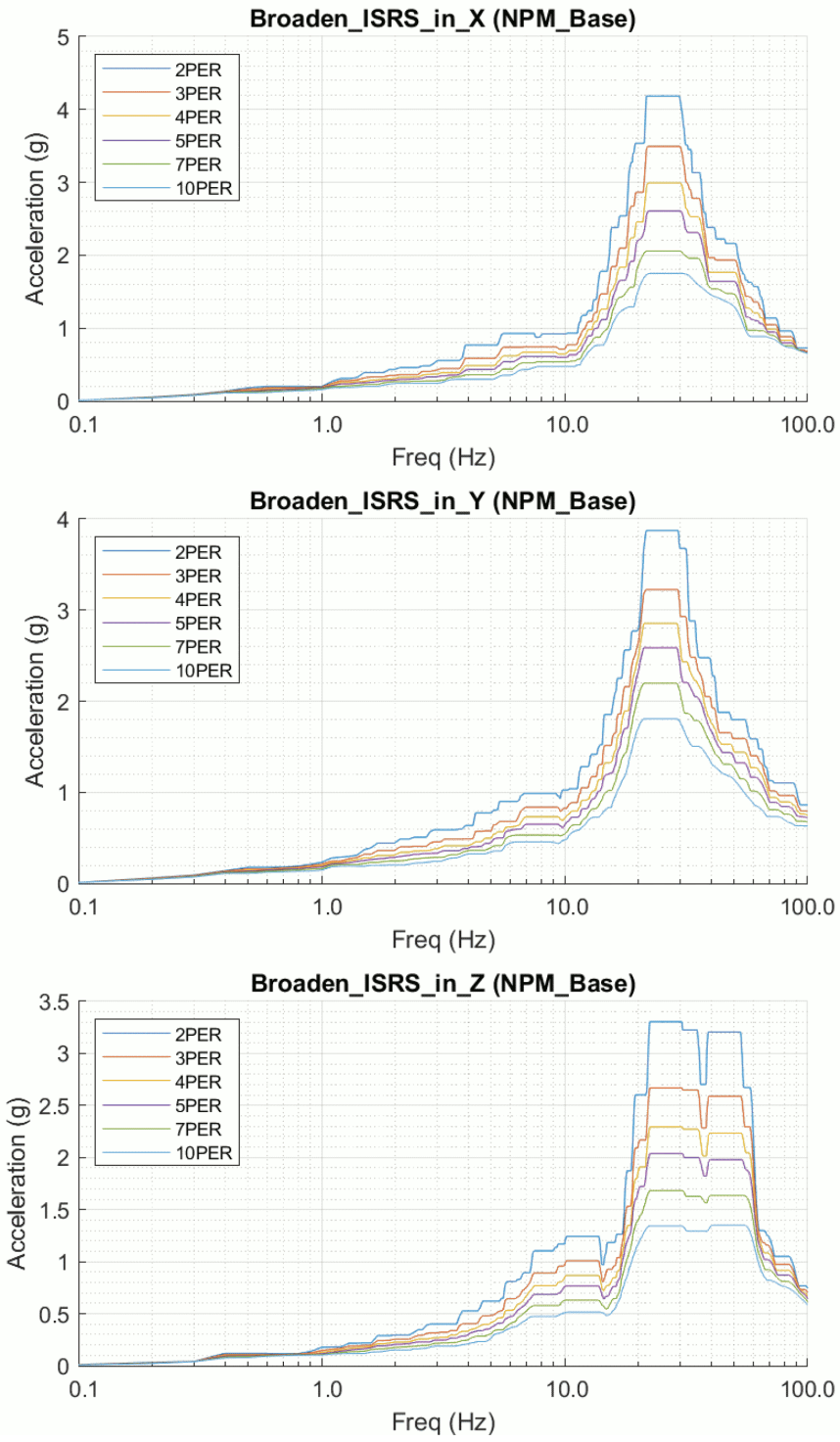


Figure 3.7.2-94: East-West, North-South, and Vertical In-Structure Response Spectra due to Certified Seismic Design Response Spectra for NuScale Power Module Lug Supports (East-West) at Elevation 95'

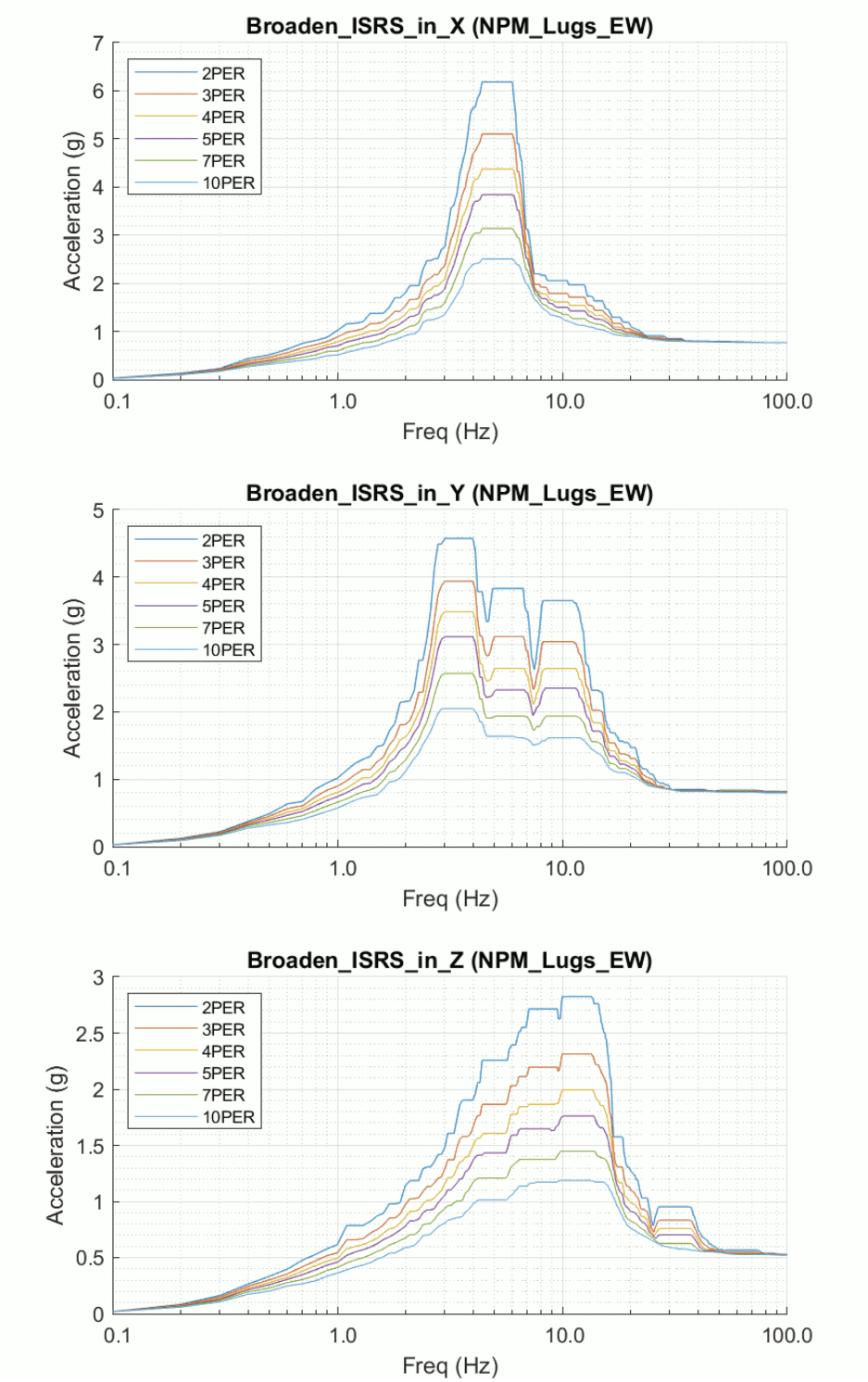


Figure 3.7.2-95: East-West, North-South, and Vertical In-Structure Response Spectra due to Certified Seismic Design Response Spectra for NuScale Power Module Lug Supports (North-South) at Elevation 95'

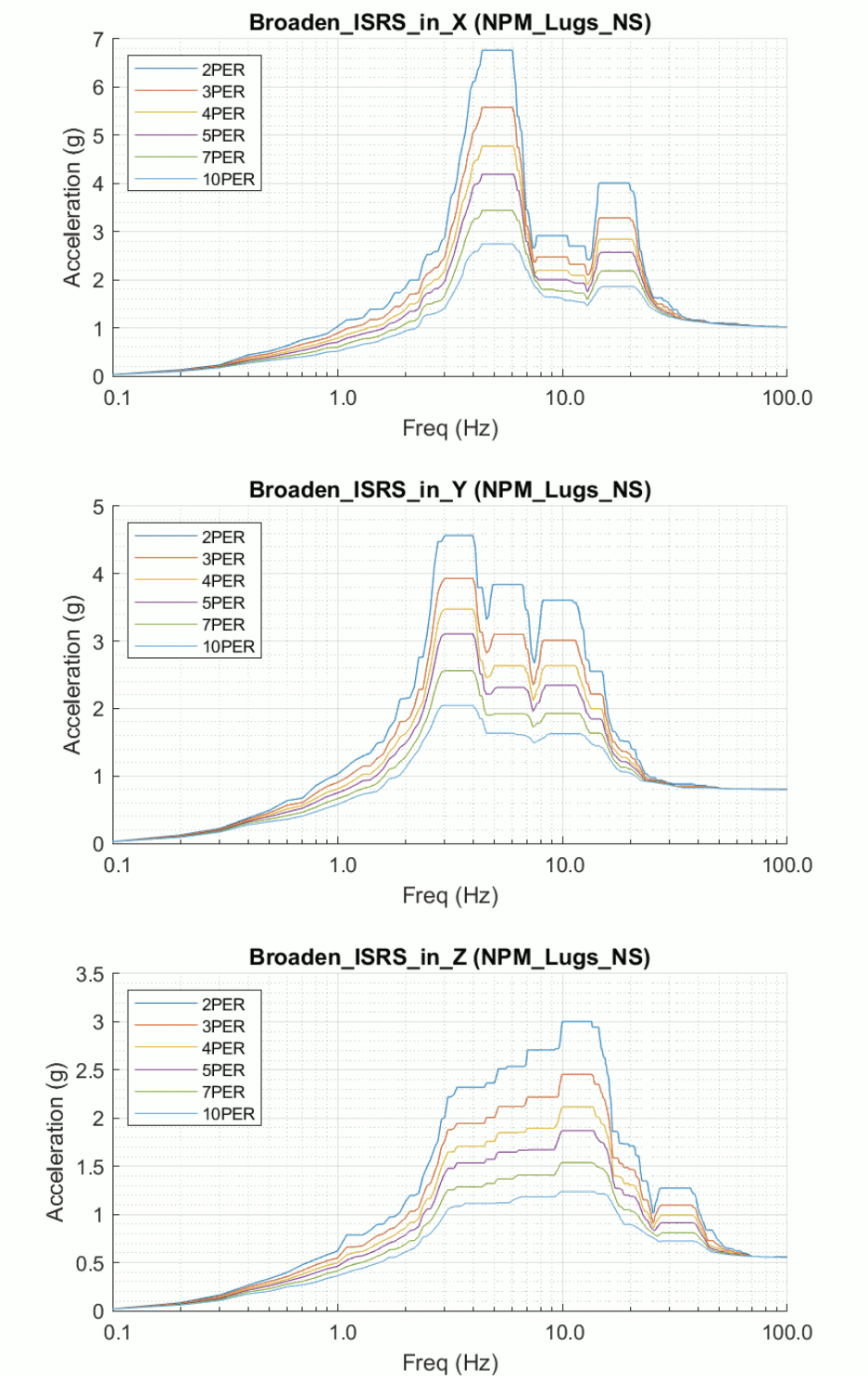


Figure 3.7.2-96: East-West, North-South, and Vertical In-Structure Response Spectra due to Certified Seismic Design Response Spectra - High Frequency for NuScale Power Module Lug Supports (East-West) at Elevation 95'

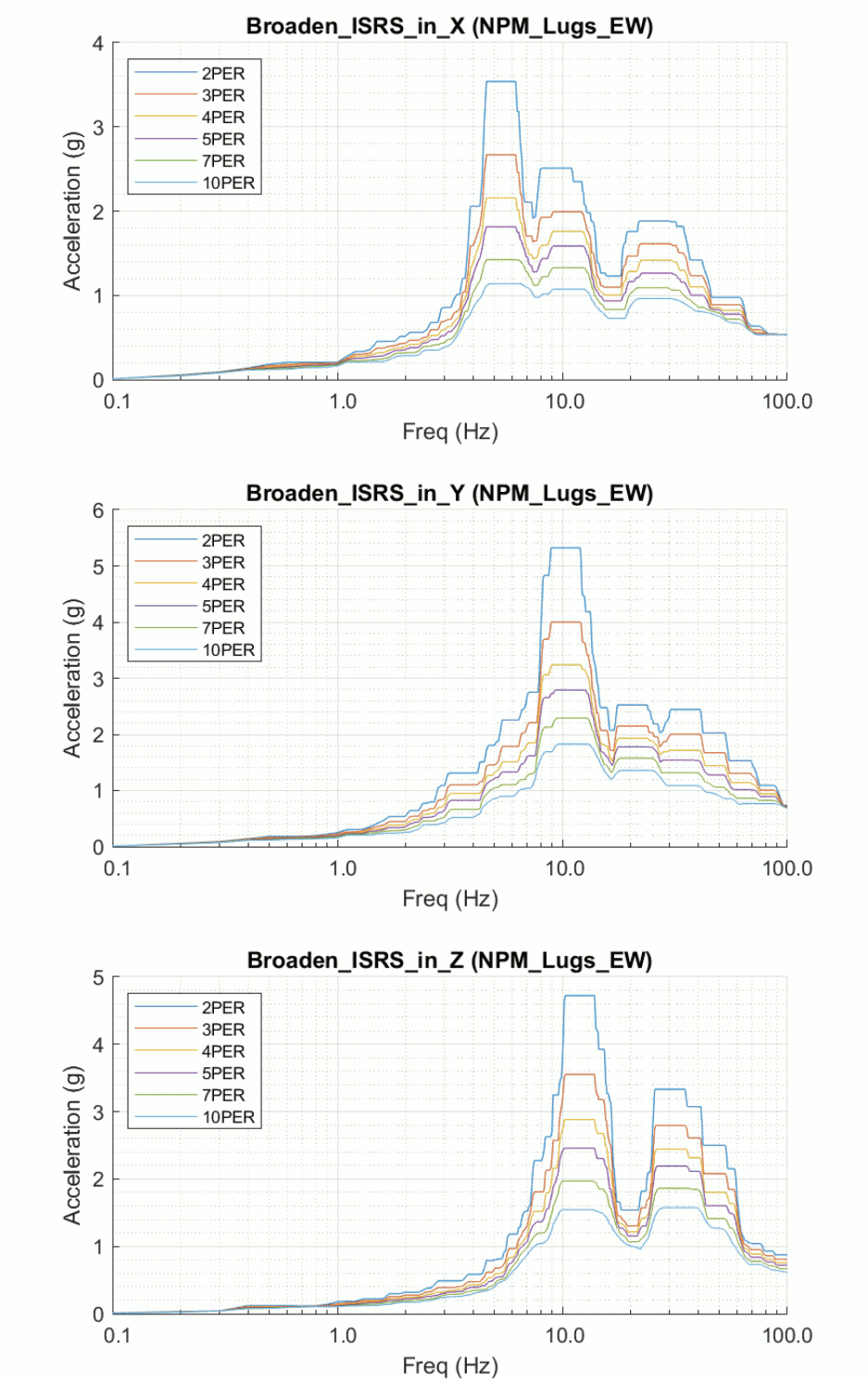


Figure 3.7.2-97: East-West, North-South, and Vertical In-Structure Response Spectra due to Certified Seismic Design Response Spectra - High Frequency for NuScale Power Module Lug Supports (North-South) at Elevation 95'

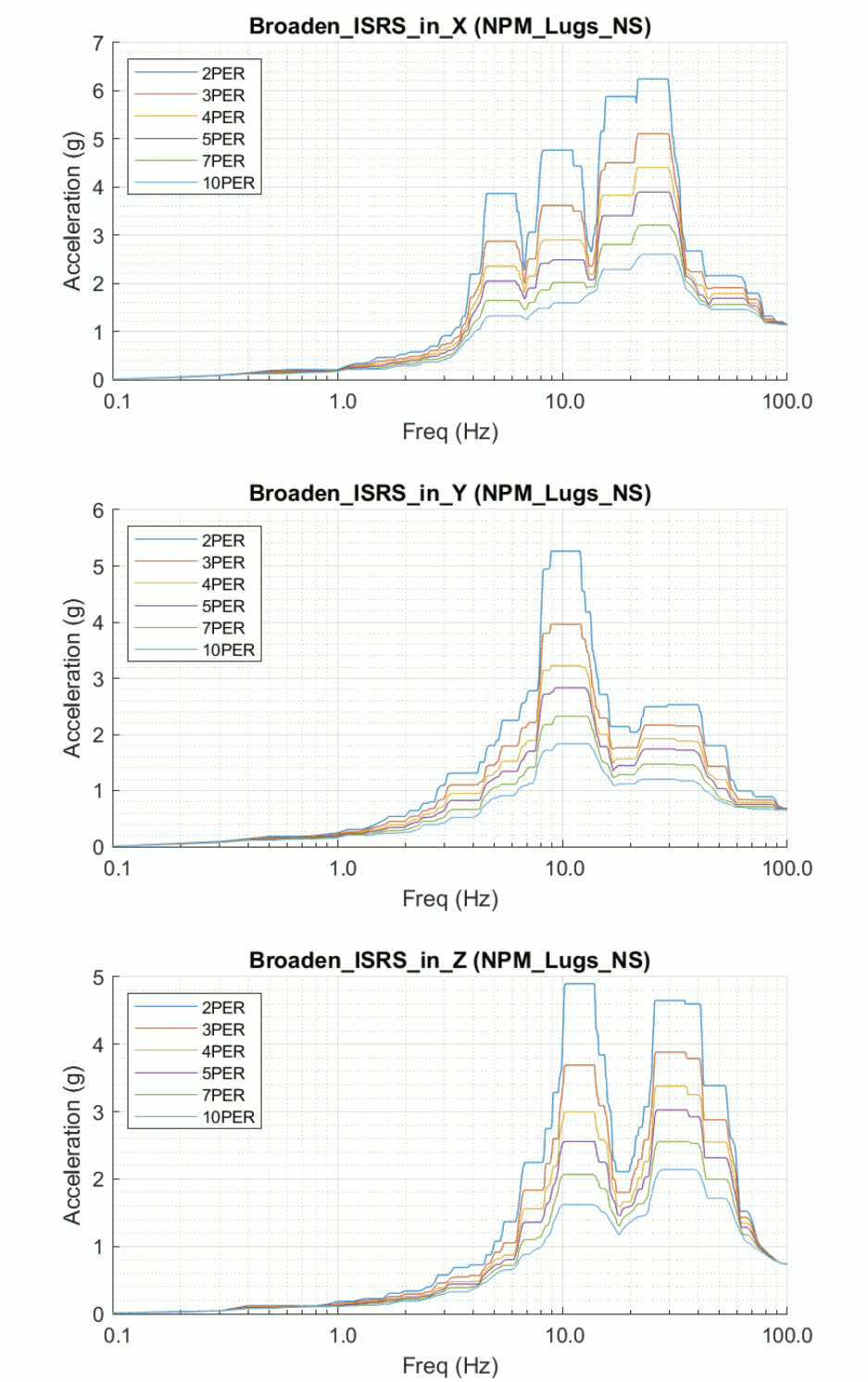
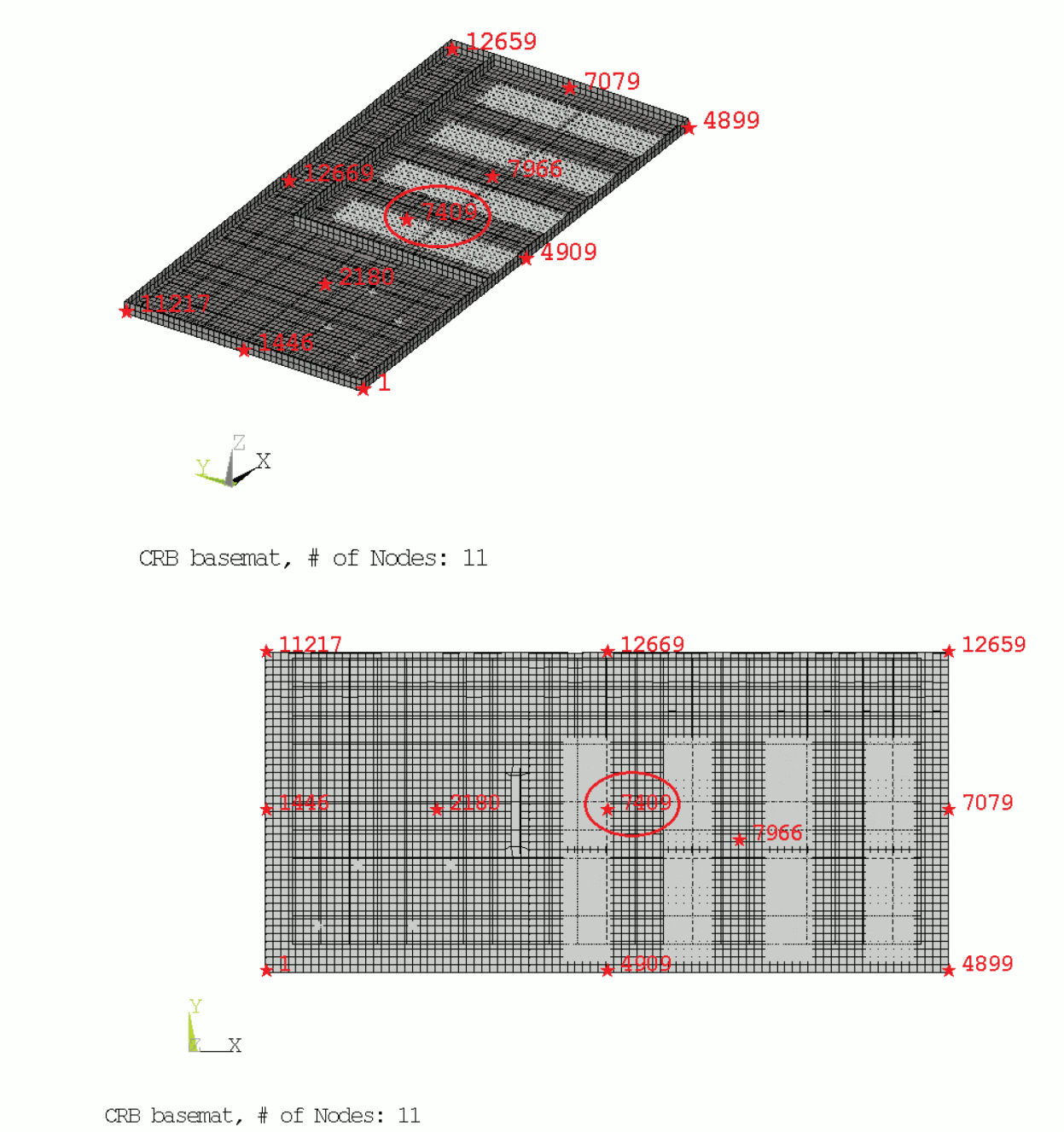


Figure 3.7.2-98: Selected Control Building Floor Nodes at Elevation 100' (Basemat) for In-Structure Response Spectra and Relative Displacement Calculation



Note: Reference node for relative displacement calculation is circled.

Figure 3.7.2-99: Selected Control Building Floor Nodes at Elevation 123' for In-Structure Response Spectra and Relative Displacement Calculation

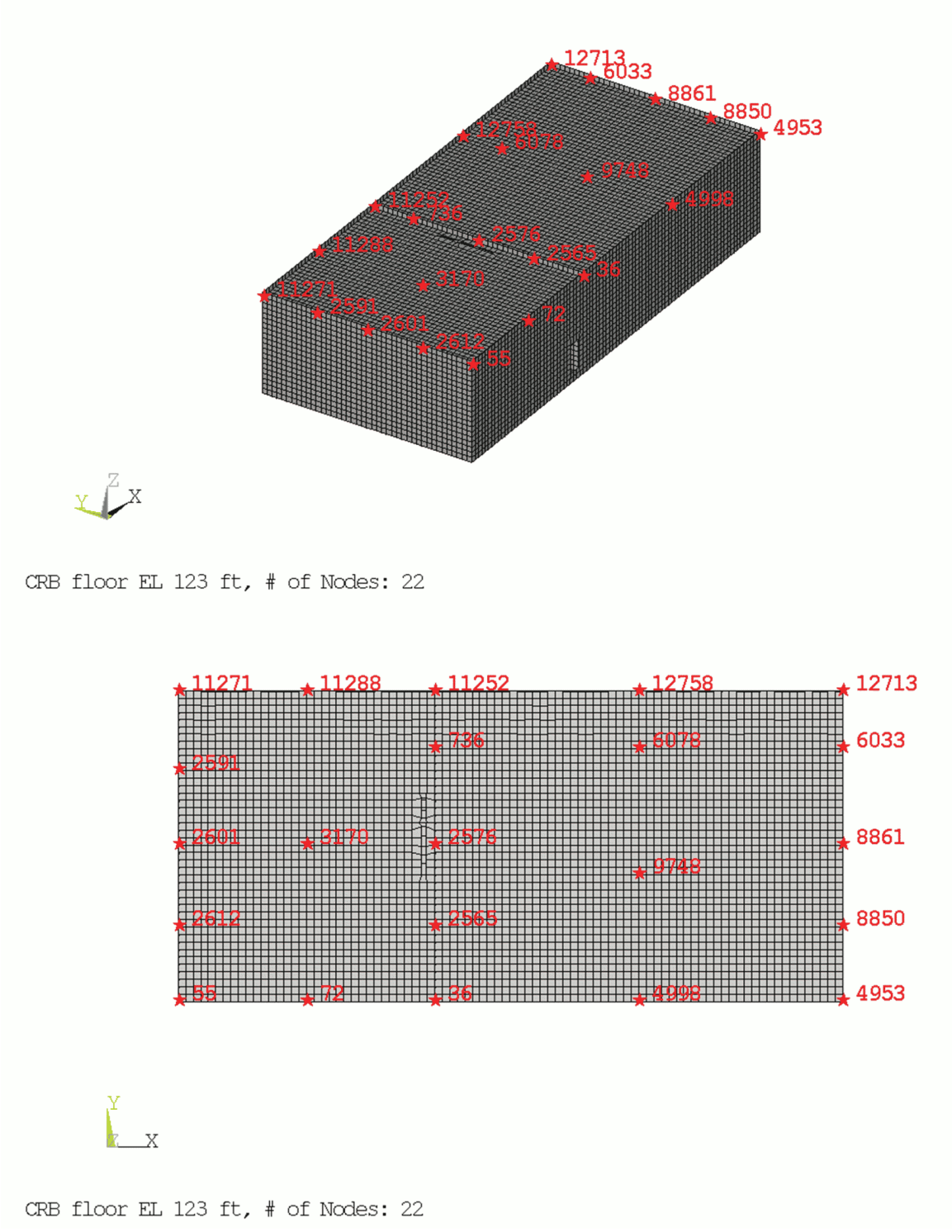


Figure 3.7.2-100: Selected Control Building Floor Nodes at Elevation 150' 3" (Roof) for In-Structure Response Spectra and Relative Displacement Calculation

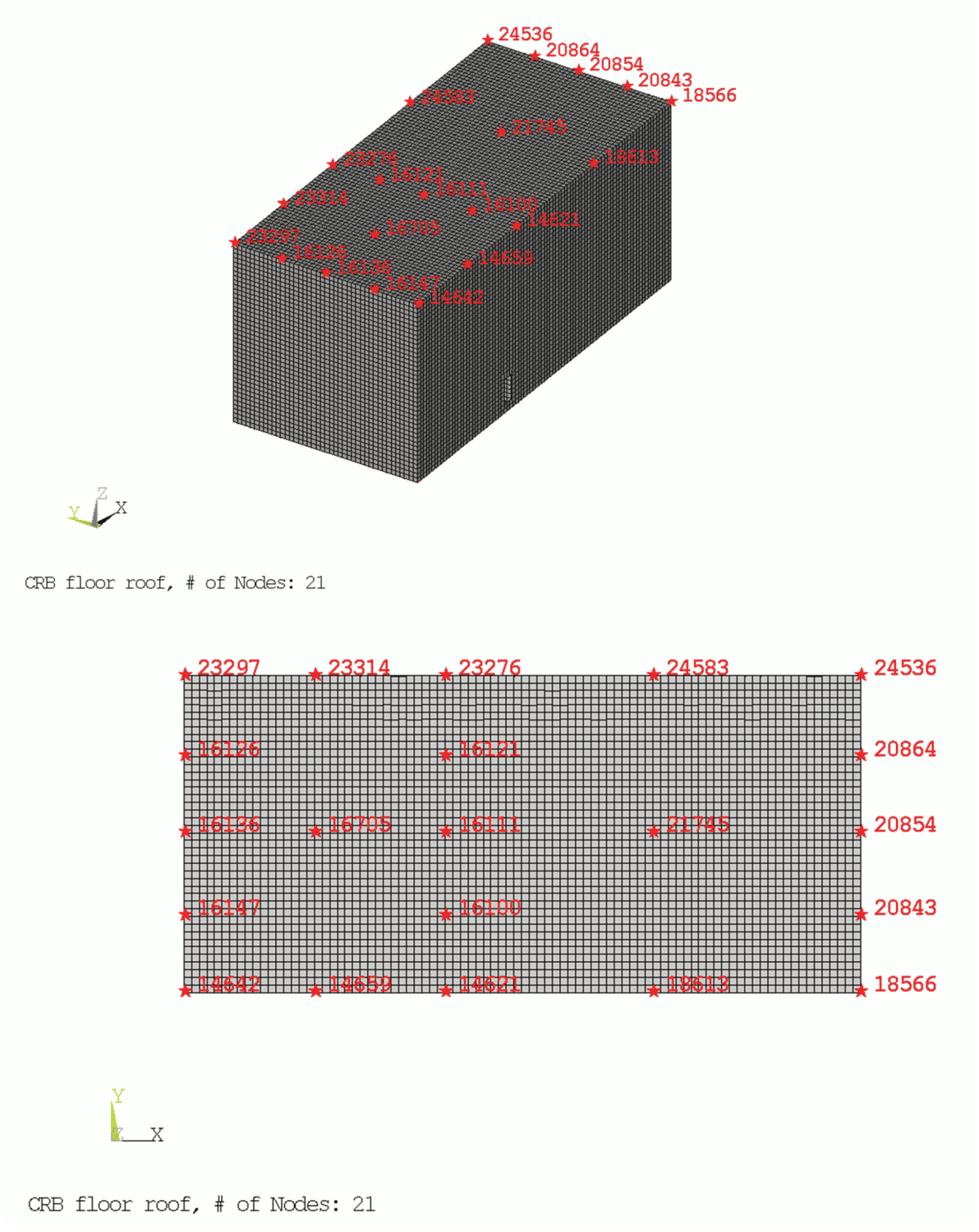


Figure 3.7.2-101: East-West, North-South, and Vertical In-Structure Response Spectra, Certified Seismic Design Response Spectra, and Control Building Basemat

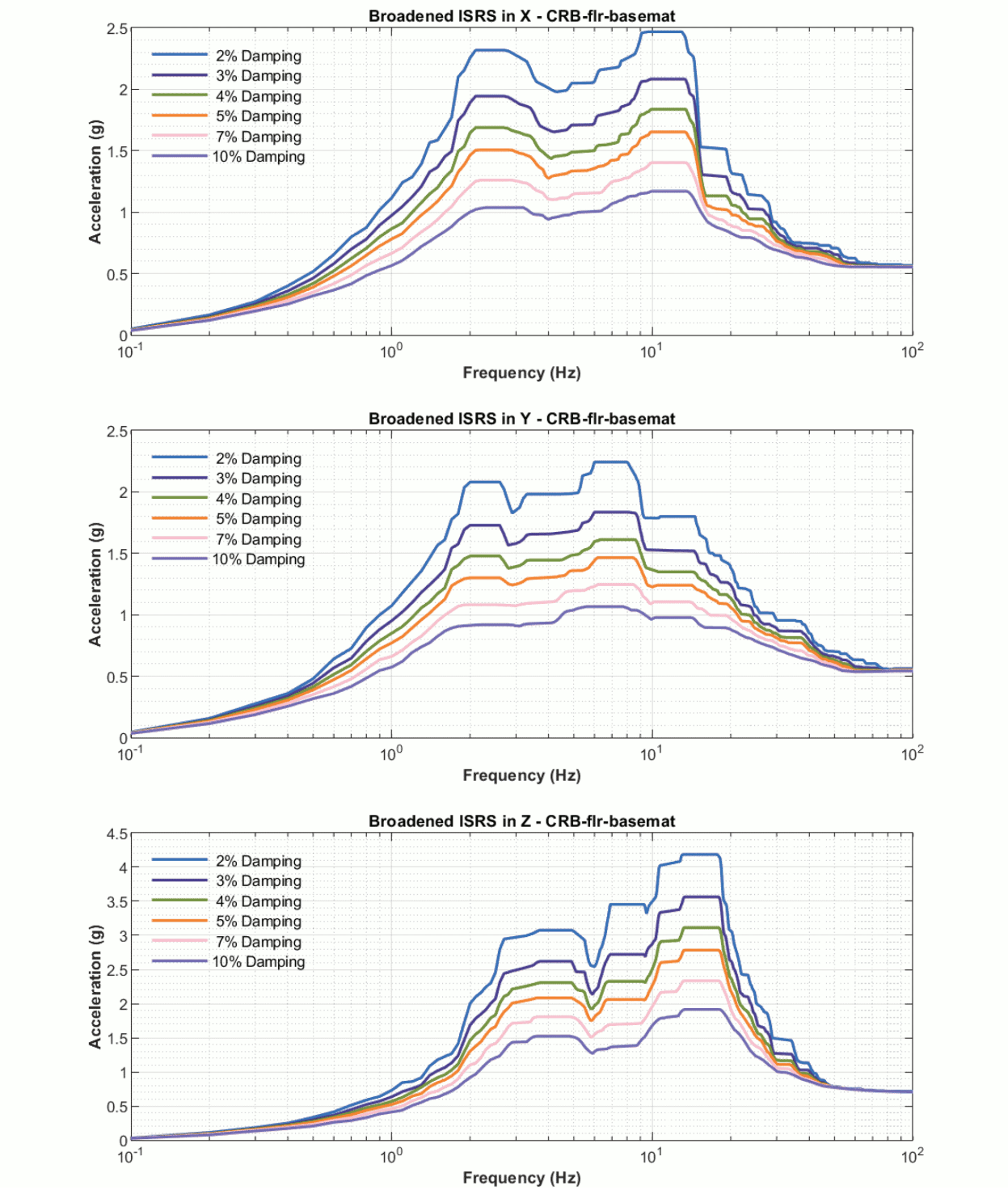


Figure 3.7.2-102: East-West, North-South, and Vertical In-Structure Response Spectra, Certified Seismic Design Response Spectra, and Control Building Floor at Elevation 123'

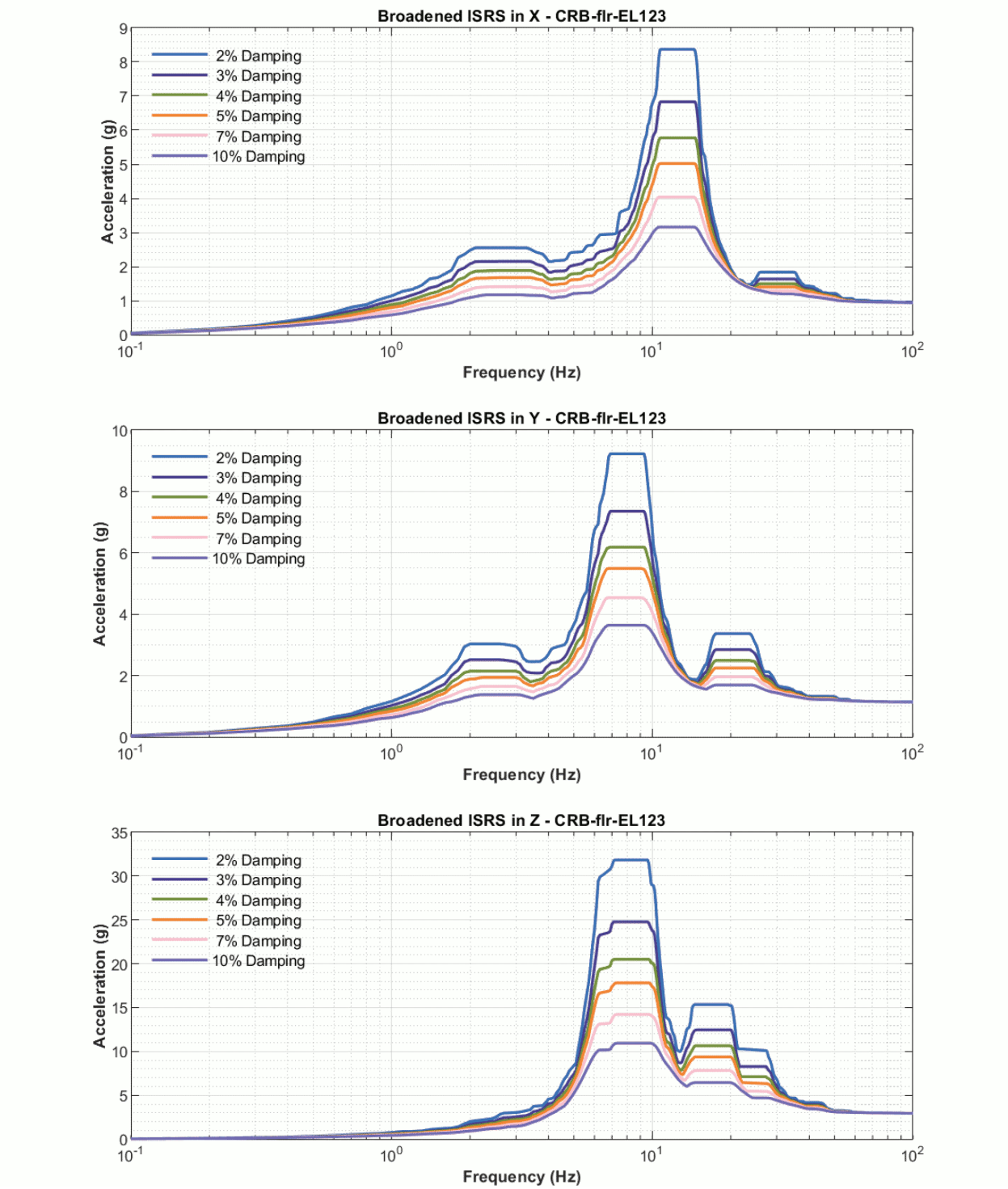


Figure 3.7.2-103: East-West, North-South, and Vertical In-Structure Response Spectra, Certified Seismic Design Response Spectra, and Control Building Roof

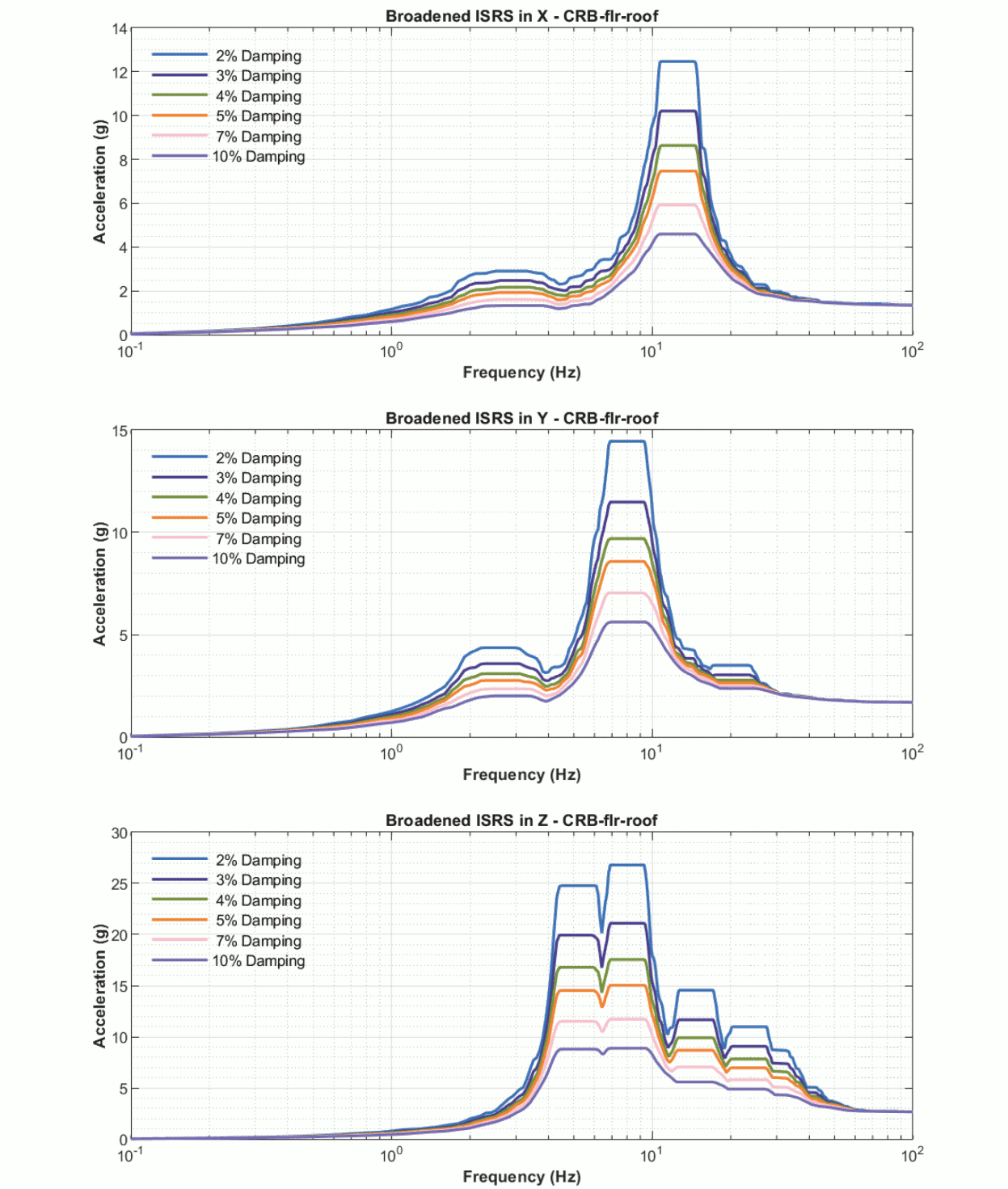


Figure 3.7.2-104: East-West, North-South, and Vertical In-Structure Response Spectra, Certified Seismic Design Response Spectra - High Frequency, Control Building Basemat

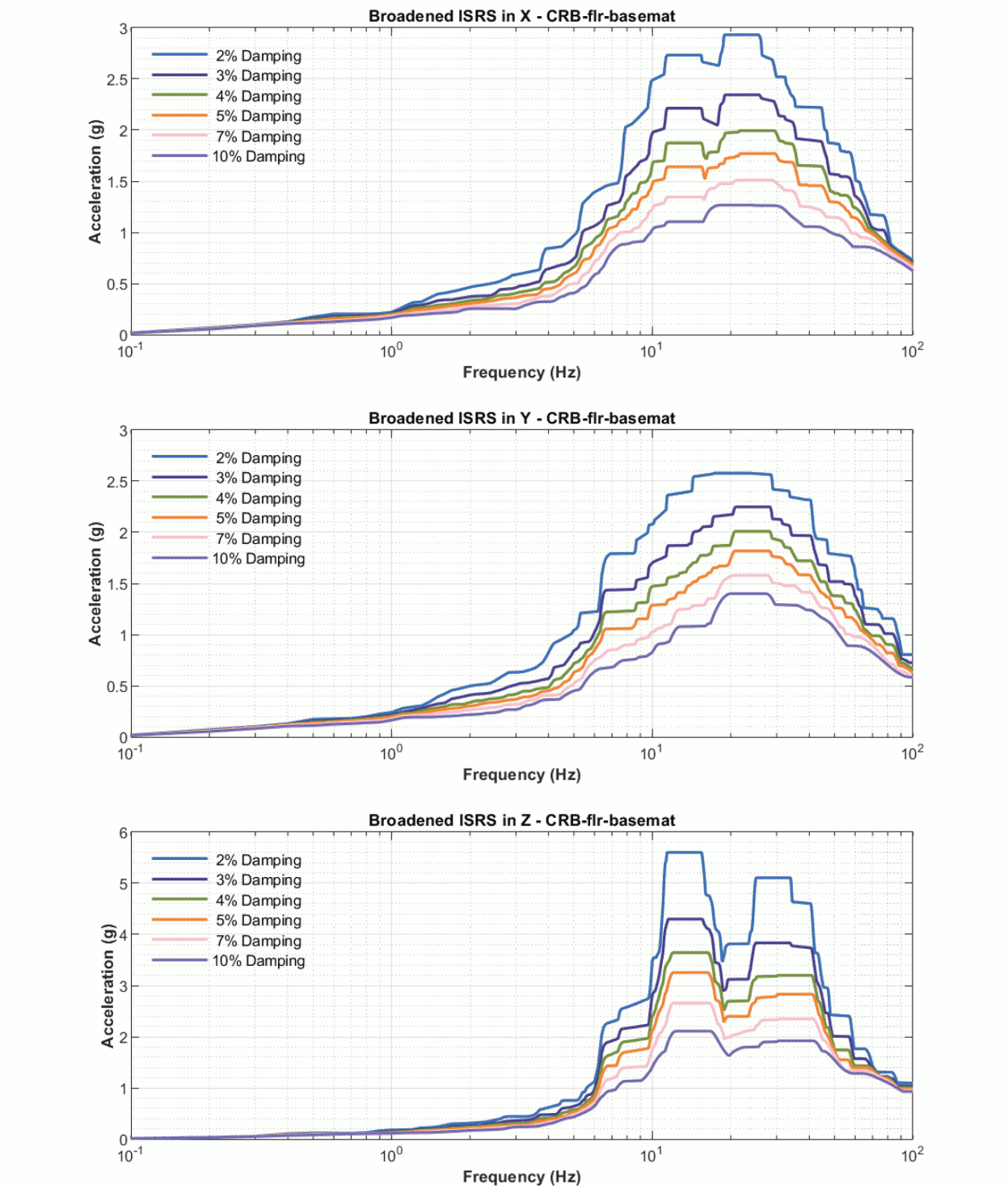


Figure 3.7.2-105: East-West, North-South, and Vertical In-Structure Response Spectra, Certified Seismic Design Response Spectra - High Frequency, Control Building Floor at Elevation 123'

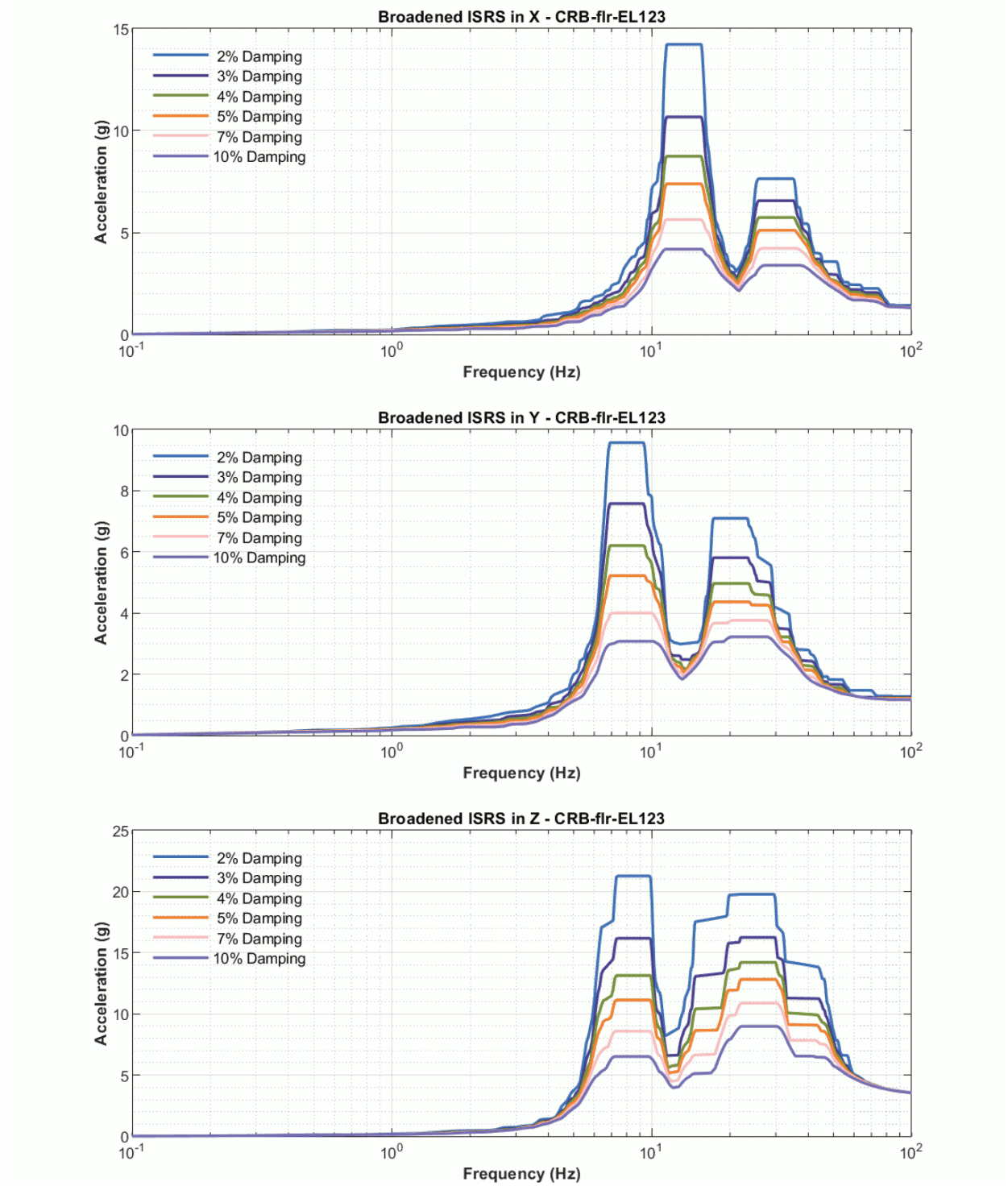


Figure 3.7.2-106: East-West, North-South, and Vertical In-Structure Response Spectra, Certified Seismic Design Response Spectra - High Frequency, Control Building Roof

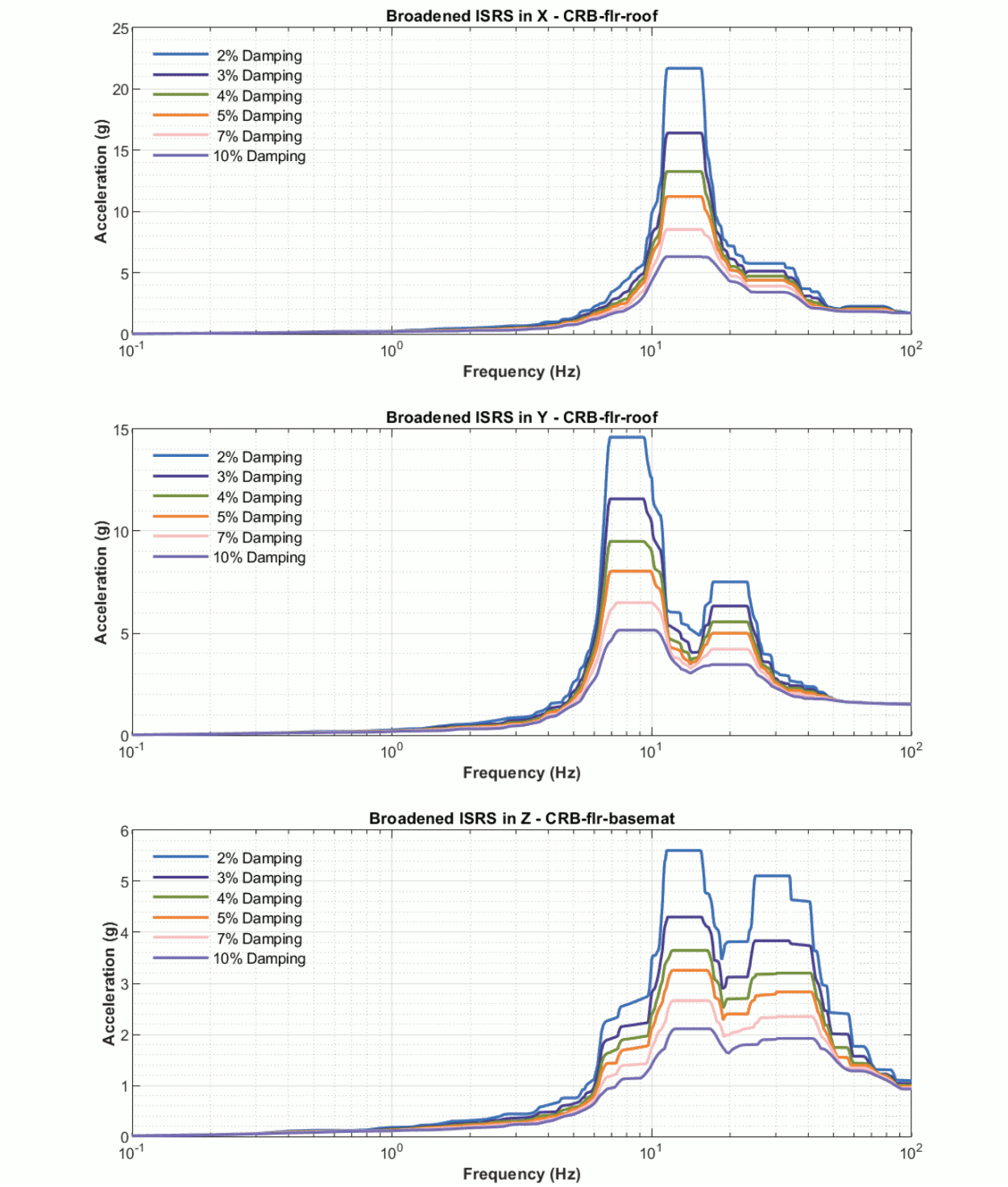


Figure 3.7.2-107: Control Building ANSYS Model (Major Mode in X-Direction, Frequency: 16.17 Hz)

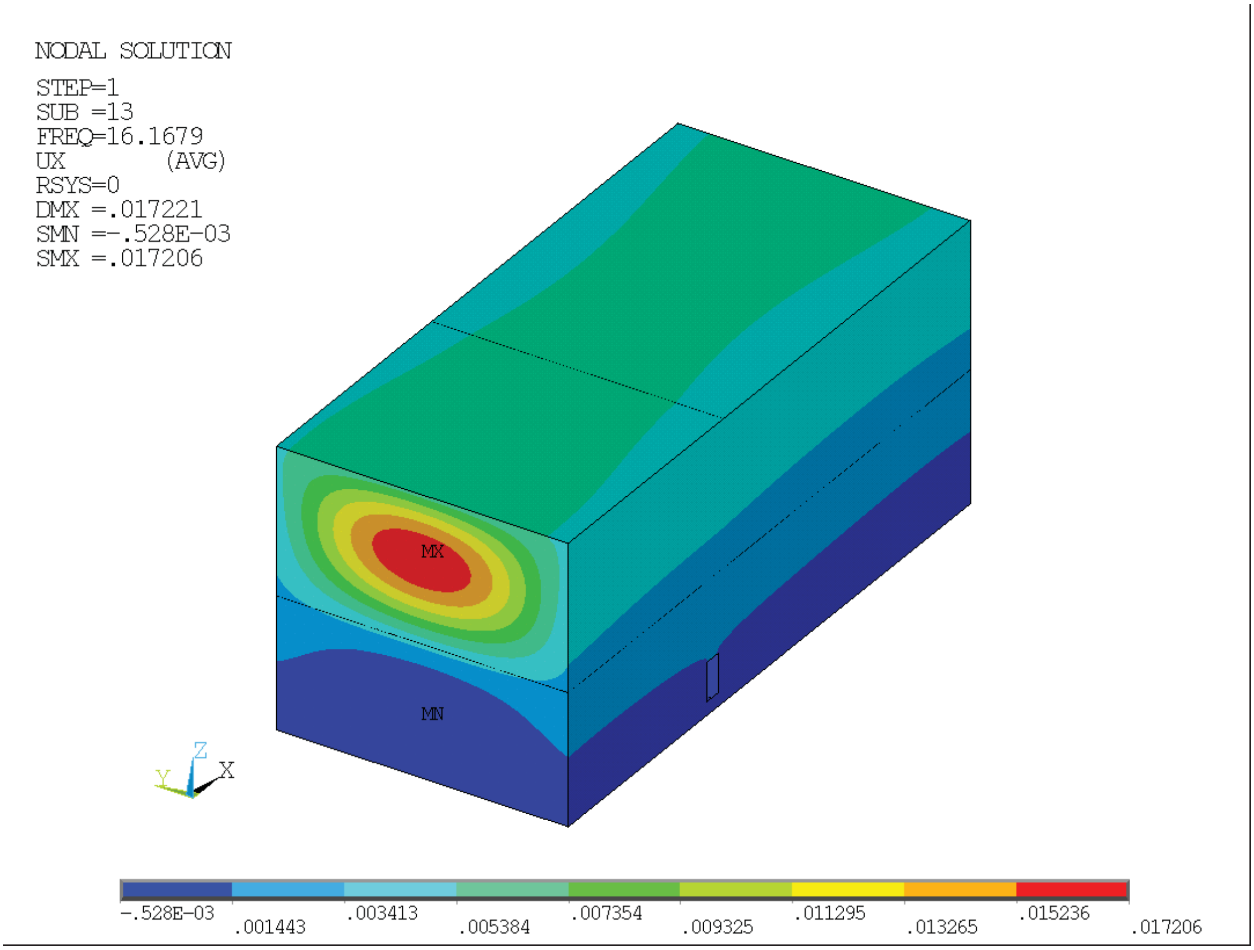


Figure 3.7.2-108: Control Building ANSYS Model (Major Mode in Y-Direction, Frequency: 11.34 Hz)

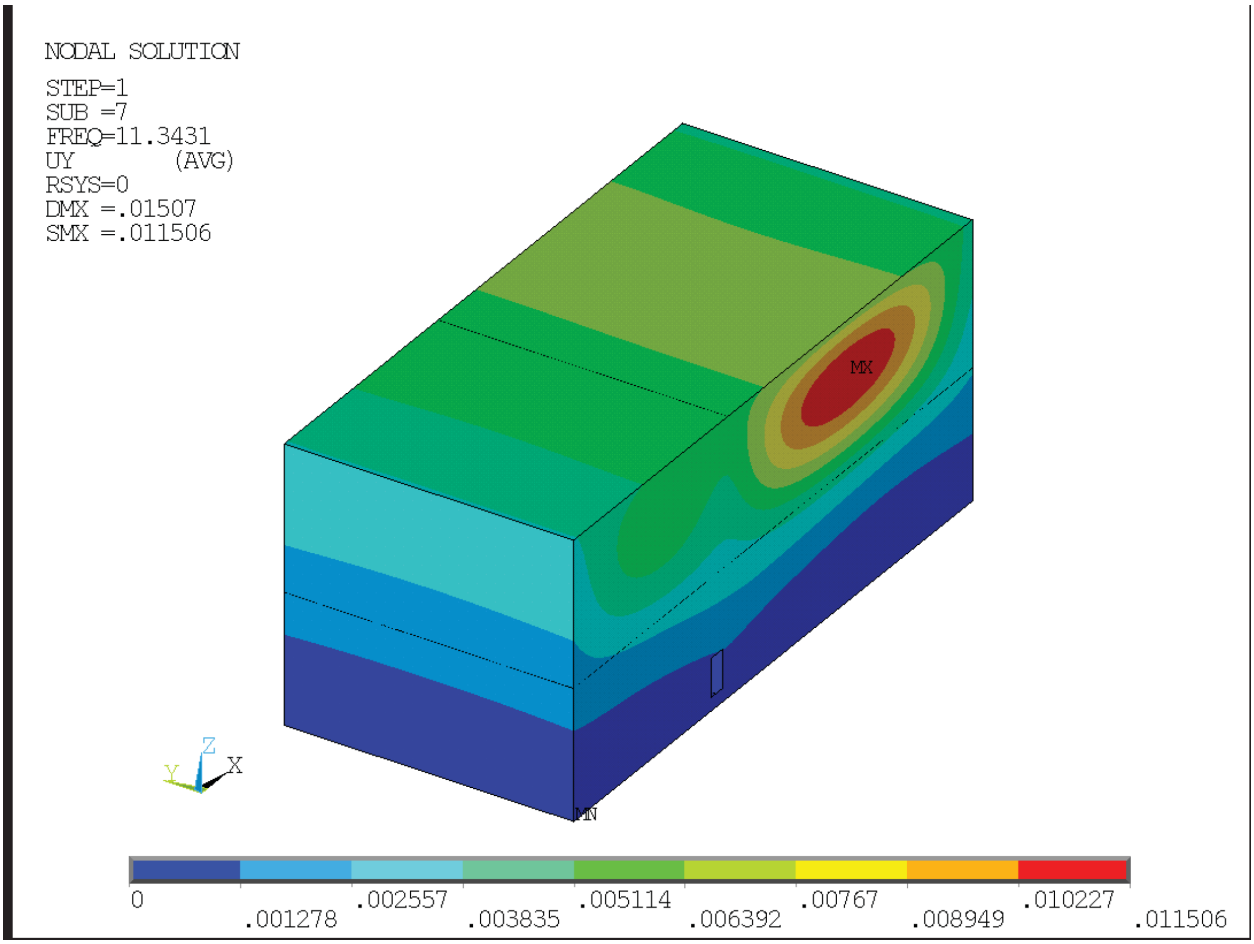
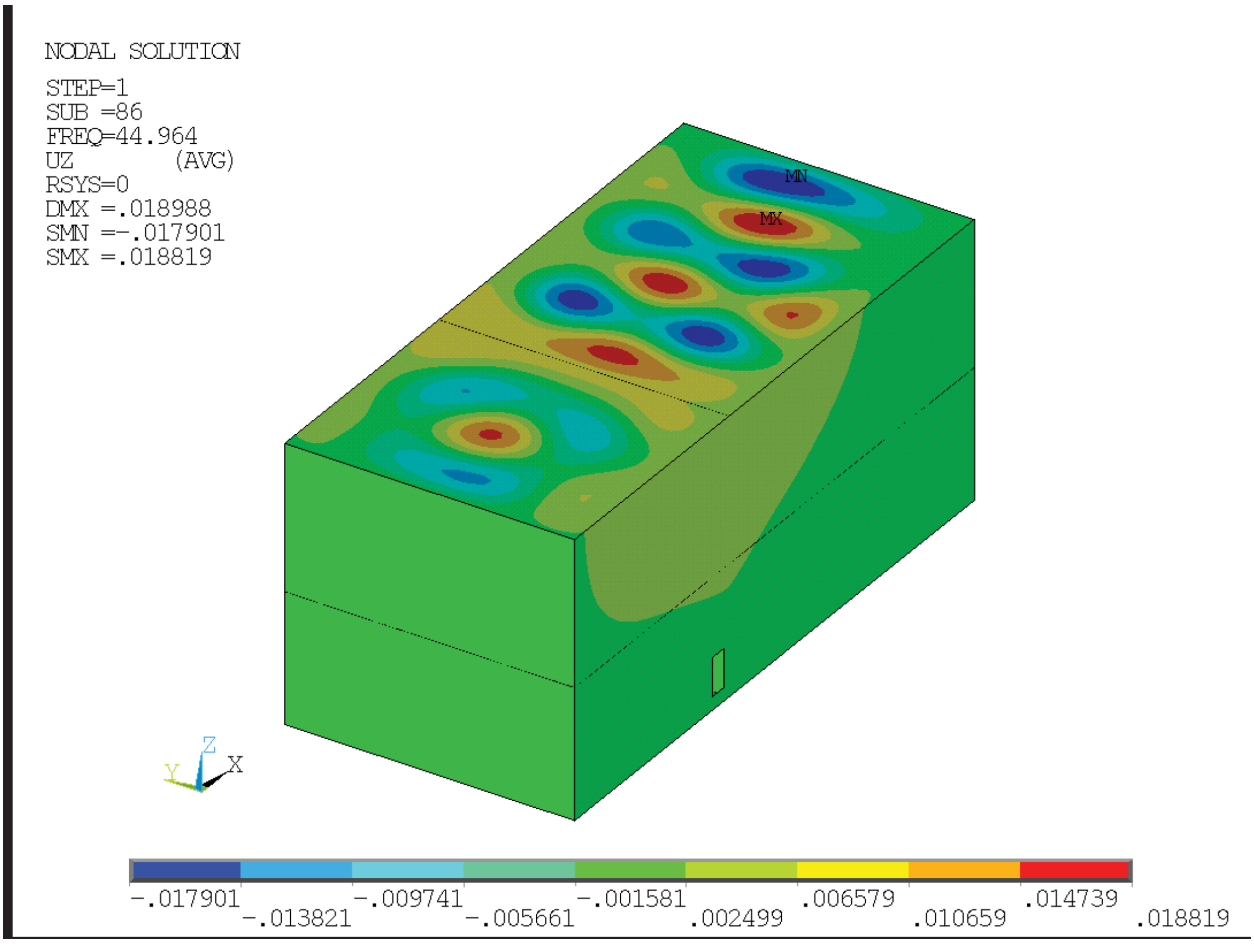


Figure 3.7.2-109: Control Building ANSYS Model (Major Mode in Z-Direction, Frequency: 44.96 Hz)



3.7.3 Seismic Subsystem Analysis

Seismic subsystems are structures, systems, and components (SSC) for which seismic forces are transmitted through the building structure as opposed to being imparted through soil. The following are subsystems in the NuScale design:

- structures such as miscellaneous steel platforms and framing that function primarily as a support for equipment
- equipment modules consisting of components, piping, supports, and structural frames
- equipment including vessels, tanks, heat exchangers, valves, instrumentation, and their supports
- distributive systems including piping and supports; electrical cable trays and supports; heating, ventilation, and air conditioning ductwork and supports; instrumentation tubing and supports; and conduits and supports.
- Each NuScale Power Module (NPM) is a subsystem. Appendix 3A summarizes the seismic analysis of the NPMs. As discussed in Section 3.7.2. the Reactor Building (RXB) seismic model includes NPMs as a combination of ANSYS shell elements and superelements.
- The fuel storage racks are a subsystem. The RXB seismic analysis includes fuel storage racks as weight only.
- The RXB crane (RBC) is a subsystem. Section 9.1.5 discusses the design of the RBC. The RXB seismic analysis includes the RBC as a beam, mass, and link model as discussed in Section 3.7.2.
- The bioshields are subsystems. The effects of bioshields are analyzed as part of the Reactor Building model as weights only. Section 3.7.3.3.1 discusses the bioshield design and analysis.

Section 3.12 describes piping systems and their supports for NuScale seismic subsystems.

3.7.3.1 Seismic Analysis Methods

Subsystems are generally evaluated using response spectrum analysis. Simple substructures may be evaluated using the equivalent static load method. These methods are described below. NuScale evaluated the NPMs using time histories as described in Appendix 3A.

3.7.3.1.1 Response Spectrum Analysis Method

In the response spectrum method of analysis, loads, stresses, and deflections are determined for each mode of the SSC being analyzed from the in-structure response spectra (ISRS). The ISRS are developed from the building analysis for each of the three directions (east-west, north-south, and vertical) as described in Section 3.7.2.

Modal responses are determined by accelerating each mode with the spectral acceleration corresponding to the frequency of that mode. The representative maximum response of interest for design is obtained by combining the corresponding maximum individual modal responses.

Equipment and components in some cases are supported at several points by either a single structure or two separate structures. The motions of the structures at each of the support points may be different. Section 3.7.3.9 addresses multiple-supported equipment and components with distinct inputs.

3.7.3.1.2 Equivalent Static Load Method

This methodology is available for the analysis of simple SSC. The equivalent static load method may evaluate

- single-point-of-attachment cantilever models with essentially uniform mass distribution, or other simple structures that can be represented as single degree-of-freedom systems.
- cantilevers with non-uniform mass distribution and other simple multiple-degree-of-freedom structures.

To obtain an equivalent static load for SSC, the method applies a factor of 1.5 to the peak spectral acceleration of the applicable ISRS. This force is applied at the center of mass of the SSC being evaluated. Results (loads, stresses, or deflections) are adjusted to account for relative motion among all points of support.

Because the equivalent static load method is a simplified approach, each analysis contains justification that use of a simplified model is realistic and results are conservative.

3.7.3.2 Determination of Number of Earthquake Cycles

The operating basis earthquake (OBE) is defined to be one-third of the safe shutdown earthquake (SSE). As such, the OBE is eliminated from explicit design or analysis in 10 CFR 50 Appendix S. Therefore, the OBE is not used for primary stress evaluations and is not included in load combinations for the design of standard plant SSC.

Fatigue analysis of SSC accounts for effects of the OBEs. The analysis assumes, during plant life, one SSE and five OBEs, with 10 maximum stress cycles per event. To meet this requirement, earthquake cycles in the fatigue analysis for the standard plant are normally composed of two SSE events with 10 maximum stress cycles each, for a total of 20 full cycles. This assumption is considered equivalent to the cyclic load basis of one SSE and five OBEs.

Alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles with an amplitude not less than one-third of the maximum SSE amplitude may be used when derived in accordance with IEEE 344-2013 Appendix D (Reference 3.7.3-7).

3.7.3.3 Procedures Used for Analytical Modeling

The analysis uses the following criteria for decoupling of the subsystem and the supporting system:

- if $R_m < 0.01$, decoupling can be done for any R_f
- if $0.01 \leq R_m \leq 0.1$, decoupling can be done if $0.8 \geq R_f \geq 1.25$
- if $R_m > 0.1$, a subsystem model should be included in the primary system model

where,

$$R_m = \frac{\text{total mass of supported subsystem}}{\text{total mass of supporting subsystem}}$$

$$R_f = \frac{\text{fundamental frequency of supported subsystem}}{\text{dominant frequency of support motion}}$$

The RXB dead plus live load weight is 337,000 kips. As such, a subsystem can be decoupled if the weight is less 3370 kips. The larger subsystems, the NPM and the RBC, have weights on the order of 2000 kips and 3000 kips and could be uncoupled. However, they are both coupled in the RXB model. The fuel storage racks are assumed to have a loaded weight of 400 kips each, and each bioshield is less than 230 kips. Therefore, these SSC are decoupled.

Distributed systems (cable trays, piping, heating, ventilation, and air conditioning) and individual components must not have significant weights that would challenge the $R_m < 0.01$ criterion.

3.7.3.3.1 Bioshields

The bioshields are nonsafety-related, not risk-significant, Seismic Category II components that provide an additional radiological barrier to reduce dose rates in the RXB and support personnel access. Bioshields are removed while an NPM is being detached and refueled. Means for lifting and handling the bioshields shall adhere to the heavy loads lifting program requirements described in Section 9.1.5.

The horizontal and vertical components are two separate pieces (not connected to each other). The horizontal bioshield is attached to bay walls using square-tube post with bolting brackets. When necessary, these bolts can be removed. This feature allows for the horizontal bioshield to be temporarily removed and placed on top of the adjacent bioshield for refueling and maintenance.

The vertical bioshield is supported at the top, where a W-shape support member is bolted to beam seats attached to the end of the pool bay walls. The vertical bioshield is also constrained to the end of the pool bay walls by

seismic restraints that resist horizontal motion. The vertical bioshield is removed and stored during maintenance and refueling.

The vertical bioshield bottom elevation is at 69 ft, below the minimum pool water level for normal operating conditions.

The structural design analysis of the horizontal bioshield is based on ACI 349-13 (Reference 3.7.3-3). The design of the vertical bioshield uses AISC N690-18 (Reference 3.7.3-5) and AISC 360 (Reference 3.7.3-4).

The bioshield is analyzed and designed to prevent failure under SSE conditions: the bioshield must not collapse or fail and strike or impair the integrity of safety-related or Seismic Category I SSC under it.

Each bioshield consists of

- a horizontal component that is 24 inches thick and composed of reinforced concrete and encased by a stainless steel liner.
- a vertical component composed of a square stainless steel tube framing system, radiation paneling consisting of borated high-density polyethylene (HDPE), and a W-shape support member.

The bioshield horizontal component sits on NPM bay walls at elevation 123 ft, and the vertical component that covers the front of the NPM bay extends to approximately elevation 70 ft. The NPM bays are 23 ft – 3 in. deep, and 20 ft –6 in. wide (the end bays to the west are 2 ft wider to provide space to accommodate bioshield attachment to RXB).

The bioshield is designed to limit radiation stream vertically and horizontally from the operating bay including provisions and design requirements in DSRS 3.7.3. Radiation shielding panels are designed to ensure occupational radiation exposure is as low as reasonably achievable as described in Chapter 12. Radiation shielding panels are composed of borated HDPE panels with 5 percent boron content. The HDPE is encased in stainless steel plate and angle assemblies. The clamped assembly restricts flame and adequate oxygen from causing combustion of HDPE panels. Off-gassing of the HDPE panels during operation is allowed as the clamped assemblies are not hermetically sealed. Encasement of HDPE eliminates it as a fire load. The vertical assembly is vented for heat removal during normal operation as well as for heat and pressure mitigation in the event of a high-energy line break and slow leak, high temperature event above the NPM. The vents are staggered on the vertical portion of the bioshield, providing enough ventilation area to allow continuous ventilation of the operating bay.

Figure 3.7.3-1 shows the plan view of the horizontal bioshields. Figure 3.7.3-2 shows the elevation view of the vertical bioshield attached to the RXB. Figure 3.7.3-3 shows an isometric view of vertical bioshield and Figure 3.7.3-4 shows a section view of the vertical bioshield support (wide flange).

COL Item 3.7-10: An applicant that references the NuScale Power Plant US460 standard design will determine the means and methods of lifting the bioshield. An applicant will demonstrate that bioshield components and connections can withstand the bioshield loads and appropriate load factors.

Reinforced Concrete Properties and Slab Capacity

The concrete and reinforcement design values are from ACI 349 (Reference 3.7.3-3). The minimum concrete cover for cast-in-place members is based on ACI 349.

The horizontal slab capacity is determined based on ACI 318 (Reference 3.7.3-1) and the vertical tube frame is based on ANSI/AISC 360 (Reference 3.7.3-4). Individual equations in ACI 318 are used for determining out of plane moment and shear capacity.

Structural Steel Material Properties

The vertical assembly of the bioshield is constructed from stainless steel tube members in the horizontal and vertical directions.

In-Structure Response Spectra

Descriptions of in-structure response spectra for multiple locations in the RXB are in Section 3.7.2. Locations at the top of the bay walls were selected to develop ISRS for the design of the bioshields. The enveloped ISRS with 4 percent damping are shown in Figure 3.7.3-5 and Figure 3.7.3-6. These figures envelop certified seismic design response spectra (CSDRS) curves and include effects of sensitivity analysis cases such as considering the effects of soil separation and different soil profiles. The figures also show the curves with the valleys removed that were used for the response spectrum analysis.

3.7.3.3.1.1 Evaluation

The vertical framing analysis was completed with a linear finite element model consisting of BEAM188 element types and masses.

The vertical bioshield is supported by stainless steel structural members that reconnect to the bay walls. Connections are designed to allow for the vertical bioshield to be lifted and removed.

The horizontal bioshield is analyzed through two separate linear models. In the first analysis, the loading is applied in the direction of gravity. In this model, boundary conditions were investigated and defined to conservatively capture stress in the slab with vertical restraints placed at the bottom of the slab. The second analysis captures load reversal effects with the slab restrained at its mid-plane like a simply supported beam. Based on the design, the gravity direction forces are carried by the bay walls in bearing. In the event of load reversal, four bolted connections

restrain the slab from moving vertically. Both models are characterized with SOLSH190 elements.

3.7.3.4 Basis for Selection of Frequencies

To avoid resonance, components are designed (or specified) so that fundamental frequencies of the component are less than one half or more than twice the dominant frequencies of the support structure. However, equipment is tested or analyzed to demonstrate that it is adequate for design loads considering the fundamental frequencies of the equipment and the support structure.

3.7.3.5 Analysis Procedure for Damping

Table 3.7.1-6 shows generic damping values for seismic analysis of SSC. Component modal damping of piping systems is described in Section 3.12.3.

3.7.3.6 Three Components of Earthquake Motion

Seismic demand is obtained for the three orthogonal (two horizontal and one vertical) components of earthquake motion from the ISRS. Each component of the earthquake motion is considered in the seismic analysis of subsystems. When the total response of the substructure is needed, it is normally obtained by combining the three directional responses using the square root of the sum of the squares (SRSS) method; for example, the SRSS method was used for designing the bioshield framing and the RBC connections. The 100-40-40 rule is an acceptable alternative to the SRSS method; for example, the 100-40-40 rule was used for designing the bioshield connections.

3.7.3.7 Combination of Modal Responses

For the response spectrum method of analysis, maximum responses such as accelerations, shears, and moments in each mode are calculated regardless of time. If mode frequencies are well-separated, the SRSS method is used; however, where structural frequencies are not well-separated, modes are combined in accordance with Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

3.7.3.8 Interaction of Non-Seismic Category I Subsystems with Seismic Category I Structures, Systems, and Components

When non-Seismic Category I SSC (or portions thereof) could adversely affect Seismic Category I SSC, they are categorized as Seismic Category II and analyzed using one of the methodologies described in Section 3.7.3.1.

For non-Seismic Category I subsystems attached to Seismic Category I SSCs, the dynamic effects of the non-Seismic Category I subsystems are included in the modeling of the Seismic Category I SSC. The attached non-Seismic Category I subsystems, up to the first anchor beyond the interface, are designed so that the CSDRS does not cause a failure of the Seismic Category I SSC.

3.7.3.9 Multiple-Supported Equipment and Components with Distinct Input

Equipment and components, in some cases, are supported at several points by either a single structure or two separate structures. Motions of the primary structure or structures at each of the support points may be different.

A suitable approach for analyzing equipment supported at two or more locations is to define a uniform response spectrum that envelopes individual response spectra at support locations. The uniform response spectrum is applied at all locations to calculate the maximum inertial responses of the equipment. This application is referred to as the uniform support motion method. In addition, the analysis considers relative displacements at support points. The maximum relative support displacements are obtained from the building structural response calculations. The support displacements are imposed on the supported equipment in the most unfavorable combination. The responses, due to the inertia effect and relative displacements, are combined by the absolute sum method.

The uniform support motion method can result in overestimating seismic responses. In the case of multiply-supported equipment in a single structure or spanning among structures, an alternate method that can be used is the independent support motion approach. For independent support motion analysis, the guidance and criteria in NUREG-1061, Volume 4, "Evaluation of Other Loads and Load Combinations," (Reference 3.7.3-8) is used.

In the independent support motion grouping procedure, structural support points that are attached to a rigid floor or structure (so that the same translationary motion, without rotation, is experienced) are considered as one group of supports. After the individual group responses are determined for each input direction, they are combined by the absolute sum method. The modal and directional responses are combined using the methods described in Section 3.7.3.6 and Section 3.7.3.7.

3.7.3.10 Use of Equivalent Vertical Static Factors

Equivalent vertical static factors are not used in the design of the Seismic Category I and Seismic Category II structures. Vertical seismic loads are generated from the soil-structure interaction analysis.

3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects due to the presence of significant eccentric masses connected to a subsystem are included in the subsystem analysis. For rigid components (natural frequencies greater than 50 Hz), the lumped mass is modeled at the center of gravity of the component with a rigid link to the appropriate point in the subsystem. For flexible components, the subsystem model is expanded to include an appropriate model of the component.

Torsional effects of eccentric masses affecting the piping design are included in the analysis described in Section 3.12.4.

3.7.3.12 Buried Seismic Category I Piping, Conduits, and Tunnels

There is a safety-related underground reinforced concrete duct bank that contains conduits connecting the RXB and CRB. Two finite element models were used for the evaluation of the duct bank. The first model has an "L" shape that traverses eastward from the CRB and then southward to a manhole. The second model has a "T" shape, with the arms traversing southward from the manhole to the RXB.

The reinforced concrete design of the duct bank and applicable load combinations are based on ACI 349-13 (Reference 3.7.3-3). The provisions of ACI 349 are only applicable to building structures and components and the duct bank does not fall under this description; however, following ACI 349 for the design of the duct bank and reinforcement is a conservative approach as the duct bank is considered a safety-related structure. Additionally, the duct bank is classified as Seismic Category I (SC-I).

The duct bank seismic analysis is considered as an SSE load and is calculated based on Section 7.1.1 of ASCE 4-16 (Reference 3.7.3-9) for buried conduit. This approach calculates the maximum axial strain in the soil due to seismic ground motion considering no slip between the soil and concrete duct bank, which is a conservative assumption. Strain calculation is based on wave velocity and maximum peak ground velocity. The calculated strain is applicable to buried structures with straight runs, bends, or intersections and can be applied to all global directions per Section 7.1.1 of ASCE 4-16. Once the strain is calculated, it is input into the finite element model to determine stresses in the duct bank.

The stresses due to the SSE load are combined with stresses due to soil overburden, live load (the larger of laydown loading and truck loading), and snow loading for evaluation, Stresses due to anchor point movement are not included because of the presence of isolation joints at the building and manhole interfaces.

3.7.3.13 Methods for Seismic Analysis of Category I Concrete Dams

The design neither includes nor requires the presence of a dam.

3.7.3.14 Methods for Seismic Analysis of Aboveground Tanks

The design does not include Seismic Category I aboveground tanks.

3.7.3.15 References

3.7.3-1 American Concrete Institute, “Building Code Requirements for Structural Concrete and Commentary,” ACI 318 – 08, Farmington Hills, MI, 2014.

3.7.3-2 American Concrete Institute, ACI 349-13, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” Farmington Hills, MI, 2014.

3.7.3-3 American Concrete Institute, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," ACI 349-13, Farmington Hills, MI.

3.7.3-4 American National Standards Institute/American Institute of Steel Construction, "Specification for Structural Steel Buildings," ANSI/AISC 360-16, Chicago, IL.

3.7.3-5 American Institute of Steel Construction, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” AISC N690 – 18, Chicago, IL, 2018.

3.7.3-6 ANSYS (Version 19.2) [Computer Program]. (2019). Ansys Electronics Solutions.

3.7.3-7 Institute of Electrical and Electronics Engineers, “IEEE Recommended Practice for Seismic Qualification of Equipment for Nuclear Power Generating Stations," IEEE Standard 344-2013, Piscataway, NJ.

3.7.3-8 U.S. Nuclear Regulatory Commission, “Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Other Loads and Load Combinations,” NUREG-1061, Volume 4, December 1984.

3.7.3-9 American Society of Civil Engineers, “Seismic Analysis of Safety-Related Nuclear Structures,” ASCE 4-16, Reston, VA.

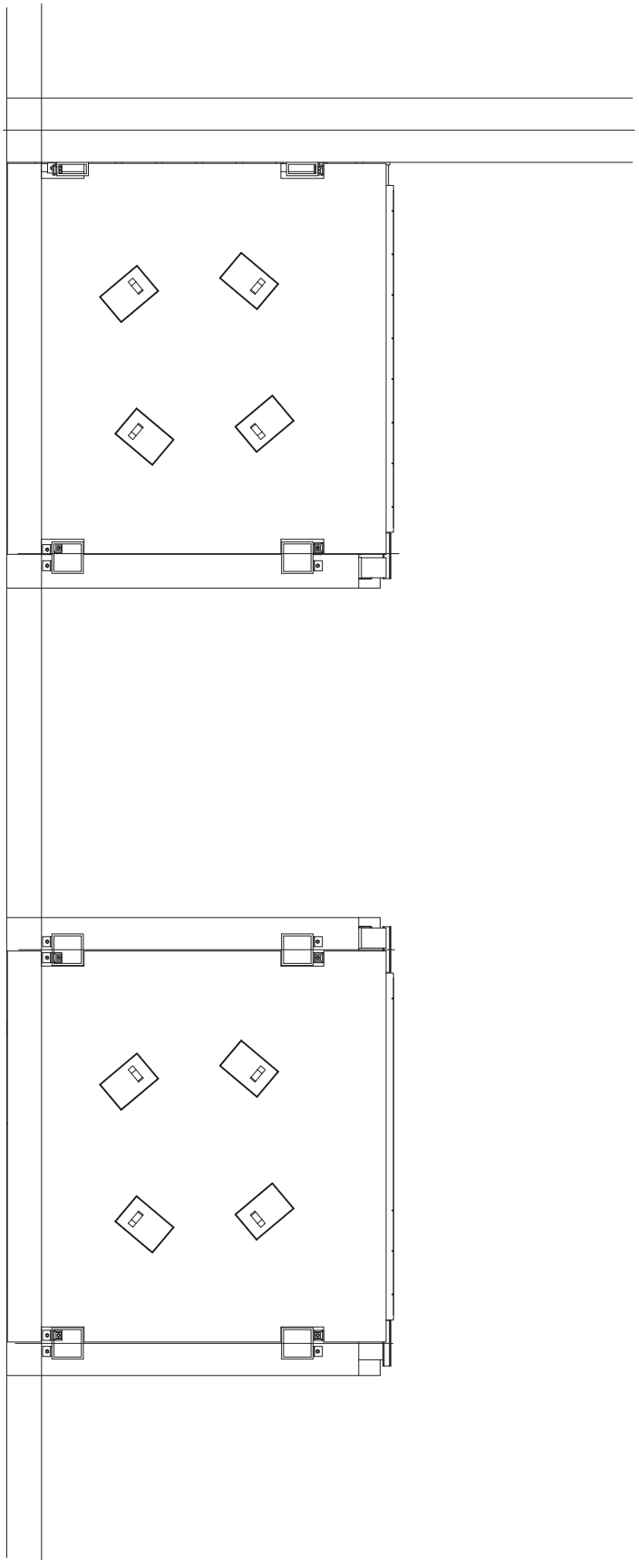


Figure 3.7.3-1: Plan View of Horizontal Bioshield Attached to Reactor Building

Figure 3.7.3-2: Elevation View of Vertical Bioshield Attached to Reactor Building

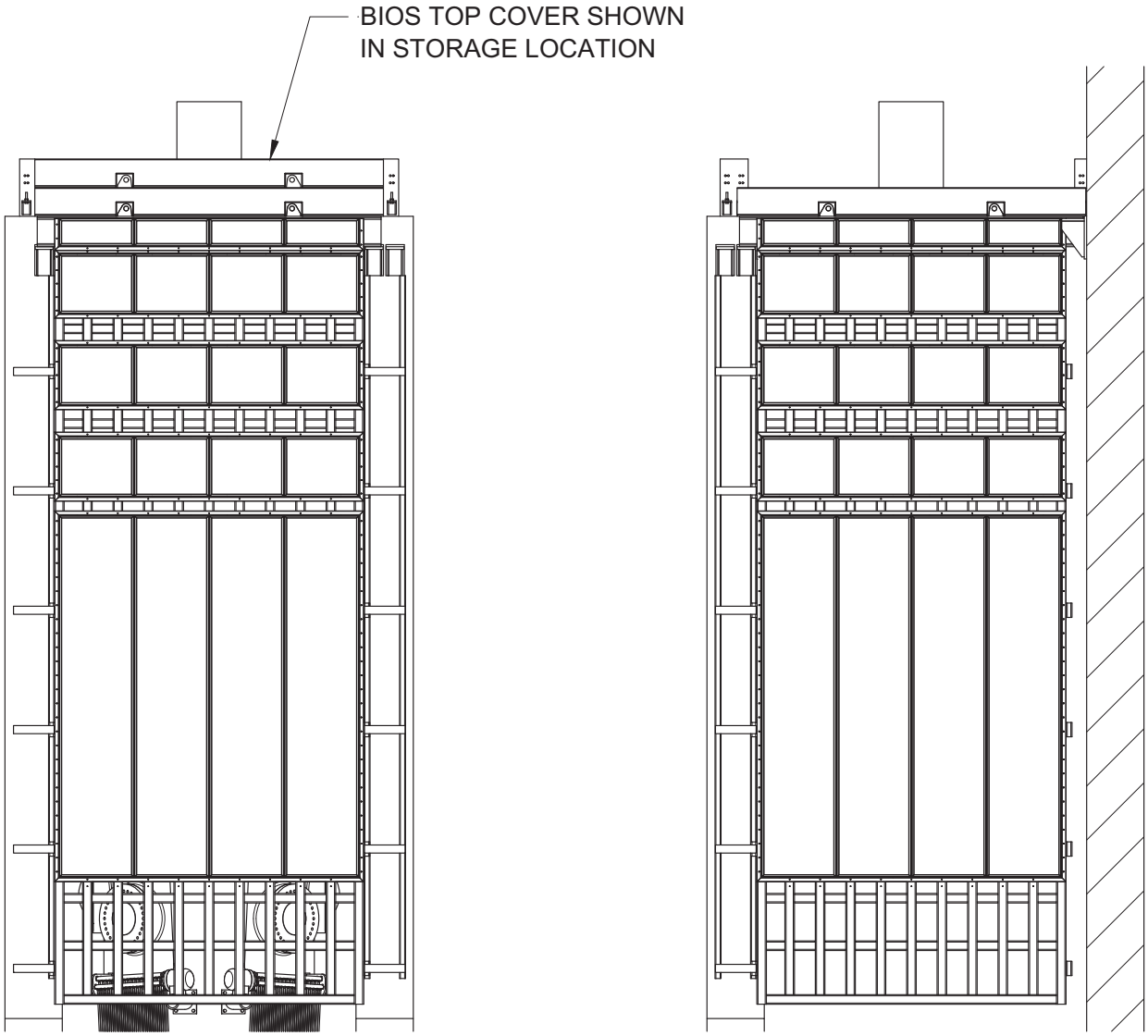


Figure 3.7.3-3: Isometric View of Vertical Bioshield

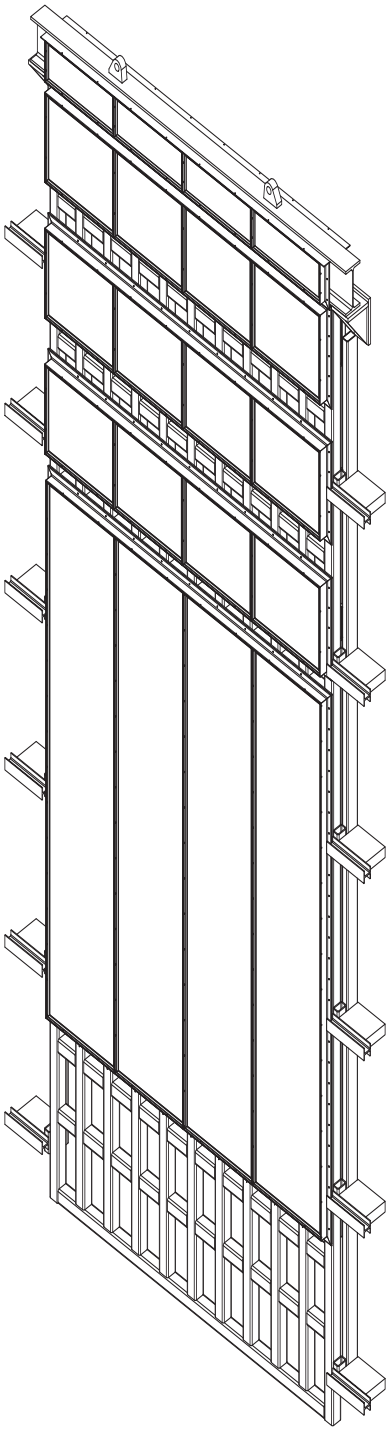


Figure 3.7.3-4: Elevation View of Vertical Seismic Restraint (Wide Flange) for Vertical Bioshield

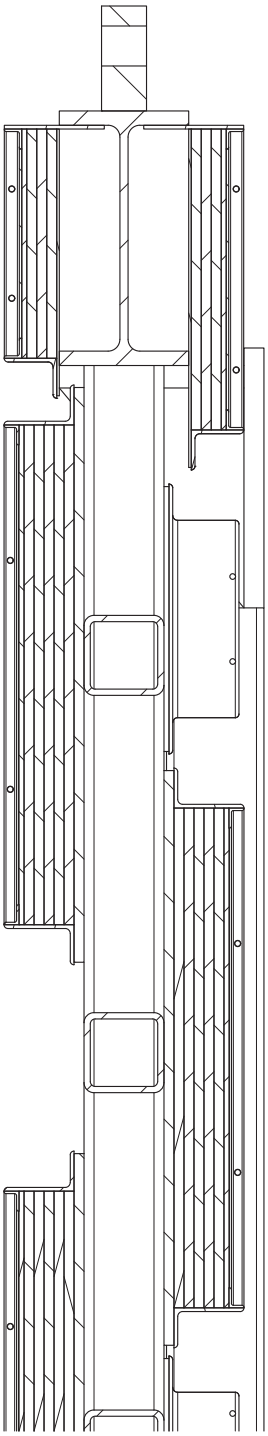


Figure 3.7.3-5: In-Structure Response Spectra at the Bioshield in X-Direction for Bioshield Enveloped In-Structure Response Spectra Using 4 Percent Damping

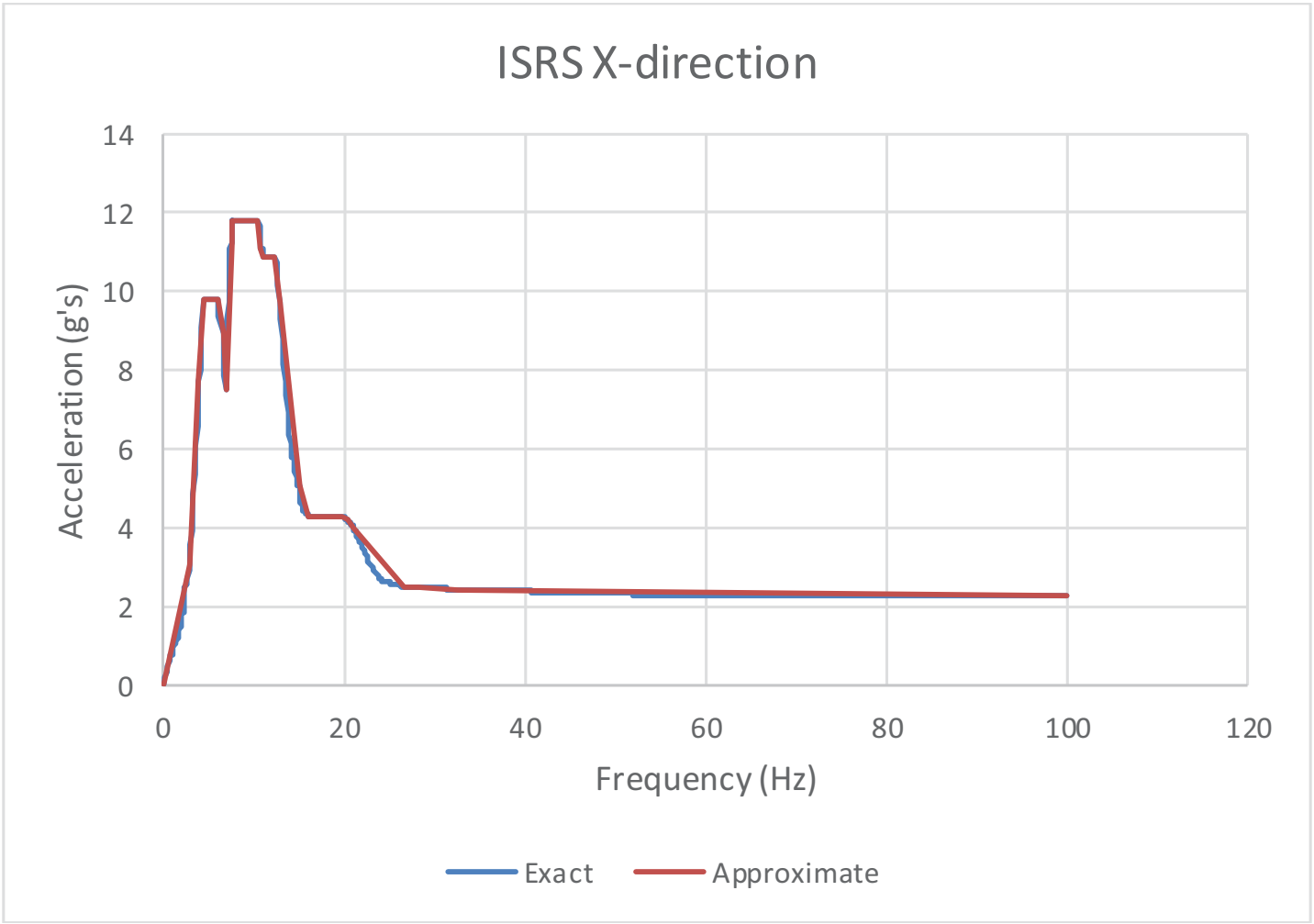
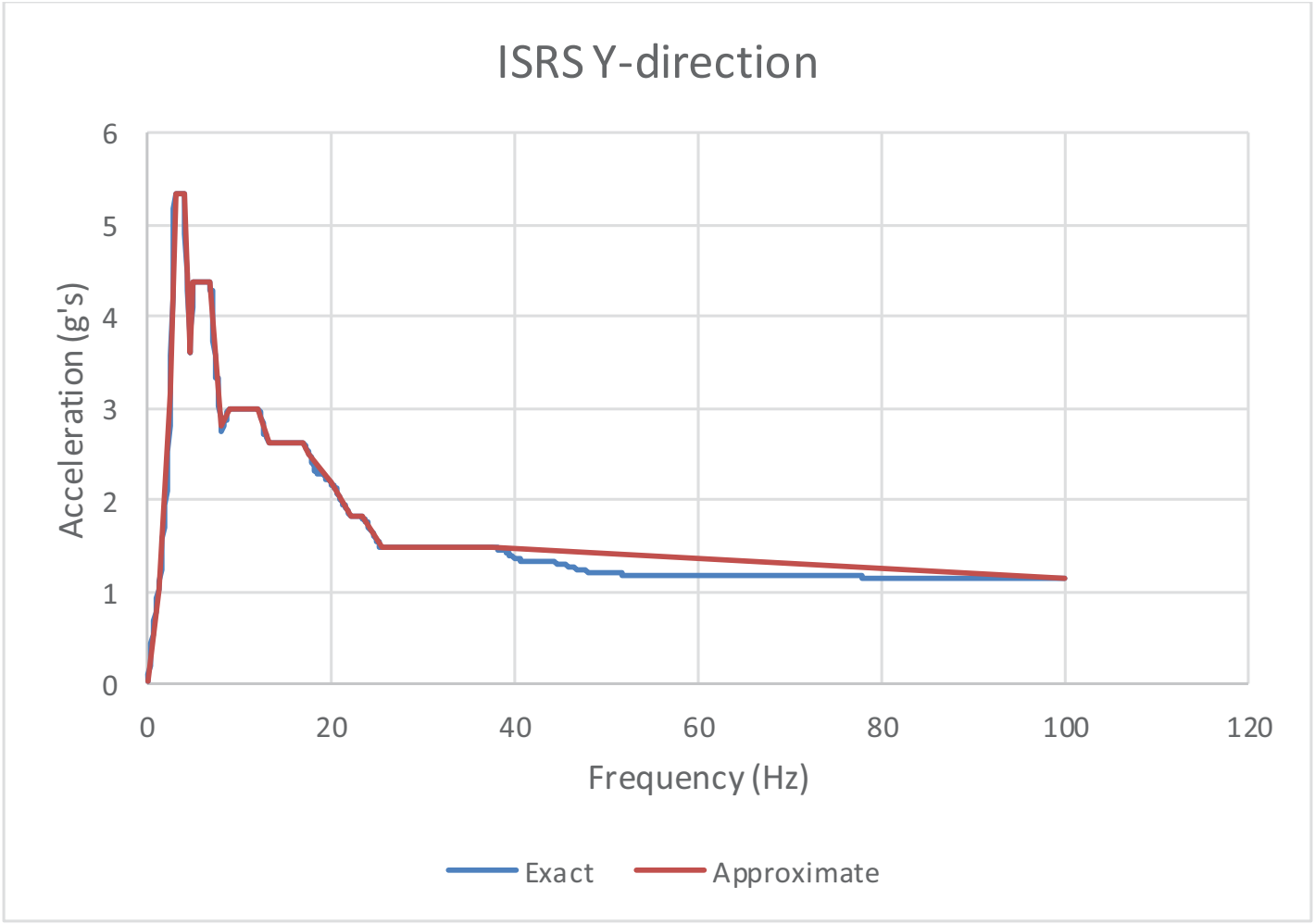


Figure 3.7.3-6: In-Structure Response Spectra at the Bioshield in Y-Direction for Bioshield Enveloped In-Structure Response Spectra Using 4 Percent Damping



3.7.4 Seismic Instrumentation

The seismic monitoring system (SMS) provides conformance with seismic instrumentation requirements of 10 CFR 50 Appendix S.

The SMS is not safety-related or risk-significant. It has no interconnection to safety-related or risk-significant systems. The SMS utilizes seismic instrumentation at various locations on the plant site, data recorders, and a controller located in the Reactor Building (RXB). Data recorders maintain a record of seismic activity. The SMS is Seismic Category I. This categorization includes components, conduits, and instruments of the SMS. The controller processes the data and provides alarm notification to the main control room (MCR) via the plant control system (PCS). Because the PCS is not a Seismic Category I system, additional Seismic Category I annunciation equipment is located in the MCR to alert operators of a seismic event. This annunciation is part of the SMS.

3.7.4.1 Comparison with Regulatory Guide 1.12

The design requires a deviation from guidance in Regulatory Guide (RG) 1.12 "Nuclear Power Plant Instrumentation for Earthquakes" because seismic instrumentation is not included inside containment. There are up to six NuScale Power Modules, each with an integral containment. The containments are flooded as part of the refueling process. The NuScale Power Modules are located within a single pool in the RXB, and are at the same elevation in the building. Instead of locating seismic instrumentation inside containment, instrumentation is located in the RXB.

3.7.4.2 Location and Description of Instrumentation

Sensors are in the free field, in the RXB, and in the Control Building (CRB). In the RXB and CRB, sensors are at locations that have been modeled as mass nodes in the building dynamic analysis so that measured motion can be directly compared to design spectra.

Sensor locations in the RXB and CRB are adequately instrumented for a seismic event by following the following criteria:

- One free-field strong motion accelerator (FFSMA) is located at the free-field ground surface; the location is consistent with site conditions and properties used to determine site-specific ground motion response spectra.
- One FFSMA is a downhole instrument located at the foundation level as close as directly below the free-field ground surface FFSMA as practical.
- One strong-motion accelerometer is located in the RXB on the basemat in the northwest boric acid storage room.
- One SMA is located in the RXB on the basemat in the northeast vestibule room.
- One SMA is located in the RXB in the northwest utilities area room.
- One SMA is located on the RXB roof.

- One SMA is located in the CRB on the basemat in the northeast corridor room.
- One SMA is located in the CRB in the MCR.

3.7.4.3 Control Room Operator Notification

The SMS provides Seismic Category I annunciation in the MCR. Separately, the SMS provides information to the MCR via the PCS.

3.7.4.4 Comparison with Regulatory Guide 1.166

Conformance with RG 1.166, “Pre-Earthquake Planning, Shutdown, and Restart of Nuclear Power Plant Following an Earthquake” Revision 1 is site specific.

3.7.4.5 Instrument Surveillance

The SMS is expected to be operable during all modes of plant operation, including periods of plant shutdown.

Maintenance and repair of the SMS to keep the maximum number of instruments in service during plant operations and shutdown.

3.7.4.6 Program Implementation

COL Item 3.7-11: An applicant that references the NuScale Power Plant US460 standard design will prepare site-specific procedures for seismic instrumentation maintenance and post-earthquake activities. Administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments in-service during plant operations and shutdown. The procedures for post-earthquake activities must provide sufficient information to determine if the level of earthquake ground motion requiring shutdown has been exceeded and appropriate corrective actions to be taken if needed.

Guidance for procedure development is in Regulatory Guide 1.12, “Nuclear Power Plant Instrumentation for Earthquakes,” Regulatory Guide 1.166, “Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake,” and EPRI Report 3002005284, “Guidelines for Nuclear Plant Response to an Earthquake” (Reference 3.7.4-1).

3.7.4.7 Reference

3.7.4-1 Electric Power Research Institute, "Guidelines for Nuclear Plant Response to an Earthquake," EPRI #3002005284, EPRI, Palo Alto, CA, 2015.

3.8 Design of Category I Structures

The NuScale Power Plant US460 standard design includes two Seismic Category I structures, portions of the Control Building and portions of the Reactor Building. Figure 1.2-1 provides a drawing of the site and general arrangement of these buildings. Section 1.2, General Plant Description, provides additional information about the site and primary structures.

3.8.1 Concrete Containment

The NuScale Power Plant US460 standard design does not use a concrete containment. The containment and the reactor vessel are integrated to form the NuScale Power Module.

3.8.2 Steel Containment

3.8.2.1 Description of Containment

3.8.2.1.1 General

The containment vessel (CNV) is an integral portion of the NuScale Power Module (NPM). The CNV houses, supports, and protects the reactor pressure vessel (RPV), reactor coolant system (RCS), and associated structures, systems, and components (SSC). The NPM is located in the Reactor Building (RXB), and the NPM (and thus the CNV) is partially immersed in the reactor pool to facilitate decay heat removal during postulated design-basis events.

The primary functions of the CNV are to provide

- the ultimate barrier against uncontrolled release of radioactivity and radiological contaminants to the environment.
- passive heat transfer from coolant inventory inside the CNV through the CNV wall to the ultimate heat sink during emergency core cooling system (ECCS) operation. Additionally, the CNV supports emergency core cooling by passive retention of coolant inventory during ECCS operation.
- nozzles and penetrations to allow transmittal of signals from SSC inside the CNV.
- nozzles and penetrations to allow for flow into and out of the CNV.
- access ports entryway into the CNV and access ports for potential maintenance of components within the CNV, such as, control rod drive mechanism (CRDM), steam generator (SG), main steam, and pressurizer heater.
- structural support to SSC located inside or attached to the CNV.

Materials in contact with the reactor pool water are corrosion-resistant and do not exhibit unacceptable degradation in service, including external surfaces of the CNV and threaded holes, which are submerged in the reactor pool. During refueling, internal surfaces of the CNV are exposed to reactor pool water, and during design basis events, are exposed to reactor coolant. Material designations are shown in Table 6.1-1 and Table 6.1-2.

The design of the CNV complies with 10 CFR 50 Appendix A General Design Criteria and Principal Design Criteria as described below.

- General Design Criterion (GDC) 1 - The CNV is subject to the design, manufacturing, and operating quality assurance requirements in the Quality Assurance Program Description.
- GDC 2 - Seismic design to withstand the effects of a safe shutdown earthquake (SSE) regarding the CNV is met by using the guidance provided in Regulatory Guide (RG) 1.29, "Seismic Design Classification for Nuclear Power Plants," Revision 5.

- GDC 4 -The CNV is designed to accommodate the effects of and be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs).
- GDC 16 - The CNV is designed to provide a leak-tight barrier and to contain the CNV design pressure during design basis events.
- GDC 50 - The CNV is designed to ensure the component, access openings, penetrations, and containment heat removal systems have the capability to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA.
- GDC 53 - The CNV is designed with provisions to permit inspection and testing for periodic verification that the CNV remains within the limits defined by the design basis.

3.8.2.1.2 Containment Configuration Description

The CNV is an American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME BPVC) Section III, (Reference 3.8.2-5) Class MC pressure vessel that, as permitted by NCA-2134(c), is constructed and stamped as a Class 1 vessel. The CNV is cooled by the ultimate heat sink. The CNV is supported by the RXB floor through the CNV support skirt and supported laterally at three RXB locations, one on each operating bay wall and pool wall through CNV support lugs. Figure 6.2-1 shows the CNV. Design characteristics, including elevations, of the CNV are shown in Table 3.8.2-1.

Table 9.2.5-1 provides ultimate heat sink parameters.

The top support structure is attached to the CNV upper head as described in Section 9.1.5. The top support structure is used to connect, disconnect, and move the NPM. The top support structure also acts as a piping support. Containment isolation valves (CIVs) are located on top of the CNV head underneath the top support structure platform. Piping from the CIVs travels up through the supports and platform of the top support structure to disconnect flanges for each piping system.

Section view of the CNV is shown in Figure 6.2-1 and Figure 6.2-2a and Figure 6.2-2b.

The CNV is partially immersed in the reactor pool within the RXB, which is a Seismic Category I structure primarily embedded in soil. Discussion of the RXB is in Section 3.8.4 and Section 3.8.5. The reactor pool provides a passive heat sink for containment heat removal under LOCA conditions. The CNV is designed to withstand the environment of the reactor pool as well as the high pressure and temperature of a design-basis event.

Section 6.2.1 discusses calculated peak CNV pressures and temperatures.

3.8.2.1.3 Containment Vessel Support

The CNV support skirt is integrally welded to the CNV lower head. The boundary between the CNV and the Class MC CNV support skirt support is at the attachment weld. The attachment weld is beyond 2t (twice the minimum required thickness of the vessel) from the CNV pressure boundary and is included in the jurisdiction of the CNV support skirt.

The attachment weld between the CNV and the CNV support lug belongs to the CNV.

The upper CNV includes the CNV upper shell subassembly, upper steam generator access subassembly, upper flange subassembly, and CNV upper head.

The RPV support ledge is welded to the CNV shell. The boundary between the CNV and the Class 1 RPV support ledge support is at the attachment weld. The attachment weld between the CNV and the RPV support ledge belongs to the CNV.

The integral RPV seismic restraint with support shims in the CNV lower head interfaces with the RPV seismic pin on the RPV lower head. The integral RPV seismic restraint feature is integral to the CNV and is designated as a Class MC vessel, along with the threaded hole used for attaching the support shim. The boundary between the support shims and the integral RPV seismic restraint on the CNV is at the surface of the RPV seismic restraint.

The CRDM support frame is bolted to the CNV upper head at the CRDM access flange. The boundary between the CNV and the Class 1 CRDM support frame support is located at the mating face of the support to the CNV. The CRDM support frame bolts, nuts, and washers belong to the Class 1 support.

The upper and lower decay heat removal system (DHRS) condenser supports are welded to the CNV shell. The boundary between the CNV and the Class 2 DHRS condenser supports is located at the attachment weld.

3.8.2.1.4 Access and Manways

The CNV has a variety of access ports and openings. On the CNV shell there are four steam generator access ports, two pressurizer access ports, and two manways. The steam generator access ports are aligned with the main steam nozzle access ports on the RPV. The pressurizer access ports are aligned with the pressurizer heater flanges on the RPV. The manways provide access to the inside of the CNV. On the CNV head there is a control rod drive mechanism access port that provides the ability to access the CRDMs bolted to the RPV head. Figure 6.2-1 shows each access opening location.

3.8.2.1.5 Piping Penetrations

Piping system penetrations are located sufficiently remote from the reactor core that neutron or gamma exposure is negligible under normal operating conditions.

Piping penetrations consist of integrally forged nozzle penetrations with welded safe ends on the internal and external ends of the nozzle, safe end-to-pipe welds on the internal end of the nozzle, and safe end-to-valve welds on the external end of the nozzle. The penetration, safe ends, safe end-to-nozzle attachment welds, and safe end-to-valve-welds are classified as Class MC. The CNV nozzles weld to the containment isolation test fixtures (CITFs), which are in turn welded to the isolation valves. The CNV main steam nozzles, DHRS nozzles, and interior and exterior safe ends attached to these nozzles are evaluated to Level B Primary Stress limits for Level C and Level D events. The safe end-to-pipe weld on the internal end belongs to the pipe and is classified the same as the pipe. Consistent with the remainder of the CNV, piping penetrations, safe ends, safe end-to-nozzle welds, and safe end-to-valve welds are constructed as ASME BPVC, Section III.

- The chemical and volume control system and the reactor coolant system containment vessel piping penetrations have internal surfaces exposed to reactor coolant water environments.
- Main steam system and feedwater system containment vessel piping penetrations have internal surfaces exposed to secondary water environments.
- The reactor component cooling water containment vessel piping penetrations have internal surfaces exposed to reactor component cooling water environments.
- The containment flooding and drain system and the containment evacuation system containment vessel piping penetrations have internal surfaces exposed to RXB pool water environments.

Each NPM has a containment system (CNTS) with a containment boundary designed to prevent or limit release of radioactive materials under postulated accident conditions. The containment boundary is formed by the CNV and by CIVs and passive containment isolation barriers that are used to prevent releases through the penetrations in the CNV. Table 6.2-4 lists penetration openings in the CNV.

Mechanical penetrations for process fluids or gases include top head penetrations for process flows, penetrations for ECCS valve actuator assemblies, and penetrations for DHRS process flows.

Section 6.2 discusses CIVs.

3.8.2.1.6 Containment Electrical Penetration Assemblies

The electrical and instrumentation and controls electrical penetration assemblies bolt to flanges on the CNV shell. The boundary between the CNV and the electrical penetration assembly is at the flange face and includes the bolting (studs/nuts).

3.8.2.1.7 Emergency Core Cooling System Trip/Reset Valve Penetrations

Each ECCS reactor vent valve and ECCS reactor recirculation valve has two trip valves and a reset valve in a trip/reset valve assembly. The trip valves are connected to the main valve's control chamber on a tubing line that allows discharge of the main valve's control chamber to the CNV. The trip/reset valve opening and closing only affects ECCS operation and does not affect containment boundary. Section 6.2 discusses the passive containment boundary. The reset valve is connected to the main valve's control chamber by the same tubing line and allows coolant injection from the CVCS into the main valve's control chamber through the RCS injection line.

Functional description of ECCS valves is provided in Section 6.3.

Each ECCS trip/reset valve assembly is a part of the containment pressure boundary and some internal parts are part of the reactor coolant pressure boundary (RCPB). The reset tubing from the RCS injection line and the trip tubing between the main valve and the trip/reset valve assembly are also part of the RCPB. Table 6.1-1 identifies the materials of construction for the ECCS.

3.8.2.1.8 Attachments

The CNV provides lateral and vertical support to the RPV at four locations. The CNV-RPV support ledge is integrally welded to the CNV inner surface. The RPV-CNV support ledge is integrally welded to the RPV outer surface. Horizontal shims are present between the support ledges for fit-up purposes and to transfer lateral loads. Vertical shims are also present between the supports for fit-up purposes. A CNV-to-RPV pin with threaded collars connects the structures and prevents vertical lift off. The hole on the CNV-RPV support ledge is oversized to allow for radial expansion of the RPV. Only one of the vertical shims is part of the CNV jurisdictional boundary.

Lateral support of the RPV during a seismic event is provided at the CNV inside surface at the bottom of the CNV by an integral seismic restraint. The lower CNV seismic support pads are bolted to the CNV lower head and interfaces with the RPV seismic cap on the RPV lower head. The support allows free vertical motion of the RPV, and acts as a lateral support during seismic events. The CNV lower seismic support pads and attachment hardware are outside the scope of Section III of the ASME Code, in accordance with ASME Subparagraph NF-1110(e).

Lateral support of the CRDMs is provided by the CNV at the CRDM support frame located on the underside of the CNV upper head.The support frame is

bolted to the CNV head. For the purposes of the CNV, the CRDM support frame is classified as an ASME BPVC, Section III Class 1 Support and constructed to the rules of ASME BPVC, Section III, Subsection NF. The boundary between the CNV and the Class 1 CRDM support frame support is located at the mating face of the support to the CNV. The CRDM support frame bolts, nuts, and washers belong to the Class 1 support.

Other items are attached to the interior and exterior of the CNV (e.g., decay heat removal system passive condensers, piping supports, access platforms and ladders, and instrument enclosures). Piping supports, including the DHRS passive condenser supports, follow ASME Code, Section III, Subsection NF. Other items are nonstructural attachments in accordance with ASME Code, Section III, Subsubarticle NE-1130 because they are not pressure retaining and do not contribute to support of the CNV. For all attachments (structural and non structural), boundary is at the surface of the CNV shell and the weld between the attachment and the CNV is considered part of the attachment.

3.8.2.2 Applicable Codes, Standards, and Specifications

3.8.2.2.1 Codes, Standards, and Specifications

Codes, standards, and specifications meet acceptance criteria in Design Specific Review Standard Section 3.8.2 and include industry standards to cover design, materials, fabrication, testing, and inspections of the CNV.

Periodic inspections and testing programs meet ASME BPVC and Operation and Maintenance (OM) Codes in accordance with 10 CFR 50.55a.

3.8.2.2.2 Code Classification and Compliance

Classification and compliance of the CNV is in accordance with the ASME BPVC. The CNV is an ASME Code Section III, Class MC component including

- access and inspection openings and associated flanges.
- penetrations for ECCS trip/reset valves and CIVs.
- openings and associated flanges for electrical penetration assemblies.

As permitted by NCA-2134(c), the CNV is constructed and stamped as an ASME BPVC, Section III, Class 1 vessel in accordance with ASME BPVC, Section III, Subsection NB except that overpressure protection is in accordance with NE-7000.

The CNV support skirt and lug are classified as an ASME BPVC, Section III, Class MC support. The CNV support skirt is constructed as an ASME BPVC, Section III, Class 1 support in accordance with ASME BPVC, Section III, Subsection NF.

3.8.2.3 Loads and Load Combinations

Stresses and fatigue for the CNV pressure retaining components are evaluated in accordance with ASME Code, Section III, Subsection NB. The loads for which the CNV is designed are:

DW	Deadweight of the CNV, which includes the weight of the structure, any internal equipment or piping systems, and enclosed water. Deadweight refers to any moments or forces due to the deadweight.
B	Buoyancy provided to the CNV by the reactor pool water.
DFL	dynamic fluid loads
OBE	operating basis earthquake
L	lifting and handling loads
TR	transportation
TAM	thermal anchor movement
P	Highest operating pressure load due to normal and abnormal operating conditions resulting from pressure variations either inside or outside the CNV. The lowest internal pressure of less than 0.1 psia during normal operating conditions is also considered. The external pressure during operating conditions is 50 psia.
TH	Transient loads due to normal operating conditions and anticipated operational occurrences, infrequent and accident, resulting from thermal and pressure variations either inside or outside the CNV.
M	Piping mechanical and thermal loads produced on the nozzle penetrations and safe ends from piping system due to pressure and thermal variations in the piping system.
DBPB	Design basis pipe break other than FWPB, MSPB, or LOCA dynamic loads due to a postulated pipe break or spurious valve actuation of the reactor safety valve, reactor vent valve, or reactor recirculation valve. This classification includes CVCS pipe breaks in RPV high point degasification, pressurizer spray, RCS discharge, and RCS injection piping inside of containment.
ECCS	Emergency core cooling system includes spurious actuation of the reactor vent valves and reactor recirculation valves.
RSV	Reactor safety valve actuation including asymmetric pressurization and valve jet impingement.
RF	Refueling loads
PRE	Bolt preload
GS	Gasket seating pressure
SSE	In a safe shutdown earthquake, the CNV is designed to withstand vertical and lateral loading due to seismic ground accelerations considering the appropriate damping values for the CNV in accordance with RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Revision 1. The operating basis earthquake (OBE) is defined as one-third of the SSE. In accordance with Appendix S of 10 CFR 50, OBE seismic loads need not be explicitly analyzed in the design analysis; however, they are considered in the fatigue analysis.

3.8.2.3.1 Pressure Loading

The CNV does not have pressure relief devices. In accordance with NE-7120(b), the CNV internal design pressure is set to bound all service level pressure transients. Internal design and external design pressures are not coupled. Internal design pressure is evaluated with 0 psia on the outside.

For fluid penetrations, internal design pressure is based on the internal design pressure of the attached piping.

The CNV external design pressure is from the reactor pool water static head. The internal design and external design pressures are not coupled. External design pressure is conservatively evaluated with an internal pressure at 0 psia.

For fluid penetration nozzles internal to the CNV, external design pressure for the nozzle is equal to the internal design pressure for the CNV. For fluid penetration nozzles external to the CNV, the external design pressure for the nozzle is equal to the external design pressure for the CNV.

3.8.2.3.2 Seismic Loading

Methodologies and structural models that are used to analyze the dynamic structural response, due to seismic loads acting on the NPM, are described in TR-121516 (Reference 3.8.2-6) and TR-121517(Reference 3.8.2-7).

3.8.2.3.3 SCRAM Loading

The SCRAM loads are mechanical loads on the RPV due to impact of dropping control rod drive shafts and control rod assemblies during a reactor trip event. These loads are transferred to the CNV through the containment vessel-reactor pressure vessel supports. At the containment vessel-reactor pressure vessel supports these loads are negligible.

3.8.2.3.4 Blowdown Loading

Short-term transients are those caused by the failure or actuation of Class 1 and 2 piping and valves, and include high-energy line breaks. Evaluation of short-term transients within the NPM is addressed in TR-121516 and TR-121517. These events potentially result in system internal pressure waves and asymmetric cavity pressurization waves exterior to the pipe break or valve outlet and require special treatment because of rapidly changing thermal hydraulic conditions and resulting dynamic mechanical loads.

3.8.2.3.5 Transient Loading

The design transients are categories as defined in ASME BPVC Service Levels for Class 1, 2, and 3 components including number of events for each design transient based on the plant design life (Section 3.9.1).

Most design-basis events are simulated using NRELAP5 as described in Section 3.9.1. Results from the NRELAP5 analysis for representative nodes and control variables for regions of the CNV are selected to provide representative time history results. Time history pressure, temperature, phase composition, velocity, and mass flow rate transient results are provided for regions inside and outside the CNV up to the outermost isolation valve. A few of the design-basis events are simple. Characterization of time history results for these events can be made based on event definition and do not require an NRELAP5 analysis to adequately analyze the event.

Design-basis events that are simulated using NRELAP5 use the NRELAP5 base model. The NRELAP5 base model contains the NPM reactor core, hydraulic regions representing the primary and secondary fluid systems, containment and reactor pool. The NRELAP5 base model includes heat structures to simulate heat transfer between the regions, and both safety and nonsafety controls to simulate plant actions and operations. Section 1.5 discusses validation of NRELAP5 software and Section 6.2.1 discusses software's use in CNV analyses.

Time-history thermal analysis data are applied to CNV finite-element thermal models to determine CNV metal temperatures for design-basis events. Resulting temperature gradients in the CNV from thermal analysis and NRELAP5 pressure transient data are then applied to a CNV structural model to determine stresses on the CNV.

3.8.2.3.6 External Environment Loading

The effects of missiles and external events such as a hurricane, tornado, aircraft hazards, and explosion pressure waves are not considered because the CNV is protected from these effects by the Seismic Category I RXB.

3.8.2.3.7 Lifting and Transportation

The lifting and handling loads analysis considers the full range of positions during transportation evolution, field installation work, transfer to and from the upender, and installation in the plant. Lifting and handling loads are also considered for the full NPM refueling evolution, including lift and transport of the NPM and its subassemblies using the Reactor Building crane, assembly and disassembly of the CNV and the RPV, and flange fastener tensioning and de-tensioning.

Transportation loads are evaluated with the CNV in the horizontal position. Shipping restraints are installed between the CNV and the RPV at the location of the lateral support lugs at the CNV upper flange.

The lifting, handling, and transportation load contains a 15 percent dynamic load factor, for a total load of 115 percent times the DW load applied at all lifting and transportation support points.

Lifting, handling, and transportation loads are not required to meet ASME stress limits. However, the Service Level B primary limits are used as the allowable limits for the lifting, handling, and transportation loads. The platform mounting assemblies are analyzed to ensure minimum safety factors of five for material ultimate strength and three for material yield strength per Reference 3.8.2-3 and are maintained for dual-load-path loading conditions considering the dynamic load factor specified above.

3.8.2.3.8 Load Combinations

The ASME Code Design, service level (Level A, Level B, Level C, Level D), and test loads and load combinations for the CNV and CNV support design are shown in Table 3.8.2-2 through Table 3.8.2-5. Load combinations meet requirements of ASME Code, Section III, Paragraph NCA-2141(b) and consider the guidance of RG 1.57, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components.” Loads and load combinations used in the analysis are a part of the method of evaluation. Alternatives to the RG 1.57 load combinations are discussed below.

3.8.2.3.9 Alternatives to Regulatory Guide 1.57 Load Combinations

Load combinations used for design of the CNV follow the same load combinations specified for the RPV, which follow guidelines in NUREG-0800, Standard Review Plan 3.9.3 for ASME Code Class 1, 2, and 3 components and component supports, and core support structures. These load combinations differ from the suggested load combinations in RG 1.57 for metal primary reactor CNTS components. Some of the differences are load combination of seismic loads with LOCA loads evaluated to Service Level C, and loads resulting from a pipe break (i.e., pipe whip and jetting).

Tables 3.8.2-2, 3.8.2-3, 3.8.2-4, and 3.8.2-5 report the applicable combination of loadings considered in the corresponding ASME Code qualifications of individual components for various service levels. The classification of the stress (i.e., primary, secondary, or peak) resulting from the loading is dependent on the geometry and the location. Table XIII-2600-1 and Table XIII-2600-2 of Mandatory Appendix XIII of Section III of the ASME Code report how to appropriately classify stresses for various locations. In addition, Paragraph XIII-1220 of Mandatory Appendix XIII of Section III of the ASME Code provides the basis for determining stresses and how those stresses should be considered. The combination of Paragraph XIII-1220 and Tables XIII-2600-1 and XIII-2600-2 are used to classify the stresses for each of the load combinations. Load combinations in RG 1.57 are intended for structures designed, fabricated, inspected, and tested to ASME Code, Section III, Subsection NE requirements. Load combinations used for the CNV are typical load combinations used for vessels designed, fabricated, inspected, and tested to ASME Code, Section III, Subsection NB requirements. Vessel load combinations and allowable limits differ from containment structures because inspection and testing requirements for vessels are more restrictive, which allows a higher design limit. Justification for why this difference is acceptable for the CNV is below.

During normal operation, the inside of the CNV is maintained under a vacuum and is partially immersed in the reactor pool. The reactor pool is the ultimate heat sink that removes residual core decay heat during normal and accident conditions. The design prevents isolated pockets of concentrated gases. The upper portion of the CNV is fabricated of F6NM, which is a martensitic stainless steel material. The bottom portion of the CNV is F6NM and FXM-19 (austenitic stainless steel). Typical PWR metal containment structures are constructed from carbon steel plate.

Pressure boundary forgings and weld filler materials are tested for mechanical and fracture toughness to requirements of ASME Code, Section III, Article NB-2000. The CNV is a shop-fabricated vessel, fabricated to the requirements of ASME Code, Section III, Article NB-4000, with all martensitic stainless steel welds post-weld heat treated in the shop. Many ASME Code requirements for an NB Class 1 and a Class MC vessel are similar. In an NB Class 1 vessel, these welds are required to have a volumetric and either liquid penetrant or magnetic particle inspection performed per ASME Code, Section III, Subarticle NB-5200. The corresponding welds in a Class MC vessel only require a fully radiographed inspection per Subarticle NE-5200.

After fabrication of the CNV is complete, a shop hydrostatic test of the vessel is performed to Article NB-6000 requirements. Before hydrostatic testing, 100 percent of the pressure boundary welds are inspected. Inspection is performed in accordance with Subarticle NB-5280 and Subarticle IWB-2200 using examination methods of ASME Code, Section V except as modified by ASME Code, Section III, Paragraph NB-5111. Hydrostatic pressure and temperature are held for a minimum of 10 minutes. The pressure is then decreased to design pressure and held, then the CNV is inspected for leaks. After the test is complete, pressure boundary welds are inspected again to the same requirements used before the test. The ASME Code, Section III, Article NB-6000 hydrostatic test is performed to a greater pressure than required by Article NE-6000. The CNV is tested to a pressure 25 percent greater than design pressure in accordance with NB-6221.
Paragraph NE-6321 specifies a minimum test pressure of only 110 percent and Paragraph NE-6322 specifies a maximum test pressure of 116 percent. Thus, the NB-6000 requirement is 15 percent greater than conventional steel containment.

The CNV design pressure and temperature bound design-basis events including a LOCA. The design condition pressure exceeds requirements of ASME Code, Section III, Paragraph NCA-2142.1(a) and NB-3112.1(a) by bounding the most severe Level A service level pressure, and requirements of Paragraph NE-7120(b) by the design not exceeding service limits specified in the design specification.

The design does not have a typical postulated LOCA compared to traditional PWR reactor coolant systems. Reactor coolant is captured by the CNV and passively recirculated through the RPV and core by the ECCS (Section 6.3). The reactor coolant level is never below the level of the top of the core and reactor coolant makeup is not required. The reactor coolant piping within the

CNV is NPS 2. Secondary-side piping for feedwater and main steam are larger. Breaks in these piping systems are excluded as discussed in Section 3.6.2. Pipe breaks for reactor coolant piping inside containment and spurious opening of a reactor safety valve or reactor vent valves are addressed in TR-121516 and TR-121517. Pipe breaks and spurious valve openings inside the CNV are evaluated as design-basis pipe break (DBPB). The DBPB load is evaluated to Level C service limits and, when combined with SSE loads, is evaluated to Level D service limits. Chemical and volume control system (CVCS) lines located outside of the CNV are evaluated for pipe breaks at Level D service limits. Section 3.6 discusses blast effects, pipe whip, and jet impingement caused by a pipe break.

The guidance in RG 1.57 recommends DBPB loads to be evaluated to Level B service limits and DBPB combined with SSE loads to be evaluated to Level C service limits. Because the CNV is designed, fabricated, and tested as an NB Class 1 vessel, evaluation of these loads to more restrictive allowable limits is conservative. The increased testing for a Class 1 vessel discussed below offsets the more restrictive allowable limit guidance in RG 1.57.

The CNV design does not include post-accident carbon dioxide inerting; thus any load due to this event is not applicable. Control of hydrogen within the CNV is discussed in Section 6.2.5.

Inservice inspection (ISI) provides an essential function for CNTS integrity by ensuring no new leakage paths are present. Age-based failure mechanisms are detected and mitigated through the compact and accessible design of the CNV, along with inspections and examinations performed in accordance with ASME Code, Section XI, Division 1. The CNV components and welds are inspectable. The CNV design allows for visual inspection of the entire inner and outer surfaces and is designed to accommodate comprehensive inspections of welds, including volumetric and surface inspections. Welds are accessible and essentially 100 percent inspection coverage can be achieved. Periodic, comprehensive ISI ensures that degradation mechanisms are detected and addressed before CNV integrity is threatened.

ASME Code, Section XI, Subsection IWE requires, for Class MC structures, 80 percent of the containment boundary be accessible for a single-side visual examination for SSC subject to normal degradation and aging.

Based on the high pressure and the safety function of the CNV, enhanced inspection requirements are needed for the CNV. Therefore, part of the CNV has augmented inspection criteria. Section 3.8.2.7 addresses CNV inspection requirements. The CNV design allows visual inspection of the entire inner and outer surfaces; therefore, developing an undetected leak through the metal pressure boundary is unlikely.

The CIVs are located outside of the CNV. The reduced ISI requirements permitted by ASME Code, Section XI, for small primary system pipe welds between the CNV and the CIVs are not applied to these welds. Welds between the CNV and the CIVs are ASME Code NB Class 1 and are

inspected with a volumetric and surface exam at each test interval. The CNV design allows comprehensive inspections of welds, including volumetric and surface inspections. Pressure boundary welds are accessible and there are no areas that cannot be inspected.

The simplicity of the NPM design includes minimizing the number of containment penetrations required. The CNV has a limited number of access openings, manways, and electrical penetration assemblies, and each penetration uses the same seal design. The CNV flange separating the upper and lower CNV assemblies uses the same seal design as the RPV and is similar to the access opening and manway seal designs. There are a limited number of containment fluid line penetrations. Fluid line penetrations are protected in one of three ways: by primary system CIVs, by a closed loop and a secondary system containment isolation valve, and by a closed loop inside and outside containment. There are no air locks, flexible sleeves, or nonmetallic boundaries in the CNV design.

The CNTS meets the intent of 10 CFR 50 Appendix J, to ensure leak tightness of the CNV and ensure new leak paths do not develop by the local leak rate testing and ISI performed on the CNV and is facilitated by the CNV design features:

- The CNV is an ASME Code Class 1 pressure vessel with no internal subcompartments.
- Preservice test and inspections are similar to RPV requirements, including hydrostatic pressure tests.
- A preservice design pressure leakage test is performed before the NPM is placed into service, as described in Section 6.2.6.
- The leakage pathways, each with similar seal designs, are tested in accordance with Type B or Type C requirements of 10 CFR 50 Appendix J.
- The ISI Program and planned CNV examinations meet ASME Code Class MC with augmented inspections criteria to ensure no new leakage pathways develop.
- Disassembly and reassembly procedures and controls for the CNV are similar to the RPV.

Containment vacuum pressure and leak rate into the CNV are constantly monitored during normal operation. The containment volume and evacuated operating conditions allow for wide-ranging detection capabilities for liquid or vapor leakage. Automatic engineered safety feature actuation systems initiate on high containment pressure; therefore, containment pressure is maintained below 9.5 psia during operations.

In summary, the CNV is made of corrosion-resistant materials, has a low number of penetrations, and no penetrations have resilient seals. Penetrations are either ASME Code, Section III, NB Class 1 flanged joints capable of 10 CFR 50 Appendix J, Type B testing or NB Class 1 welded nozzles with

isolation valves capable of 10 CFR 50 Appendix J, Type C testing. The use of welded nozzles and testable flange seals at the containment penetrations ensure that 10 CFR 50 Appendix J Type B and Type C testing provides an adequate assessment of overall containment leak rate.

Use of typical RPV load combinations for Class 1 vessels is more applicable to the CNV than using the load combinations in RG 1.57 because of the increased quality of the fabrication, inspection, and testing required by ASME Code, Section III, Subsection NB for a Class 1 vessel. The intent of RG 1.57 is satisfied by evaluating LOCAs and seismic loads. Evaluations of these loads are to allowable limits, which provide a design that performs its intended function during design-basis events.

3.8.2.4 Design and Analysis Procedures

The CNV design and analysis conform to requirements of ASME Code, Section III, Subarticle NB-3200 and CNV support design and analysis conform to requirements of Subarticle NF-3200. The CNV fabrication conforms to requirements of Article NB-4000 and Article NF-4000. Nondestructive examination of pressure retaining and integrally attached materials meet the requirements of Article NB-5000 and Article NF-5000.

Detailed analyses of ASME Code primary stresses for the CNV use a combination of standard textbook hand calculations for simple structures, such as nozzles, and the ANSYS general purpose finite element program for more complex geometry, such as the CNV top head. Other ASME Code evaluations are performed using ANSYS. Buckling of the torispherical lower head is evaluated using ASME Code Case N-759-2. Limit analyses to determine lower bound limit buckling loads may be employed in lieu of Code Case N-759-2. Evaluation of buckling, or elastic instability that results in collapse, is considered as part of the limit load analysis.

Stress analyses are performed using the load combination defined in Section 3.8.2.3. The allowable limits are in accordance with ASME Code, Section III, Subarticle NB-3200 and NF-3200. Allowable limits are based on the mean metal temperature for the applicable service level or a conservative higher temperature (i.e., design temperature).

Computer code verification, validation, configuration control, and error reporting and resolution are performed according to the quality assurance requirements of Chapter 17.

3.8.2.4.1 Containment Vessel Stress Analysis

The CNV is evaluated for deadweight, buoyancy, internal pressure, blowdown loads, seismic, thermal, and pressure transient loads.

Minimum wall thickness for nozzles on the CNV shell, nozzle reinforcement, and limits of reinforcement along the CNV wall and normal to the CNV wall are in accordance with ASME Code, Section III, Subarticle NB-3300. If rules of

NB-3300 are not satisfied, then Subarticle NB-3200 design by analysis is applied as permitted by Paragraph NB-3331(c).

Integrity of the pressure-retaining function of the CNV is provided by compliance with the ASME Code. The evaluation of the stress levels and fatigue usage for the CNV pressure boundary is calculated for specified loading conditions discussed in Section 3.8.2.3 and demonstrates that the values are less than the allowable limits.

The CNV shell, top head, and bottom head are evaluated for buckling during normal operating conditions. During normal operating conditions, a vacuum exists inside the CNV that causes an external pressure on the outside surface of the CNV. Also, during a Level D seismic event, the CNV sees a vertical compressive load that is also checked for buckling. Buckling checks are made using ASME Code Case N-759-2. Limit analyses to determine lower bound limit buckling loads may be employed in lieu of Code Case N-759-2.

Additionally, buckling is checked on the inside knuckle region of the top head and bottom head. Internal pressure causes compression in the knuckle that is checked using hand calculation based on equation 4.3-19 from ASME Code, Section VIII, Division 2.

Piping and electrical penetrations are evaluated using the loads and load combinations discussed in Section 3.8.2.3. The effects of the penetration loads on the CNV top head shell are also evaluated.

Stress and fatigue results are evaluated in accordance with ASME Code, Section III, Subarticle NB-3200 limits. The fatigue analysis of the CNV process fluid penetrations considers the effects of the PWR environment in accordance with RG 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," and NUREG/CR-6909 (Reference 3.8.2-1). The ASME Code design report summarizes results of CNV analyses and evaluations.

3.8.2.4.2 Containment Vessel Lateral Support Lugs

The CNV is supported in the RXB operating bay by lateral support lugs located on the CNV upper shell. The CNV lateral support lugs are attachments to the CNV, as defined by ASME Code, Section III, Paragraph NB-1132.1(a), and use the rules of ASME Code, Section III. The lateral support lugs are constrained by the NPM lug restraints located on the NPM bay walls. The loads and load combination discussed in Section 3.8.2.3 are used to evaluate the CNV support lug. Stress and fatigue results are evaluated in accordance with Subarticle NB-3200 limits.

3.8.2.4.3 Containment Vessel Lower Support

The CNV is an upright cylinder with heads at the top and bottom. A support skirt at the bottom of the CNV provides vertical and horizontal support. In

addition, lateral support lugs near the top of the vessel provide support against the reactor compartment walls. The CNV along with the support skirt is immersed in the below-grade reactor pool, which provides a passive heat sink.

The bottom of the CNV is supported vertically and laterally by the CNV support skirt. The CNV support skirt is an ASME Code Class MC support that is constructed as an ASME Code, Section III, Class 1 support in accordance with the requirements of Article NF-4000. The support skirt is located below the CNV bottom head. Vertical support is provided by bearing on the reactor pool floor. The loads and load combination discussed in Section 3.8.2.3 are used to evaluate the CNV support skirt. Stress and fatigue results are evaluated in accordance with ASME Code, Section III, Subarticle NF-3200 limits.

3.8.2.4.4 Containment Vessel Reactor Pressure Vessel Supports

The-CNV-to RPV support ledges are discussed in Section 3.8.2.1.8. The loads and load combination discussed in Section 3.8.2.3 are used to evaluate the CNV-to-RPV support ledges. Stress and fatigue results are evaluated in accordance with ASME Code, Section III, Subarticle NF-3200 limits.

3.8.2.4.5 Containment Vessel Ultimate Capacity

NuScale performed a series of non-linear (plastic) three-dimensional finite element analysis to determine the ultimate pressure capacity of the CNV; the analyses conform to guidance in Appendix A of NUREG/CR-6906 (Reference 3.8.2-2). The failure criteria that determine the ultimate pressure capacity of the CNV are based on guidance in RG 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure." Technical report TR-121516, CNV Ultimate Pressure Integrity (Reference 3.8.2-6), addresses details of the predicted containment internal pressure capacity above design pressure. The CNV is assumed to fail when one of the following criteria are met:

- A maximum global membrane strain away from discontinuities of 1.5 percent is reached.
- Loss of bolt preload occurs at any bolted CNV opening.
- Buckling occurs in the knuckle of the upper or lower CNV head.
- A flange gap that exceeds the calculated allowable values is reached at the outer O-ring of any bolted CNV opening.
- Solution divergence occurs.

Technical report TR-121516 describes the methodology and the results of the CNV ultimate pressure integrity evaluation.

The analysis evaluates this failure mode to determine if it is the limiting condition for the pressure capacity of the containment. Buckling is considered in terms of computing the ultimate pressure capacity of the CNV.

3.8.2.4.6 Containment Vessel Radiation Exposure Effects

The materials of construction of the lower CNV do not lend themselves to fracture toughness concerns resulting from radiation degradation effects. Section 6.2 provides further discussion.

3.8.2.4.7 Containment Vessel Cyclic Fatigue

The CNV is evaluated for fatigue based on the ASME Code, Section III, Paragraph NB-3222. Applicable cyclic, dynamic, pressure, and thermal transient loads and load combinations, discussed in Section 3.8.2.3, are considered in the fatigue evaluation. For CNV process fluid penetrations classified as ASME Code Class 1, the fatigue analysis considers effects of the PWR environment in accordance with RG 1.207 and NUREG/CR-6909.

Section 3.7.3 discusses operating basis earthquake seismic loads and analysis.

3.8.2.5 Structural Acceptance Criteria

The design limits imposed on the parameters that quantify structural behavior of the containment comply with the ASME BPVC, Section III, Division 1, Subsection NB, and RG 1.57. Design parameters include allowable stresses, strains, and gross deformations. The ASME Code limits for defined load combinations are in Table 3.8.2-2 through Table 3.8.2-5. The CNV is fabricated, installed and tested according to ASME Code, Section III, Subsection NB and Subsection NF.

The CNV is designed to meet the maximum leakage rate as discussed in Section 6.2 and TR-123952 (Reference 3.8.2-4). Items that form the CNV pressure boundary and supports are stamped in accordance with the applicable section of the ASME Code used for their design or fabrication.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

The CNV uses no special construction techniques. Materials of construction are shown in Table 6.1-1 and Table 6.1-2.

3.8.2.7 Testing and Inservice Inspection Requirements

Nondestructive examination of the CNV pressure-retaining and integrally attached materials meet the requirements of ASME Code, Section III, Article NB-5000 and NF-5000 using examination methods of ASME Code Section V except as modified by NB and NF.

A non-destructive examination plan is prepared and implemented for the examinations to be performed to satisfy the fabrication and preservice examination requirements of ASME Code, Section III, Article NB-5000 and Article NF-5000, as applicable, and Section XI.

For CNV pressure boundary items defined as augmented inspections, preservice examinations are in accordance with ASME Code, Section III, Subsubarticle NB-5280 and ASME Section XI, Subarticle IWB-2200 using examination methods of ASME Code, Section V except as modified by NB-5111. These preservice examinations include essentially 100 percent of the pressure boundary welds. Final preservice examinations are performed after hydrostatic testing but before code stamping.

Section 6.2 describes inservice inspection of the CNV.

The design requirement to perform a CNV preservice design pressure leakage test is performed as specified in Section 6.2. The requirement of this test is to examine for visible leakage from CNV bolted flanged connections before the NPM is placed into service. Stress conditions as a result of this test are bounded by the hydrostatic conditions and no additional stress check or load combination is required to address this test. Fatigue cycles created by this test are included in the cycles alloted for the hydrostatic test.

Containment vessel flanges are tested in accordance with 10 CFR 50 Appendix J, Type B criteria. Each electrical penetration assembly is pressure tested periodically in accordance with 10 CFR 50 Appendix J, Type B criteria.

The Type B test pressure is the containment peak accident pressure. The leak rate is established by Containment Leakage rate Program.

Pneumatic testing at a pressure not to exceed 25 percent of design pressure may be applied before a hydrostatic test as a means of locating leaks, in accordance with ASME Code, Section III, Paragraph NB-6112.1(b).

Hydrostatic testing of the CNV is done in accordance with NB-6000. The CNV is pressurized using water to a minimum pressure of 1500 psig and a maximum pressure of 1590 psig, by measuring at the bottom of the CNV. Following a minimum time of 10 minutes at the hydrostatic test pressure, pressure is reduced to design pressure and held while examining for leaks.

If the CNV is hydrostatically tested with the RPV installed, both primary and secondary sides of the RPV are vented to the CNV to preclude a differential pressure external to the RPV greater than considered for design of the RPV.

The hydrostatic test procedure includes measures for sampling the test fluid (water) that contacts the CNV during hydrostatic testing.

Drain water is tested following hydrostatic testing for compliance with the purity requirements. The hydrostatic test procedure includes corrective actions to be taken (e.g., circulating flushes or fill and drains) in the event the exit fluid exceeds purity requirements.

Immediately following hydrostatic testing, the CNV is drained and dried by circulating air until the exit air dew-point temperature is less than 50 degrees F. The circulating air is oil free and does not contain combustion products from the

heating source. The temperature of the dry heated air is controlled to preclude damage to the steam generators (SGs) due to excessive differential temperature.

The shop hydrostatic tests of the CNV are witnessed by authorized personnel.

No leakage indications at the examination pressure are acceptable.

3.8.2.8 References

3.8.2-1 U.S. Nuclear Regulatory Commission, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials," (Draft Report for Comment) NUREG/CR-6909.

3.8.2-2 U.S. Nuclear Regulatory Commission, "Containment Integrity Research at Sandia National Laboratories - An Overview," NUREG/CR-6906, July 2006.

3.8.2-3 American National Standards Institute, "Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10000 Pounds (4500 kg) or More," ANSI N14.6-1993, LaGrange Park, IL.

3.8.2-4 NuScale Power, LLC., "NuScale Containment Leakage Integrity Assurance", TR-123952, Revision 0.

3.8.2-5 American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code*, 2017 Edition, no addenda, Section III, “Rules for Construction of Nuclear Facility Components,” Division 1 – Subsection NB, Class 1 Components, July 1, 2013.

3.8.2-6 NuScale Power, LLC, "Containment Vessel Ultimate Pressure Integrity," TR-121516, Revision 0.

3.8.2-7 NuScale Power, LLC, "NuScale Power Module Short-Term Transient Analysis", TR-121517, Revision 0.

Table 3.8.2-1: Design and Operating Parameters

Parameter	Value
Upper vessel diameter (approximate)	177 in.
Lower vessel diameter (approximate)	135 in.
Height from support base to crown of CNV top head cover (top auxiliary mechanical access structure not included) (approximate)	76 ft
Bottom of RXB elevation (reactor pool floor)	26 ft
Top of CNV elevation (approximate)	101 ft
Design internal pressure(1)	1,200 psia
Design temperature	CNV: 600 °F Support Skirt: 212 °F
External design pressure(1)	50 psia(2)
Normal operating internal pressure (nominal)	< 1 psia
Normal operating external pressure (nominal)	50 psia(2)
Normal operating temperature (nominal)	100 °F
Materials	Table 6.1-1 and Table 6.1-2.

Notes:

1) The internal and external pressures are not coupled. Internal pressures are evaluated with external pressure at 0 psia, and external pressures are evaluated with internal pressure at 0 psia. For example, the external design pressure is 50 psia with 0 psia internal pressure.
2) Includes a reactor pool water static head pressure. The external design pressure is also used for normal operating conditions.

Table 3.8.2-2: Load Combinations for Containment Vessel and Head ASME BPVC Stress Analysis

Plant Event	Service Level	Load Combination	Allowable Limit
Design	Design	P + DW + B	Design
CNV Hydrostatic Test	Test	P + DW + B	Test
Normal Operating Transients	A	P + DW + B + TH	A
Refueling	N/A	DW + B + RF	B
Continued Operating Transients	B	P + DW + B + TH	B
		P + DW + B + TH + ECCS	B
		P + DW + B + TH ± (OBE or SSE)	B
Design Basis Pipe Break	C	P + DW + B + TH ± MAX(DBPB, ECCS, RSV)	C
SG Tube Rupture		P + DW + B + TH	C
Rod Ejection Accident	D	P + DW + B + TH	C
MSPB and FWPB		P + DW + B + TH + MAX(MSPB,FWPB)	D
Pipe Breaks + SSE		P + DW + B + TH ± SRSS[SSE,MAX(DBPB,ECCS,RSV)]	D

Table 3.8.2-3: Load Combinations for Containment Vessel Bolt ASME BPVC Stress Analysis

Plant Event	Service Level	Load Combination	Allowable Limit
Design	Design	P + DW + B + M ± DFL	Design
CNV Hydrostatic Test	Test	P + DW + B + M	Test
Normal Operating Transients	A	P + DW + B + TH + M ± DFL	A
Refueling	N/A	DW + B + RF	B
Continued Operating Transients	B	P + DW + B + TH + M ± DFL	B
		P + DW + B + TH + M ± DFL + ECCS	B
		P + DW + B + TH + M ± DFL ± (OBE or SSE)	B
Design Basis Pipe Break	C	P + DW + B + TH + M ± DFL± MAX(DBPB, ECCS, RSV)	C
SG Tube Rupture		P + DW + B + TH + M ± DFL	C
Rod Ejection Accident	D	P + DW + B + TH + M ± DFL	C
MSPB and FWPB		P + DW + B + TH + M ± DFL + MAX(MSPB,FWPB)	D
Pipe Breaks + SSE		P + DW + B + TH + M ± DFL ± SRSS[SSE,MAX(DBPB,ECCS,RSV)]	D

Table 3.8.2-4: Load Combinations for Containment Vessel Bolted Connections ASME BPVC Stress Analysis

Plant Event	Service Level	Load Combination	Allowable Limit
Design	Design	P + DW + B + PRE + GS	Design
CNV Hydrostatic Test	Test	P + DW + B + PRE + GS	Test
Normal Operating Transients	A	P + DW + B + PRE + GS + TH	A
Refueling	N/A	DW + B + PRE + GS + RF	B
Continued Operating Transients	B	P + DW + B + PRE + GS + TH	B
		P + DW + B + PRE + GS + TH + ECCS	B
		P + DW + B + PRE + GS + TH ± (OBE or SSE)	B
Design Basis Pipe Break	C	P + DW + B + PRE + GS + TH ± MAX(DBPB, ECCS, RSV)	C
SG Tube Rupture		P + DW + B + PRE + GS + TH	C
Rod Ejection Accident	D	P + DW + B + PRE + GS + TH	C
MSPB and FWPB		P + DW + B + PRE + GS + TH +MAX(MSPB,FWPB)	D
Pipe Breaks + SSE		P + DW + B + PRE + GS + TH ± SRSS[SSE,MAX(DBPB,ECCS,RSV)]	D

Table 3.8.2-5: Load Combinations for Class 1 and Class 2 Supports ASME Stress Analysis

Plant Event	Service Level	Load Combination	Allowable Limit
Design	Design	DW + PRE + M ± DFL + TAM	Design
Normal Operating Transients	A	DW + PRE + M ± DFL+ TAM	A
Refueling	N/A	DW + LL+ PRE + M + RF + TAM	B
Continued Operating Transients	B	DW + PRE + M ± DFL + TAM	B
		DW + PRE + M ± DFL + TAM + ECCS	B
		DW + PRE + M ± DFL + TAM ± (OBE or SSE)	B
Design Basis Pipe Break	C	DW + PRE + M ± DFL + TAM ± MAX(DBPB,ECCS, RSV)	C
SG Tube Rupture		DW + PRE + M ± DFL + TAM	C
Rod Ejection Accident	D	DW + PRE + M ± DFL + TAM	C
MSPB and FWPB		DW + PRE + M ± DFL + TAM + MAX(MSPB, FWPB)	D
Pipe Breaks + SSE		DW + PRE + M ± DFL +TAM ± SRSS[SSE,MAX(DBPB,ECCS,RSV)]	D

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containments

The NuScale Power Module does not use internal structures (compartments, pedestals, or walls). Section 3.8.2 discusses connections between the containment vessel and the reactor vessel.

3.8.4 Other Seismic Category I Structures

The Seismic Category I structures are portions of the Reactor Building (RXB) and portions of the Control Building (CRB). These buildings are site-independent and designed for the certified seismic design response spectra (CSDRS) and the certified seismic design response criteria - high frequency (CSDRS-HF) described in Section 3.7.1. The static and seismic analyses are performed using ANSYS (Reference 3.8.4-1). Software used for performing seismic analysis of SC-I SSCs conforms with the requirements for computer software as per the NuScale Quality Assurance Program Description (QAPD) (Reference 3.7.1-12).

3.8.4.1 Description of the Structures

3.8.4.1.1 Reactor Building

The RXB consists of reinforced concrete (RC) basemat and slabs and steel-plate composite (SC) walls, and is designed to withstand the effects of natural phenomena (earthquake, rain, snow, wind, tornado, hurricane) without affecting operability of the safety-related structures, systems, and components (SSC) in the building. The overall dimensions of the building are 231.5 ft, 155.5 ft, 171 ft in east-west, north-south, and vertical directions. Section 1.2 provides figures showing the conceptual site layout, including location of the RXB at the site, and elevation and section view drawings for the building. Section 3.7.2 provides additional figures from the finite element model used for RXB analysis and design.

A description of the foundation is provided in Section 3.8.5. Building models are developed to include this change in basemat thickness. Floor slabs are 24-36 inches thick. Floor slab at elevation 100 ft includes 3 ft-deep T-beams in the east-west span. The roof is a composite section consisting of a concrete slab and steel girders. The building also includes steel-plate composite walls with various thicknesses in east-west and north-south directions.

The ground floor or grade level top of concrete (TOC) is elevation 100 ft. The bottom of the foundation concrete is typically elevation 17 ft. There are some portions that extend deeper. Actual site grade is approximately six inches below baseline TOC and sloped away from the structures. However, the terms "grade" and "site grade" refer to elevation 100 ft. The embedment of the RXB is approximately 83 ft.

The predominant feature of the RXB is the ultimate heat sink pool (Section 9.2.5). This pool consists of the spent fuel pool, refueling pool, and the reactor pool. This large pool occupies the center of the building. The reactor pool bays house up to six NuScale Power Modules (NPMs). The structural analysis assumes all six NPMs are in their respective bays. Section 3.7.2 describes a study of the dynamic effects of an earthquake that occurs when operating with fewer than six modules.

Structural views showing the primary walls on each floor are provided in Figure 3.7.2-60 through Figure 3.7.2-87.

3.8.4.1.2 Control Building

The CRB is a reinforced concrete building and its primary function is to house the main control room and the Technical Support Center. Section 1.2 provides figures showing the conceptual site layout, including location of the CRB at the site, and elevation and section view drawings for the building. Section 3.7.2 provides additional figures from the finite element model used for CRB analysis and design.

Structural view showing the primary walls on each floor is provided in Figure 3.7.2-11 through Figure 3.7.2-16.

3.8.4.1.3 Radioactive Waste Building

The RWB is a reinforced concrete building and its primary function is to house nonsafety-related SSC. Below grade portions of the building are classified as RW-IIa structure. The above-grade portion of the building is a steel-framed Seismic Category III (SC-III) structure with the exception of an RW-IIa isolated enclosure used for waste sorting and storage. Section 1.2 provides figures showing the conceptual site layout, including location of the RWB at the site, and elevation and section view drawings for the building.

3.8.4.1.4 Reactor Building Components

3.8.4.1.4.1 Bioshields

The bioshields consist of two separate pieces: a horizontal concrete slab and a vertical steel frame. Section 3.7.3 discusses the bioshields. The horizontal component is comprised of a thick reinforced concrete slab encased by a stainless steel liner. The vertical component is comprised of a square stainless steel tube framing system, radiation paneling, and a W-shape support member.

3.8.4.1.4.2 Reactor Building Pool Liner

The pool liner is designed to accommodate effects of and be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents per GDC 4 requirements. The RXB pools are the ultimate heat sink for the NPMs. The ultimate heat sink is discussed in Section 9.2.5. The pool liner's main function is to prevent potential pool inventory leakage. The liner is stainless steel. The RXB analysis includes the liner as a dead weight. The liner plate and its supporting components are Seismic Category I. The pool leakage detection system is discussed in Section 9.1.3.

3.8.4.1.4.3 Reactor Building Equipment Door

The Reactor Building Equipment Door (RBED) provides a physical barrier between RXB and RWB and access for importing and exporting large

equipment into the RXB including NPMs. Section 19.5 describes the RXB equipment door.

The weight of the 725,000 lb RXB equipment door is included in the RXB model.

3.8.4.1.4.4 Dry Dock Gate

The dry dock gate (DDG) is a large swing-type steel gate which separates or allows passage between the dry dock and the refueling pool (RFP). The structure of the DDG is a welded steel frame which is skinned with steel plate. The DDG and its components are designed to Seismic Category II requirements. It is analyzed for the worst-case load combination, a scenario of an empty dry dock with seismic and hydrodynamic loading. The weight of the 176,110 lb dry dock pool gate is included in the RXB model.

3.8.4.1.4.5 Over-Pressurization Vents

Over-pressurization vents (OPV) support the RXB by providing means of venting the atmosphere of the RXB to prevent over-pressurization within the RXB. The pool area rupture disk and vent, CVC pipe chase blow-off panels, steam gallery blow-off panels, CVC HX room always-open vents, and MHS Equipment room always-open vent are the safety-related components in the OPV.

3.8.4.1.5 Other Structures

3.8.4.1.5.1 Fuel Storage Racks

The design and analysis for the fuel storage racks are described in Section 9.1

3.8.4.1.5.2 Fuel Handling Machine

Design aspects of the fuel handling machine are described in Section 9.1.4.

3.8.4.1.5.3 Reactor Building Crane

The Reactor Building crane (RBC) is a bridge crane that rides on rails anchored to the RXB at elevation 145 ft 6 inches. The RBC is part of the overhead heavy load handling system and is discussed in Section 9.1.5. For analysis of the RXB, the RBC is included as a beam, mass, and spring model as described in Section 3.7.2.

3.8.4.1.5.4 Reactor Flange Tool

The reactor flange tool (RFT) is used with the RBC to support and position the NPM during disassembly and reassembly of the reactor pressure

vessel during refueling, maintenance, and inservice inspection activities. The RFT stores and supports the cask during cask loading.

The RFT is shared among all six NPMs.

The RFT supports the lower portion of the reactor vessel, containing the core, during refueling operations.

Items, including welds, in the RFT that constitute the load path for support of the NPM or are credited for seismic restraint are classified as Seismic Category I. Non-load-bearing items that do not provide a seismic restraint function are classified as Seismic Category III.

COL Item 3.8-1: An applicant that references the NuScale Power Plant US460 standard design will provide the design of the reactor flange tool.

3.8.4.1.6 Platforms and Miscellaneous Structures

The RXB and CRB use platforms and miscellaneous structures (e.g., ladders, guard rails, stairs). These components are constructed of steel beams, angles, channels, tubing, and grating. Platforms and miscellaneous structures may be Seismic Category I, II, or III depending on their safety function and potential interaction with Seismic Category I SSC. The seismic analysis includes these SSC as part of the standard floor load.

3.8.4.1.7 Buried Conduit and Duct Banks

The design has safety-related buried duct bank that goes from the RXB to the CRB. Section 3.7.3 provides additional details for buried SC-I piping and conduits.

3.8.4.1.8 Buried Pipe and Pipe Ducts

The design does not include buried safety-related pipes or pipe ducts.

3.8.4.1.9 Masonry Walls

Non-structural and non SC-I masonry walls are used as partition walls in the Reactor Building and in the Control Building. The mass of these walls are included as part of the standard floor load.

3.8.4.1.10 Modular Construction

Modular construction techniques are used extensively in the nuclear industry. The design of the Seismic Category I RXB structural walls includes steel-plate composite modular subsystems. Steel-plate composite walls allow for off-site fabrication, eliminating the need for construction formwork and reinforcing bars (rebar), while meeting strength, stability, and seismic requirements, thereby increasing the efficiency and productivity of construction. Modular

construction techniques (including sacrificial steel) that do not alter the design, normal construction techniques, or analysis are acceptable.

3.8.4.2 Applicable Codes, Standards, and Specifications

Codes, standards, and specifications meet the Design Specific Review Standard (DSRS) Section 3.8.4 acceptance criteria. In addition, Section 3.8.4 relies on independent standards to cover the design, materials, fabrication, testing, and inspections of Seismic Category I structures and basemats.

Regulatory guides applicable to design and construction of the Seismic Category I portions of the RXB and CRB are discussed in Section 1.9. Table 3.8.4-4 presents the seismic categories and design codes for structures.

For American Society of Testing and Materials (ASTM) standards, the design uses the latest endorsed version at the time of construction.

3.8.4.3 Loads and Load Combinations

The concrete and steel load combinations to be considered for structural design and analysis of the Seismic Category I portions of the RXB and CRB are based on American Concrete Institute (ACI) 349 (Reference 3.8.4-2) and ANSI/AISC-N690 (Reference 3.8.4-3). Load combinations for the analysis are shown in Table 6-2 and Table 8-2 of TR-0920-71621-P-A, Revision 1 (Reference 3.8.4-6).

Symbols for the design loads are listed below and discussed in the following sections.

- D = Dead loads, including piping, equipment, and partitions
- F = Loads due to weight and pressures of fluids
- H = Loads due to weight and static pressure of soil, water in soil, or other materials
- L = Live loads due to occupancy and moveable equipment
- L_r = Roof live load
- R_o = Piping and equipment reaction loads
- R_a = Piping and equipment reaction loads due to a postulated accident
- T_o = Thermal loads due to normal operating temperatures
- T_a = Thermal loads due to accident condition temperatures
- R = Rain load
- S = Snow load
- S_e = Extreme snow load
- W = Straight line wind load
- W_t = Loads due to the design basis tornado

- W_h = Loads due to the design basis hurricane
- E_o = Seismic load due to an operating basis earthquake
- E_{ss} = Seismic load due to a Safe Shutdown Earthquake (SSE)
- C_{cr} = Loads due to the Reactor Building crane
- P_a = Pressure loads due to accident conditions
- Y_j = Jet impingement load generated by a postulated high energy line break
- Y_r = Loads due the impact from a postulated high energy line break
- Y_m = Missile impact load, or related internal moments and forces, due to a high energy line break

3.8.4.3.1 Dead Loads (D)

Dead loads in the RXB consist of the self-weight of the structure, such as the walls, roof, and slabs, and other large permanent loads. This loading includes the weight of the pool water, the NPMs, the bioshields, and the RBC. In addition, equipment weights for large components are estimated and included as concentrated point loads (or if several pieces of equipment are located closely together, an equivalent uniform load is applied over the respective area).

Dead loads in the CRB consist of the self-weight of the structure, such as the walls, roof, slabs, steel beams, and columns. Other dead loads are the main control room and the control room habitability system. In addition, weights for large equipment and components are estimated and included as concentrated point loads (or if several pieces of equipment are located closely together, an equivalent uniform load is applied over the respective area).

3.8.4.3.1.1 Concrete Self-Weight

The concrete self-weight is obtained by multiplying the volume of concrete for each structural element in the building by the reinforced concrete density of 150 pcf.

3.8.4.3.1.2 NuScale Power Module Weight

The NPMs are included in the seismic model of the RXB. Discussion of the RXB seismic model is in Section 3.7.2.

3.8.4.3.1.3 Liner Plate Dead Weight

The reactor pool is surrounded by steel-composite walls. The weight of these walls is included in the modeled steel-plate composite wall weight. The bottom of the pool is lined with steel plate. This load is applied to the reactor pool floor.

3.8.4.3.1.4 Bioshield Weight

The bioshields are included in the model.

3.8.4.3.1.5 Reactor Building Crane Weight

The RBC is a bridge crane on a crane rail. The RBC is included as a beam and spring model.

3.8.4.3.1.6 Fuel Storage Rack Weight

The weight of the fuel storage racks are included in the model, fully loaded with fuel assemblies.

3.8.4.3.1.7 Module Assembly Equipment Weight

The module assembly equipment facilitates delivery of the NPMs to the reactor pool.

Self Propelled Mobile Transporters (SPMTs) move the NPM from the assembly building to the RXB. The SPMTs distribute the transported load over multiple wheels and do not require rails for guidance or support.

3.8.4.3.1.8 Stair and Elevator Weight

The stairs and elevator are components that are considered in the analysis. The individual stair components considered in the stairwell dead weight calculation include the main stair structure, landing area, stair stringer, stair pan treads, and railing.

3.8.4.3.1.9 Equipment Weights

Table 3.8.4-1 summarizes RXB equipment weights per floor. The NPM, bioshields, and RBC are not included in the per-floor summary because these components are applied in the analysis model as described above. The majority of equipment loads are applied as either concentrated nodal masses or uniformly distributed masses.

Table 3.8.4-2 summarizes the CRB equipment weights per floor. These loads are applied to the CRB model as concentrated nodal masses.

3.8.4.3.1.10 Uniform Equivalent Dead Load

The uniform equivalent dead load for the RXB and CRB accounts for pieces of equipment weighing less than 1000 lbs not accounted for in equipment dead loads and for cable trays, piping, and ducts. The RXB and CRB floors are designed using a uniform equivalent dead load of 50 psf. The equivalent dead load and the weight of architectural finishes are 50 psf for both the RXB and CRB roofs.

3.8.4.3.1.11 Reactor Flange Tool Weight

The reactor flange tool (RFT) is included in the RXB model as an equipment weight, detailed in Section 3.8.4.3.1.10. The RFT weighs 210,000 pounds.

3.8.4.3.2 Liquid Loads (F)

The liquid load consists of the water pressure exerted on the walls in the reactor pool, refueling pool, spent fuel pool, and dry dock during static and seismic conditions. The water is modeled by FLUID30 elements with a material density that is activated by applying acceleration. In reactor bays, the volume occupied by the NPMs is subtracted from the pool volume. The CRB does not have liquid loads.

3.8.4.3.3 Earth Pressure (H)

The embedded exterior walls of the buildings are subjected to lateral soil pressure loads induced by two types of loads as described below:

- Static Soil Pressure - induced by the weight of soil, hydrostatic pressure, and a surcharge load at grade level.
- Soil-Structure-Interaction Dynamic Soil Pressure - soil pressure determined from the double building analysis using the RXB and RWB.

For the static soil pressure, the lateral soil pressure is calculated assuming that the soil is completely confined and cannot move. The soil is also considered to be submerged for the total embedment depth because the water table is close to grade level. Therefore, the total horizontal pressure from the submerged soil is calculated as the sum of the hydrostatic pressure and the lateral soil pressure considering the buoyant effect. Because the water provides a buoyant effect, the effective pressure is calculated using the difference between the soil density and water density. For the RXB, the embedment depth used in the mathematical model is 83 ft.

The buoyant force is the upward pressure exerted on the bottom of the foundation during a saturated condition. It is the equivalent weight of the water that would otherwise occupy the below grade volume of the structure. The buoyant force is equal to the volume of the building below grade multiplied by the density of water.

Maximum Hydrostatic Pressure

γ_ω = 62.4 pcf unit weight of water, H = 83’ ft embedment depth, u = γ_ω
H = 5179 psf

Effective Lateral Pressure

γ_{sat} = 130 pcf unit weight of saturated soil, $\gamma_b = \gamma_{sat} - \gamma_\omega$ = 67.6 pcf buoyant unit weight

H = 83 ft embedment depth, K_o = 0.5 coefficient of pressure at rest, $p_{he} = K_o \gamma_b H$ = 2805 psf

Surcharge Loads

p_q= 250 psf surcharge load, $p_{hq} = K_o\, p_q$ = 125 psf

Total Maximum Lateral Soil Pressure

The total maximum lateral soil pressure at a depth H is the sum of the hydrostatic pressure, the effective lateral pressure, and the surcharge lateral pressures calculated above.

$p_h = u + p_{he} + p_{hq}$ = 8110 psf

3.8.4.3.4 Live Loads (L)

Live loads are a non-permanent weight based on the maximum loads expected by the use and occupancy of the structure. The RXB live loads include floor area loads, lay down loads, fuel transfer casks and equipment handling loads, and similar items.

The RXB uses a base live load of 150 psf. Electrical, telecom, HVAC equipment rooms, stairways, mezzanines, walkways, laboratories, and laundry rooms have a live load of 100 psf instead. The floor live loads are not applied on areas occupied by equipment, whose weight is specifically included as a uniform equipment load or a significant concentrated equipment load.

The CRB uses a base live load of 100 psf. The main control room has 50 psf of live load.

3.8.4.3.5 Roof Live Loads (L_r)

A load of 50 psf is used for the roof live load of both structures.

3.8.4.3.6 Pipe and Equipment Reactions (R_o)

Pipe reactions during normal operation or shutdown conditions are based on the most critical transient or steady state condition.

The CRB does not have high-energy piping; R_o is not a load for the CRB.

3.8.4.3.7 Accident Pipe and Equipment Reactions (R_a)

Pipe and equipment reactions under thermal conditions are generated by the postulated pipe break, including (R_a). This reaction includes their dead load, live load, thermal load, seismic load, thrust load, and transient unbalanced internal pressure loads under abnormal or extreme environmental conditions.

The CRB does not have high-energy or high temperature piping; R_a is not a load for the CRB.

3.8.4.3.8 Operating Thermal Loads (T_o)

Thermal loads are caused by a temperature variation through the walls between the interior temperature and the external environmental temperature. In addition, in the RXB, a thermal gradient could occur in the walls surrounding the reactor pool. For RC members, Section 1.3 of ACI 349.1R (Reference 3.8.4-4) states that thermal gradients should be considered in the design of reinforcement for normal conditions to control concrete cracking. However, a thermal gradient less than approximately 100 degrees F need not be analyzed because such gradients do not cause significant stress in the reinforcement or strength deterioration. The walls of RXB are SC walls. For SC walls, AISC N690 states that for operating thermal conditions, the stiffness of the faceplates and the cracked concrete infill without additional cracking due to thermal effects is adequate to represent the stiffness of SC walls as thermal gradients due to operating temperatures are small and they develop over significant time. Per AISC N690, this effective stiffness is used to calculate the thermal demands that are included in required design strengths.

As shown in Table 2.0-1, the external temperature site parameters for the NuScale standard structures are zero percent exceedance dry bulb values of -40 degrees F and +115 degrees F. The external surface soil temperature is assumed to be 21 degrees F in the winter and 92 degrees F in the summer. These temperatures vary with soil depth down to an elevation of 50 feet, where a constant 46 degrees F is held in the winter and 74 degrees F in the summer.

The RXB has a design internal air temperature of 70 degrees F, and a design pool temperature range of 65 degrees F to 120 degrees F. These temperatures are used to determine the stresses and displacements.

The CRB has a maximum temperature differential of 118 degrees F, based on an external temperature of -40 degrees F and an internal temperature of 78 degrees F. This gradient is determined not to affect the design stresses in the building. T_O is not a load for the CRB.

3.8.4.3.9 Accident Thermal Loads (T_a)

The maximum post accident temperature in the RXB is assumed to be 212 degrees F above the pool water level. The boiling temperature of water

under 50ft of water is 262 degrees F.. This temperature is used in conjunction with the external temperature to determine the stresses and displacements.

The CRB does not have high-energy or high-temperature piping; T_a is not a load for the CRB.

3.8.4.3.10 Rain Load (R)

Both the RXB and CRB roofs have drainage systems in place to limit the average water depth. Therefore, a rain load is assumed bounded by the snow load and extreme snow load.

3.8.4.3.11 Snow Loads (S)

As shown in Table 2.0-1, a roof snow load of 50 psf is assumed for normal load combinations. Equation 3.8-1 (taken from Equation 7-1 of Reference 3.8.4-5) is used to convert from ground-level snow loads to roof snow loads. An exposure factor of 1.0 is used. A thermal factor of 1.0 is used. An importance factor of 1.2 is used for Seismic Category I buildings (Section 3.2) and an importance factor of 1.0 is used for the other buildings.

$$p_f = 0.7 C_e C_t I p_g \qquad \text{Eq. 3.8-1}$$

where,

p_f is the roof snow load,

C_e is the exposure factor,

C_t is the thermal factor,

I is the importance factor, and

p_g is the ground snow load.

3.8.4.3.12 Extreme Snow Loads (S_e)

A wet roof snow load of 75 psf is assumed for extreme environmental load combinations. Extreme ground-level snow loads are converted to extreme roof snow loads using Equation 3.8-1 in the same manner described in Section 3.8.4.3.11.

3.8.4.3.13 Wind Loads (W)

The design wind load pressure on the RXB is 154 psf. This load is 137 psf for the CRB. Wind loads are developed as described in Section 3.3 based on the site parameters in Table 2.0-1.

3.8.4.3.14 Tornado Wind Loads (W_t) and Hurricane Wind Loads (W_h)

Tornado wind loads and hurricane wind loads are developed as described in Section 3.3 based on the site parameters in Table 2.0-1. The hurricane wind load pressure is 400 psf. The tornado wind case has two components: a design tornado wind load and a uniform differential pressure load. The differential pressure always acts away from the structure, whereas the design tornado wind load is applied in one of four configurations: north+east, north+west, south+east, or south+west. Thus, the differential pressure adds to the wind load on some faces and subtract on the opposite. The design tornado wind load has a magnitude of 224 psf and the differential pressure is 230 psf, though only half of it is applied.

The CRB hurricane wind load pressure is 230 psf. Tornado wind for the CRB is similar to the RXB, except the magnitude of the design tornado wind load is reduced to 199 psf.

3.8.4.3.15 Operating Basis Earthquake Seismic Loads (E_o)

The operating basis earthquake is defined as one-third of the SSE. Earthquake loads from the operating basis earthquake (E_o) are not evaluated.

3.8.4.3.16 Safe Shutdown Earthquake Seismic Loads (E_{ss})

The SSE for the site-independent evaluation of the RXB and CRB is the CSDRS and the CSDRS-HF using data from Table 2.0-1. The SSE seismic loads (E_{ss}) are derived from evaluation of the structures using ground motion accelerations from the CSDRS and the CSDRS-HF as described in Section 3.7.

Seismic dynamic analyses of the buildings consider 100 percent of the dead load and 25 percent of the floor live load during normal operation, and 75 percent of the roof snow load as the accelerated mass.

3.8.4.3.17 Crane Load (C_{cr})

Crane load comes from the RBC. The RBC is described in Section 9.1.5 and Section 3.7.3.

Crane live loads are used for design of the runways beams, connections, and crane supports. These crane live loads are due to the moving crane and include the maximum wheel load, vertical impact, lateral impact, and longitudinal impact loads.

The maximum wheel load for the RBC is produced by the weight of the bridge, plus the sum of the maximum lift capacity and the weight of the trolley positioned on its runway at the location where the resulting load effect is maximum. The hook and trolley are assumed to align with the crane wheel location. Therefore, the trolley and lift load are assumed to act 100 percent on

the ends. The bridge weight is distributed 50 percent to each end. There are 16 crane wheels at each end of the crane.

There are no large cranes in the CRB; C_{cr} is not a load for the CRB.

3.8.4.3.18 Accident Pressure Loads (P_a)

Accident pressure loads within a compartment or the entire building are due to differential pressure generated by a postulated pipe rupture, including dynamic effects due to pressure time-history. In the RXB an accident pressure of 6 psi is evaluated in the pool area to account for the energy release of a high-energy line break.

There are no accident pressure loads in the CRB; P_a is not a load for the CRB.

3.8.4.3.19 Jet Impingement Load (Y_j)

Jet impingement is a localized load on the structure due to the steam/water jet from a high energy line break.

There are no high energy lines in the CRB; Y_j is not a load for the CRB.

3.8.4.3.20 Pipe Break Reaction Loads (Y_r)

The pipe break reaction load is a localized load on the structure generated by the pipe hanger that is due to a high-energy line break.

There are no high-energy lines in the CRB; Y_r is not a load for the CRB.

3.8.4.3.21 Missile Impact Loads (Y_m)

The missile impact load is a localized load on the structure due to the whipping high-energy line or a missile from a high-energy line break. Internal missile loads, if they occur, are localized loads.

There are no high-energy lines in the CRB; Y_m is not a load for the CRB.

3.8.4.3.22 Other Loads

3.8.4.3.22.1 Construction Loads

Construction loads are loads from events and activities during construction. These loads are developed in accordance with Standard SEI/ASCE 37-02, “Design Loads on Structures During Construction.” Construction loads are not included when determining seismic loads.

3.8.4.3.22.2 Operation with Less than 6 NuScale Power Modules

The design allows for operation with fewer than six NPMs. The building analysis is performed with all six NPMs in place. However, a study is performed as described in Section 3.7.2 to evaluate the dynamic effects of an earthquake when operating with fewer than six NPMs. That study concluded that dynamic effects on the building with fewer than six modules installed would be similar to dynamic effects when all six modules are in place.

3.8.4.3.23 Turbine Missile Loads

Turbine missile loads are developed and defined in Section 3.5.

3.8.4.4 Structural Modeling and Analysis Procedures

The RXB and CRB models described in Section 3.8.4.1 are analyzed for the static loads listed in Section 3.8.4.3. The models are connected to the surrounding soil using bonded contact. This soil provides the boundary conditions for the two models. Dead, live, and snow loads are accounted for by applying surface effect elements with specified additional masses. The ANSYS APDL command "CMACEL" is used to activate these masses and create the desired load. Some additional dead loads related to the soil and surcharge are applied as pressure. Earth pressure is accounted for by pressure loads specified with a gradient such that the magnitude increases with depth. Wind, tornado, hurricane, and missile impact loadings are applied as surface loads on elements and point loads on nodes.

The RXB and CRB are considered in both uncracked and hybrid cracked cases. Appendix 3B describes the procedure for deriving the hybrid cracked states. Wind, tornado, hurricane, and windborne missile impact loads are considered only for the uncracked case, as these are not combined with seismic loads.

3.8.4.4.1 Reactor Building Analysis

ANSYS Model of the Reactor Building

The RXB consists mostly of SOLSH190 elements, 8-noded structural solid shell elements capable of analyzing the relatively thick wall and slab sections. The roof is modeled with SHELL181 elements. The FLUID30 elements represent the RXB pool and are activated for the hydrostatic load case.

SOLID185 elements, which are 8-noded structural solid elements, make up the backfill. The outside faces of the backfill are covered in CONTA174 and TARGE170 surface elements, which connect to a larger soil superelement represented by a MATRIX50 element type. Additionally, SURF154 and MASS21 element types are used to apply additional mass to surfaces and points, respectively.

A reference coordinate system is located 12 in. below the bottom of the RXB basemat, centered in the north-south direction and 45 in. west of the westernmost wall. The X direction points east, Y points north, and Z is vertically upwards. Figure 3.7.2-60 provides a view of the RXB ANSYS model, and Figure 3.7.2-61 provides the section views.

3.8.4.4.1.1 ANSYS Model for Thermal and Pressurization Analysis

For the thermal analysis of the RXB steel-plate composite and reinforced concrete walls and slabs, SOLSH190 elements are replaced with ANSYS 3D thermal solid elements (SOLID70). This element has eight nodes with a single-degree-of-freedom, temperature, at each node. The SHELL181 elements are replaced with ANSYS 3D thermal shell elements (SHELL131). Note, the section properties (multiple-layer material properties and layer thickness) of SOLSH190 elements do not work for SOLID70 elements. However, the structural material properties have no impact on the thermal analysis.

Thermal analyses are performed for the combination of lowest soil and outside air temperature and highest pool water and inside air temperatures. This combination results in the largest through-wall temperature gradient.

Steady-state thermal analysis is performed for both normal operation and accident conditions. Since the thermal analysis models have only a single element through the thickness, it is useful only for representing steady state temperature distributions in the walls and slabs. In this case the temperature distribution through walls and slabs is essentially linear except near locations of discontinuity in thermal boundary conditions or near the perimeter of the wall or slab.

Thermal loads in the thermal analysis are:

- Convections or surface thermal loads applied at the exterior and interior wall and roof surfaces (bulk temperatures correspond to the outside and inside air temperatures).
- Constant temperature boundaries (soil and pool temperatures given in Section 3.8.4.3.8).

Final temperatures from the thermal analysis are then applied as body force nodal loads in a subsequent structural analysis. Thermal strains are then calculated by ANSYS as shown below:

$eth = \alpha(T-T_{ref})$

Where:

eth = thermal strain

α = coefficient of thermal expansion

T = final temperature

T_{ref} = initial or reference temperature (70°F)

Thermal stresses are then obtained after a structural analysis of the model with the imposed thermal strains.

3.8.4.4.2 Control Building Analysis

ANSYS Model of the Control Building

The CRB is built with SHELL181 elements. MASS21 elements are used for the dead load case to represent discrete equipment loads. SURF154 elements are used to represent the dead load of the partition walls and beam decking as well as live and snow loads. Mass for SHELL181 elements is accounted for by applying material properties with the corresponding densities specified. For MASS21 and SURF154 elements, real constants are used to input applied mass, depending on the load case of interest.

The Seismic Category II (SC-II) portion of the CRB is represented by a simple SHELL181 slab with reaction point loads applied, depending on the load case of interest. This step is done to capture the influence that the SC-II portion has on the soil near the SC-I portion of the CRB.

TARGE170 and CONT174 elements connect the CRB basemats to the underlying SOLID185 soil elements. Options are set such that the shell thickness of the base slab is accounted for and contact is specified to be in the bonded state.

A reference coordinate system is located at the southwest corner of the CRB at grade level (elevation 100 ft). Thus, the X direction points east, Y points north, and Z is vertically upwards. Figure 3.8.4-1 and Figure 3.8.4-2 provide isometric and perspective views of the CRB ANSYS model.

3.8.4.5 Structural Design and Acceptance Criteria

The structural design and acceptance criteria for the seismic category RXB and CRB are described in TR-0920-71621-P-A, Revision 1 (Reference 3.8.4-6).

Appendix 3B provides structural design evaluation results for selected sections of the RXB and CRB.

Section 3.8.5.6 identifies acceptance criteria applicable to additional basemat load combinations.

3.8.4.5.1 Design Summary Report

Appendix 3B documents that the Seismic Category I structures meet acceptance criteria presented inSection 3.7 and Section 3.8.

Deviations from the design are tracked as required by 10 CFR Part 50, Appendix B, and evaluated consistent with the methods and procedures of Section 3.7 and Section 3.8.

3.8.4.6 Materials, Quality Control and Special Construction Techniques

3.8.4.6.1 Materials

The principal construction materials for structures including steel-plate composite walls are concrete, reinforcing steel, structural steel, stainless steel, bolts, anchor bolts, and weld electrodes. Table 3.8.4-3 provides specifics of materials considered for structural design.

3.8.4.6.1.1 Concrete

Structural concrete used in the Seismic Category I portions of the RXB and of the CRB conforms to ACI 349 as supplemented by RG 1.142 and ACI 301. The structural concrete has compressive strength (f'c) of 5000 and 7000 psi.

Specific concrete mix is developed based on site conditions. Concrete mixes are designed in accordance with ACI 211.1, using materials qualified and accepted for this work. The mixes are based on field testing of trial mixtures with actual materials used. Concrete constituents conform to the following codes:

Cement

Cement conforms to ASTM C150.

Aggregates

Aggregates conform to ASTM C33.

ASTM Standards C1260 and C1293 are used in testing aggregates for potential alkali-silica reactivity. Low-alkali cement is used in concrete with aggregates that are potentially reactive.

Admixtures

Air-entraining admixtures, if used, conform to ASTM C260. Chemical admixtures, if used, conform to ASTM C494. Fly ash and pozzolan admixtures, if used, conform to ASTM C618.

Water

Water and ice for mixing is clean, with a total solids content of not more than 2000 ppm.

Construction

Construction, including placement, inspection, and testing is performed in accordance with the following codes and standards:

- ACI 301 Specifications for Structural Concrete for Buildings
- ACI 304R Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
- ACI 305.1 Specification for Hot-Weather Concreting
- ACI 306.1 Specification for Cold-Weather Concreting
- ACI 347 Recommended Practice for Concrete Formwork
- ACI SP-2 Manual of Concrete Inspection
- ASTM C94 Specification for Ready-Mixed Concrete

3.8.4.6.1.2 Reinforcing Steel

Reinforcing steel for major structures is deformed billet steel bars conforming to ASTM designation A615 grade 60 or ASTM A706 grade 60. The placement of concrete reinforcement is in accordance with ACI -349. Reference 3.8.4-6 summarizes requirements for designing reinforced concrete.

Reinforcing development length and splice length is calculated by formulas in ACI 349.

Welded wire fabric for concrete reinforcement conforms to ASTM A185 (plain wire) or A497 (deformed wire).

The standard plant design does not use coated reinforcing steel.

3.8.4.6.1.3 Steel-Plate Composite Walls

TR-0920-71621-P-A (Reference 3.8.4-6) summarizes requirements for designing steel-plate composite walls.

3.8.4.6.1.4 Connections

Steel Bolts and Studs

Bolts of type ASTM A307 with lock washers may be used for stairs, ladders, purlins, and girts only. Other bolted connections use high-strength bolts of ASTM A490 or A325 material.

Steel studs meet ASTM A108 and Structural Welding Code-Steel AWS D1.1/D1.1M.

Anchor Bolts

Anchor bolts are of type ASTM F1554, 36 ksi or 55 ksi yield strength material. Where higher strengths are required by ASTM F1554, 105 ksi yield strength materials are used.

If post-installed anchors are used for supports, the flexibility of base plates is accounted for in determining the anchor bolt loads. Post-installed anchors are also qualified for seismic loading if required.

Welds

Welding electrodes are E70XX unless otherwise noted on drawings or within specification for ASTM A36 steel and E308L-16 or equivalent for ASTM A240, type 304- L stainless steel.

3.8.4.6.1.5 Other

Grating

Grating is welded and galvanized steel, "Metal Bar Type," conforming to ANSI/NAAMM MBG 531-00 and ANSI/NAAMM MBG 532-00. Grating is stainless steel.

Masonry Walls

There are no safety-related reinforced masonry walls in Seismic Category I structures.

3.8.4.6.2 Quality Control

Chapter 17 addresses the Quality Assurance Program.

3.8.4.6.3 Special Construction Techniques

Modular construction, where wall or slab elements (or the rebar reinforcement) are pre-fabricated and then incorporated into the building, is used when possible. This process is expected to leave sacrificial (non-structural) steel in the buildings. Typically this steel is reinforcing beams underneath slabs. The uniform distributed dead load applied in the structural and seismic analyses encompasses the weight of this steel.

3.8.4.7 Testing and Inservice Inspection Requirements

There is no testing or inservice surveillance beyond the quality control tests performed during construction, which is in accordance with ACI 349 and AISC N690 (Reference 3.8.4-3).

COL Item 3.8-2: An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific program for monitoring and maintenance of the Seismic Category I structures in accordance with the requirements of 10 CFR 50.65 as discussed in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Monitoring is to include below grade walls, groundwater chemistry if needed, base settlements, and differential displacements.

3.8.4.8 Evaluation of Design for Site-Specific Acceptability

The RXB and CRB are designed to remain operable and to transmit forces, moments, and accelerations so that contained, safety-related SSC remain operable during and following an earthquake with a spectra equal to the CSDRS or the CSDRS-HF. Operability during and following the earthquake is accomplished by confirming the buildings meet code-acceptance criteria if situated on a soft soil site, a hard soil/soft rock site, a rock site, or a hard rock site. The entire analysis described in Section 3.8.4 does not need to be re-performed if it can be shown that non-seismic loads are less than those produced by the site parameters in Table 2.0-1 and that the forces experienced in the building from the site-specific earthquake are less than those produced from the CSDRS and CSDRS-HF.

The comparison of the non-seismic parameters is performed as described in Section 2.0. A direct comparison of seismic inputs cannot be made. Therefore, results of the site-specific seismic analysis are compared as described below.

The site-specific foundation input response spectra are compared to the CSDRS and CSDRS-HF (used as the foundation input response spectra for the site-independent analysis). This comparison demonstrates that the site-specific seismic input is bounded by the input used for design.

In-structure response spectra at 5 percent damping are used for comparison within the buildings, as discussed in Section 3.7.2. The design in-structure response spectra (ISRS) may be used as a surrogate for the forces and moments. If the site-independent ISRS are larger than the site-specific ISRS, the forces and moments are bounded for the design. The ISRS comparisons are done specifically at the NPM skirt supports and lug restraints to confirm that the forces and accelerations that the NPMs experience are acceptable. In addition, the ISRS at the RBC wheels are checked. The RBC is the only other of large, risk-significant SSC. As a general check of the buildings, the ISRS are compared at grade, bioshields, and at the roof of the RXB; and at the main control room, grade level, and the roof of the Seismic Category I portion of the CRB. This comparison is accomplished by confirming that site-specific characteristics/results are bounded by the US460 standard design parameters/results.

3.8.4.9 References

3.8.4-1 ANSYS (Release 19.2) [Computer Program]. (2019). Canonsburg, PA: ANSYS Incorporated.

3.8.4-2 American Concrete Institute, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," ACI 349-06, Farmington Hills, MI.

3.8.4-3 American National Standards Institute/American Institute of Steel Construction, "Specification for Safety-Related Steel Structures for Nuclear Facilities," ANSI/AISC N690-12, Chicago, IL.

3.8.4-4 American Concrete Institute, "Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures," ACI 349.1R-07, Farmington Hills, MI.

3.8.4-5 American Society of Civil Engineers/Structural Engineering Institute, "Minimum Design Loads for Buildings and Other Structures," ASCE/SEI 7-05, Reston, VA.

3.8.4-6 NuScale Power, LLC, “Building Design and Analysis Methodology for Safety-Related Structures,” TR-0920-71621-P-A, Revision 1.

3.8.4-7 NuScale Power, LLC Quality Assurance Program Description, Revision 0, MN-122626, November 2022.

Table 3.8.4-1: Summary of Reactor Building Equipment

Elevation	Total (kip)
EL. 55’-0”	1,190
EL. 55'-0" to EL 70'-0"	364
EL. 70'-0" to EL 85'-0”	395
EL. 85'-0" to EL 100'-0"	97
EL. 100'-0" to EL 126'-0"	2,886
EL. 126'-0" to EL. 145'-0" [1]	311
Total	**5,242**

Notes:

1. Equipment at elevation 145’-6” is included at elevation 145’-0”.

Table 3.8.4-2: Summary of Elevations of Control Building Equipment

Elevation	Total (kip)
EL. 100'-0"	699
EL. 123'-0"	4.4
EL. 145'-0"	2.9
Total	**706.3**

Table 3.8.4-3: Material Properties

Item	Value
Concrete Compressive Strength	f'_c = 5,000 psi typical f'_c = 7,000 psi for the RXB roof slab and floor slabs at EL 70', 126', and 146.4'.
Concrete Poisson's Ratio	v_c= 0.17
Concrete Density	ρ_c= 150 pcf
Concrete Coefficient of Thermal Expansion	α_c= 5.5 E-06
Reinforcing Steel Yield Strength	F_y = 60 ksi
Reinforcing Steel Modulus of Elasticity	E_s = 29,000 ksi

Table 3.8.4-4: Seismic Categories and Design Codes

Structure/Loading	Seismic Category I	Seismic Category II*
Concrete structures	ACI 349	ACI 349
Steel structures	AISC N690	AISC N690
Cold rolled members	AISI	AISI
Minimum design loads	ASCE 7	ASCE 7
Seismic analysis/design	ASCE 4	ASCE 4

*Seismic Category II SSC that are not part of a structure's primary vertical or horizontal load resisting system may be designed to the codes and standards of Seismic Category III SSC (ACI 318 and AISC 360). However, interaction of non-seismic Category I structures with Seismic Category I SSC shall be addressed as required by DSRS 3.7.2.

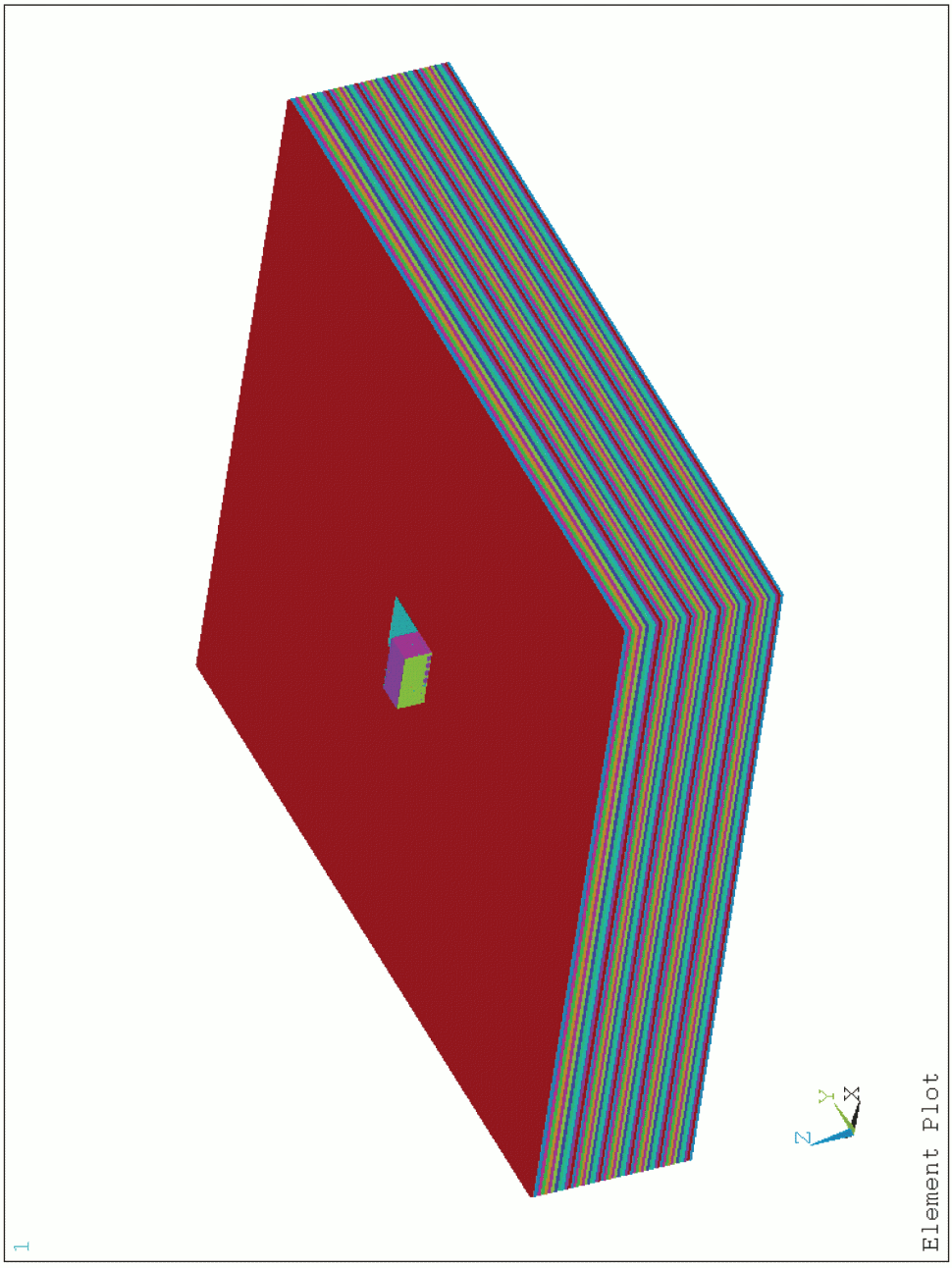


Figure 3.8.4-1: Control Building ANSYS Model With Soil

*Colors reflect material property numbers assigned to the various walls.



Figure 3.8.4-2: Control Building ANSYS Model Without Soil

*Colors reflect material property numbers assigned to the various walls, slabs, and roof.

3.8.5 Foundations

3.8.5.1 Description of Foundations

The Seismic Category I Buildings are portions of the Reactor Building (RXB) and portions of the Control Building (CRB). The CRB is located northwest of the RXB, and there is an underground ductbank between the two buildings. The Radioactive Waste Building (RWB) is classified as RW-IIa below grade and SC-III above grade with the exception of RW-IIa enclosure and is approximately 30 feet from the RXB and connected to it by three tunnels. The RXB, CRB, and RWB are described in Section 1.2 and Section 3.8.4. Foundations of the RXB and CRB are described below.

Reactor Building Foundation

The RXB basemat foundation is 96 in. thick. The bottom of the basemat elevation of the reactor building is located at 83 ft below grade (EL 17 ft). Under the pool area, the basemat has an additional 1.0 ft thick layer of concrete at the top surface with reinforcement to control shrinkage and cracking. This additional layer of basemat is considered to be non-structural and, hence, the basemat design is conservatively performed for a section thickness of 8.0 ft (96 in.). The basemat measures approximately 230 feet by 155 feet. The foundation top of concrete (TOC) elevation is 25 feet. The foundation for the refueling pool area has a top of concrete elevation of approximately 26 feet.

The basemat is modeled using solidshell (SOLSH190) elements. This element is useful for simulating shell structures with a wide range of thicknesses (from thin to moderately thick).

Control Building Foundation

The CRB is a reinforced concrete (RC) structure with RC walls. Dimensions of the SC-I portion of the building are approximately 120 ft, 55 ft, and 50 ft in east-west (X), north-south (Y), and vertical (Z) directions, respectively.

The building consists of a 5 ft thick basemat. Concrete elements are modeled using shell (SHELL181) elements.

3.8.5.2 Applicable Codes, Standards and Specifications

Codes, standards, and specifications used to design and construct the RXB and CRB are identified in Section 3.8.4. These codes are applicable to the foundations as well.

3.8.5.3 Design and Analysis Procedures

3.8.5.3.1 Reactor Building Stability Analysis Model Description

ASCE 43-19 states that deeply embedded structures with a center of gravity below the site grade elevation do not need to demonstrate sliding or

overturning stability. The RXB has a center of gravity more than 20 ft below the site grade elevation and, thus, does not need to be checked for these cases. However, the calculation is performed for completeness.

A static force equilibrium method is used to determine the factor of safety (FOS) for the RXB against overturning, sliding, and uplift. For each of these cases, a free body diagram is developed illustrating forces resisting and potentially causing the instability. Equations defining the resisting forces are developed. General parameters of the RXB structure are presented in Table 3.8.5-2a and used to numerically quantify the resisting forces.

Results from the RXB portion of the double building harmonic analysis are post processed. From this analysis, transfer functions in three directions for each soil type of interest are extracted, interpolated, and convolved with input seismic motions to retrieve a time history of resultant forces and moments at the center of the RXB basemat. These time histories are then used to form the basis of the demand forces and are compared against resisting forces to establish the FOS.

Although the deeply embedded RXB does not need to be checked for sliding and overturning instability, the fraction of passive and active pressure that is required to reach an acceptable FOS of 1.1 is still determined. Full mobilization of these pressures require a large amount of soil displacement, but by setting a target FOS and calculating the percentage of the passive and active pressures required to reach that level, it is shown that a small fraction is sufficient. To be conservative and simplify the problem, a single factor is calculated and applied to both the passive and active pressures simultaneously.

When the FOS are greater than or equal to 1.1, the RXB is deemed stable with 8 percent of ultimate passive and active pressure is activated. The FOS is found to be acceptable against instability and thus nonlinear analysis is not required. Since the passive earth pressure coefficient, K_p, is found to be smaller than the at-rest earth pressure coefficient, K_0, the embedded walls do not need to be designed for passive pressure.

3.8.5.3.2 Control Building Stability Analysis Model Description

The uplift, sliding, and overturning stability analysis of the CRB is performed in two steps. The first step is based on linear elastic analyses and the second on nonlinear analysis discussed in Section 3.8.5.3.3.

In the first step, a force equilibrium method is employed to determine the FOS by comparing the driving and resisting forces for overturning, sliding, and uplift conditions. Results from the CRB harmonic analysis (Section 3.7.2) are post processed. The FOS values are calculated for the load combinations of $D+F'+W_h$ and $D+F'+E_s$ where D is the dead load, F' is the buoyancy, W_h is the hurricane load, and E_s is the safe-shutdown earthquake load as further detailed in Section 3.8.5.4. For the first load combination, the hurricane load is

selected instead of wind or tornado. Hurricane loads have the highest amplitude of the applied external pressure.

In both analyzed load combinations, the dead load and the friction between the base of the model and the soil act as the resisting forces. The static coefficient of friction is 0.58, as described in Section 3.8.5.4.3. Since the base of the CRB model is located 5 ft below grade, contribution of the soil around the basemat to the stability of the CRB is expected to be negligible. Therefore, this calculation considers the CRB to be a surface-founded structure.

For the load combination with the seismic demand, an acceptable FOS is not met against sliding and overturning. To make a detailed assessment, nonlinear transient analyses are performed for seismic events and soil libraries as described in the following sections.

3.8.5.3.3 Control Building Nonlinear Analysis Model Description

Stability of the CRB model under seismic load is further evaluated through a set of nonlinear transient analyses. In these analyses, the nonlinearity stems from the interface between the CRB base and the underlying soil, which is modeled as a frictional surface that allows both sliding and gap formation.

To reduce the computational effort for the nonlinear transient analyses to an acceptable level, a new CRB model is created with a coarser mesh than the one used for the seismic analysis. This coarse CRB finite element model is built with an average element size of 5 ft using ANSYS SHELL181 elements. The coarse CRB model is verified through a modal analysis. The buoyancy load is explicitly applied at the base as pressure.

The static and kinetic coefficients of friction are presented in Table 3.8.5-6.

For each nonlinear transient analysis, acceleration time histories from the corresponding linear seismic harmonic analyses are imposed as reference frame acceleration, after increasing their amplitudes by 10 percent, which is equivalent to a factor of safety of 1.1. The acceleration time histories vary depending on the soil library and the seismic event of interest.

In the nonlinear transient analysis, Rayleigh damping is used by setting the target damping level to 7 percent at 50 Hz. In order to avoid potential complications associated with the use of mass contribution in a full transient Rayleigh damping analysis, only the stiffness matrix is used to introduce damping to the system. This is equivalent to having a linear damping ratio that increases with frequency. Appendix 3B describes the procedure for deriving the hybrid cracked CRB model.

The Hilber-Huges-Taylor (HHT) implicit time integration method is used for the transient analysis. The HHT method is known to give more accurate results than the Newmark algorithm with the same time integration step and it provides more control over the numerical damping. The convergence of the

solution is verified by repeating the analysis for selected cases with reduced time steps.

COL Item 3.8-3: An applicant that references the NuScale Power Plant US460 standard design will identify local stiff and soft spots in the foundation soil and address these in the design, as necessary.

3.8.5.4 Loads and Load Combinations

The loads and load combinations used for the design of the RXB and CRB, including the design of the foundations, are discussed in Section 3.8.4.

Stability Load Combinations

The load combinations used for the assessment of stability (flotation, uplift, sliding, overturning) for RXB and CRB are discussed below.

Four load combinations are considered:

A. D+H+W

B. D+H+E_s

C. D+H+ (W_t or W_h)

D. D+F'+E_s

where:

D = Dead load

H = Weight and pressure of soil

W = Operating basis wind load

E_s = Load effects from the SSE

W_t = Loads generated by the design basis tornado

W_h = Loads generated by the design basis hurricane

F’ = Flotation (buoyancy) due to design basis flood

E_s is added to load combination D for conservatism. The RXB reaction forces generated by wind are orders of magnitude less than those generated by the SSE event. As such, load combinations A and C are ignored for the RXB analysis.

For the CRB, the reaction forces generated by wind, and hurricane differ by a scaling factor, with the hurricane reaction forces having the highest amplitude.

Also, since the base of the CRB is shallow (5 ft), the soil weight and pressure acting around the basemat are ignored. Thus, the load combinations of interest are reduced to two:

Load Combination I: D + F' + W_h

Load Combination II: D + F' + E_s

The loads are discussed in Section 3.8.4.

3.8.5.4.1 Lateral Soil Force and Seismic Loads

The RXB is an embedded structure and, therefore, the surrounding soil contributes significantly to the stability of the structure.For the RXB, the surrounding soil imposes lateral soil pressures. The seismic inertia loads cause sliding and overturning forces.

The RXB exterior walls that are embedded below grade are subjected to static and dynamic lateral soil pressure loads. The soil-structure interaction analyses considers the dynamic soil pressure due to a seismic event. The static lateral soil pressure induced by the weight of soil, hydrostatic pressure, and a surcharge load at grade level is calculated.

The CRB exterior walls are not subject to static and dynamic lateral soil pressure loads because the CRB is not embedded in the soil.

The static lateral soil pressure values on walls are established in Section 3.8.4. The RXB static lateral soil pressure values are converted to force in accordance with the following example for the static effective soil force on the RXB north and south wall.

$$F_{y_soil} = L_{EW} \times K_o \times \left[P_{surcharge} \times H_{embed} + \frac{1}{2}(\gamma_{soil}) \times H_{embed}^{2} \right] \qquad \text{Eq. 3.8-2}$$

where

K_0	Soil Coefficient of Pressure at rest = 0.5
H_{embed}	RXB Embedment = 83 ft (Table 3.8.5-2a)
L_{EW}	RXB East-West Length between Exterior Faces of 4' Walls = 230 ft (Table 3.8.5-2a)
$P_{surcharge}$	Surcharge = 0.250 ksf (Table 3.8.5-2a)
$\gamma soil$	Soil Density = 0.13 kcf (Table 3.8.5-2a)

The soil forces on the east-west (F_x) and north-south (F_y) walls are 36,429 kips, and 54,233 kips, respectively.

Parameters used for the stability analysis of the RXB are presented in Table 3.8.5-2a. Control Building stability input evaluation parameters are presented in Table 3.8.5-6.

3.8.5.4.2 Effective Vertical Load

For stability evaluations, the effective vertical load of the building is an important stabilizing force. There are three components of vertical forces involved in the calculation of the flotation stability:

1) Dead weight of the building

2) Buoyancy load from the water table at grade, which reduces effective dead weight

3) Frictional forces acting on the exterior walls below grade. Since the CRB is a surface-based structure, this force does not contribute to its stability

Using these forces, the effective vertical load, W_{eff}, calculation is described below.

Reactor Building Effective Dead Weight with Water Table Buoyancy

The effective dead weight, or buoyant weight, is calculated using the following equation:

$$W_{eff}=W_{RXB}-F_{buoy} \quad \text{Eq. 3.8-3}$$

where,

W_{RXB} = 326,000 kips.

Therefore,

W_{eff} = 326,000 - 184,200 = 141,800 *kips*

The friction load on the basemat that helps resist sliding is calculated as per equation:

$$F_{base_fric}=W_{eff}*\mu_{fric} \quad \text{Eq. 3.8-4}$$

F_{base_fric}=141,800*0.577=81,800 *kips*

Control Building Effective Dead Weight with Water Table Buoyancy

The buoyancy load is calculated for the water level at grade. As the CRB base is at 5 ft depth below grade and the water unit weight is 62.4 pounds per cubic ft (pcf), the buoyancy force acting on the CRB is calculated as 1,983.8 kip.

The effective dead weight is calculated by subtracting buoyancy load from the dead weight as 19,492 kip.

3.8.5.4.3 Frictional Resistance Loads

The coefficient of friction between concrete and soil is established, based on the angle of internal friction Φ, where Φ is equal to 30°($\mu_{fric}=\tan\Phi$).

Frictional resistance loads are considered to stabilize the structure against floating, sliding, and overturning loads since the RXB is a deeply embedded structure.

The frictional resistance consists of two force resultant components, listed below:

1) Total sliding frictional resistance on foundation surface from effective vertical load, W_{eff}

$$\mu fric = \tan\Phi \quad \text{Eq. 3.8-5}$$
$$F_{base_fric} = W_{eff} * \mu fric$$

2) Friction forces resulting from at-rest earth pressures.

The at-rest earth pressure coefficient is:

$$K_0 = 1 - \sin\Phi \quad \text{Eq. 3.8-6}$$

Now, the total pressure force acting on each wall is derived. Forces labeled with an “x” act in the EW direction while forces with a “y” act in the NS direction.

$$F_{x_soil} = L_{NS} \times K_0 \times \left[P_{surcharge} \times H_{embed} + \frac{1}{2}(y_{soil}) \times H_{embed}^{2} \right] \quad \text{Eq. 3.8-7}$$

$$F_{y_soil} = L_{EW} \times K_0 \times \left[P_{surcharge} \times H_{embed} + \frac{1}{2}(y_{soil}) \times H_{embed}^{2} \right] \quad \text{Eq. 3.8-8}$$

The normal friction force arising from each of these pressure loads is then calculated.

$$F_{x_fric} = F_{x_soil} \times \mu_{fric} \quad \text{Eq. 3.8-9}$$

$$F_{y_fric} = F_{y_soil} \times \mu_{fric} \quad \text{Eq. 3.8-10}$$

Finally, the resultant total friction load on the walls that help resist flotation is found.

$$F_{v_fric} = 2 \times (F_{x_fric} + F_{y_fric}) \quad \text{Eq. 3.8-11}$$

Using the parameters from Table 3.8.5-2a, Equation 3.8-6 through Equation 3.8-11 is numerically solved:

$$K_0 = 1 - \sin 30 = 0.5$$

$$F_{x_soil} = 155.5 \times 0.5 \times \left[250 \times 83 + \frac{1}{2}(130) \times 83^2\right] = 36{,}429{,}000\ lbs = 36{,}429\ kips$$

$$F_{y_soil} = 231.5 \times 0.5 \times \left[250 \times 83 + \frac{1}{2}(130) \times 83^2\right] = 54{,}233{,}000\ lbs = 54{,}233\ kips$$

$$F_{x_fric} = 36{,}429 \times 0.577 = 21.019\ kips$$

$$F_{y_fric} = 54{,}233 \times 0.577 = 31{,}292\ kips$$

$$F_{v_fric} = 2 \times (21,019 + 31,292) = 104{,}622\ kips$$

3.8.5.4.3.1 Passive and Active Earth Pressures and Corresponding Friction Force Derivation

Although the deeply embedded RXB does not need to be checked for sliding and overturning instability, the amount of passive and active earth pressure that is needed to meet a FOS of 1.1 is still calculated.

First, the passive earth pressure coefficient is determined.

$$K_p = \frac{1 + \sin\Phi}{1 - \sin\Phi} \quad \text{Eq. 3.8-12}$$

With this coefficient, the total passive pressure force acting on each wall is derived. Forces labeled with an “x” act in the EW direction while forces with a “y” act in the NS direction.

$$F_{p_x} = \frac{1}{2} L_{NS} \times K_p \times \gamma_{soil} \times H_{embed}^{2} \quad \text{Eq. 3.8-13}$$

$$F_{p_y} = \frac{1}{2} L_{EW} \times K_p \times \gamma_{soil} \times H_{embed}^{2} \quad \text{Eq. 3.8-14}$$

Next, the active earth pressure coefficient is determined.

$$K_a = \frac{1 - \sin \Phi}{1 + \sin \Phi} \quad \text{Eq. 3.8-15}$$

The total active pressure force acting on each wall is now derived. Forces labeled with an “x” act in the EW direction while forces with a “y” act in the NS direction.

$$F_{a_x} = \frac{1}{2} L_{NS} \times K_a \times \gamma_{soil} \times H_{embed}^{2} \quad \text{Eq. 3.8-16}$$

$$F_{a_y} = \frac{1}{2} L_{EW} \times K_a \times \gamma_{soil} \times H_{embed}^{2} \quad \text{Eq. 3.8-17}$$

Using the parameters from Table 3.8.5-2a, Equation 3.8-12 through Equation 3.8-17 is numerically solved:

$$K_p = \frac{1 + \sin 30}{1 - \sin 30} = 3$$

$$F_{p_x} = \frac{1}{2} 155.5 \times 3 \times 130 \times 83^2 = 208{,}892{,}000 \; lbs = 208{,}892 \; kips$$

$$F_{p_y} = \frac{1}{2} 231.5 \times 3 \times 130 \times 83^2 = 310{,}987{,}000 \; lbs = 310{,}987 \; kips$$

$$K_a = \frac{1 - \sin 30}{1 + \sin 30} = 0.33$$

$$F_{a_x} = \frac{1}{2} 155.5 \times 0.33 \times 130 \times 83^2 = 23{,}210{,}000 \; lbs = 23{,}210 \; kips$$

$$F_{a_y} = \frac{1}{2} 231.5 \times 0.33 \times 130 \times 83^2 = 34{,}554{,}000 \; lbs = 34{,}554 \; kips$$

3.8.5.4.3.2 Overturning Moment Resistance in East-West Direction Derivation

With the resisting forces established in the above equations, the resultant resisting moments is now calculated. Figure 3.8.5-4 and Figure 3.8.5-5 help illustrate these moments and their associated moment arms.

First, moments resulting from friction forces on the east and west walls along with the active pressure friction force acting on the north or south wall are established.

$$M_{x_fric} = 2 \times \left(F_{x_fric} \frac{L_{NS}}{2} \right) \quad \text{Eq. 3.8-18}$$

$$M_{ax_\text{fric}} = \mu_{fric} \times F_{a_y} \times L_{NS} \quad \text{Eq. 3.8-19}$$

The moment resulting from the effective dead weight is calculated.

$$M_{x_dead} = W_{eff} \frac{L_{NS}}{2} \quad \text{Eq. 3.8-20}$$

Moments resulting from the passive and active pressure are determined. Note that the active pressure works in the opposite direction and is thus considered a demand load, not a resisting load.

$$M_{p_x} = F_{p_y} \frac{H_{embed}}{3} \quad \text{Eq. 3.8-21}$$

$$M_{a_x} = F_{a_y} \frac{H_{embed}}{3} \quad \text{Eq. 3.8-22}$$

Using the parameters from Table 3.8.5-2a, Equation 3.8-18 through Equation 3.8-22 is numerically solved:

$$M_{x_fric} = 2 \times \left(21{,}019 \times \frac{155.5}{2} \right) = 3{,}268{,}000 \; kip - ft$$

$$M_{x_fric} = 0.577 \times 34{,}554 \times 155.5 = 3{,}100{,}000 \; kip - ft$$

$$M_{x_dead} = 141{,}800 \times \frac{155.5}{2} = 11{,}025{,}000 \; kip - ft$$

$$M_{p_x} = 310{,}987 \times \frac{83}{3} = 8{,}604{,}000 \; kip - ft$$

$$M_{a_x} = 34{,}554 \times \frac{83}{3} = 956{,}000\ kip - ft$$

3.8.5.4.3.3 Overturning Moment Resistance in North-South Direction Derivation

First, moments resulting from friction forces on the north and south walls along with the active pressure friction force acting on the east or west wall are established.

$$M_{y_fric} = 2 \times \left(F_{y_fric} \frac{L_{EW}}{2} \right) \quad \text{Eq. 3.8-23}$$

$$M_{ay_fric} = \mu_{fric} \times F_{a_x} \times L_{EW} \quad \text{Eq. 3.8-24}$$

The moment resulting from the effective dead weight is calculated.

$$M_{y_dead} = W_{eff} \frac{L_{EW}}{2} \quad \text{Eq. 3.8-25}$$

Moments resulting from the passive and active pressure are determined. Note that the active pressure works in the opposite direction and is thus considered a demand load, not a resisting load.

$$M_{p_y} = F_{p_x} \frac{H_{embed}}{3} \quad \text{Eq. 3.8-26}$$

$$M_{a_y} = F_{a_x} \frac{H_{embed}}{3} \quad \text{Eq. 3.8-27}$$

Using the parameters from Table 3.8.5-2a, Equation 3.8-23 through Equation 3.8-27 is numerically solved:

$$M_{y_fric} = 2 \times \left(31{,}292 \times \frac{231.5}{2} \right) = 7{,}244{,}000\ kip - ft$$

$$M_{ay_fric} = 0.577 \times 23{,}210 \times 231.5 = 3{,}100{,}000\ kip - ft$$

$$M_{y_dead} = 141{,}800 \times \frac{231.5}{2} = 16{,}413{,}000\ kip - ft$$

$$M_{p_y} = 208{,}892 \times \frac{83}{3} = 5{,}779{,}000\ kip - ft$$

$$M_{a_y} = 23{,}210 \times \frac{83}{3} = 642{,}000\ kip - ft$$

3.8.5.4.3.4 Factor of Safety Derivation

The factors of safety against flotation, sliding, and overturning is derived using the equations presented above. Several terms related to demand come from the seismic results covered in Section 3.7.2.

Figure 3.8.5-1 through Figure 3.8.5-5 provide free body diagrams of the forces at play when establishing each FOS.

For terms relating to passive and active forces/moments, a scaling factor, C_i, is introduced. This factor is iterated over until the minimum factor of safety across all fields investigated is equal to the acceptable quantities provided in Section 3.8.5.6.

$$FOS_{float} = \frac{W_{RXB} + F_{v_fric}}{F_{buoy} + F_{v_seis}} \quad \text{Eq. 3.8-28}$$

Uplift is computed for a flooding event acting simultaneously with the maximum vertical seismic force. The factors of safety against flotation, sliding, and overturning is derived using the equations presented below.

For terms relating to passive and active forces/moments, a scaling factor, C_i, is introduced. This factor is iterated upon until the minimum factor of safety across all fields investigated is equal to the acceptable value of 1.1. Figure 3.8.5-1 through Figure 3.8.5-5 provide the free body diagrams of the forces at play when establishing each FOS as noted by equations below.

$$FOS_{float} = \frac{W_{RXB} + F_{v_fric}}{F_{buoy} + F_{v_seis}} \quad \text{Eq. 3.8-29}$$

$$FOS_{slide_x} = \frac{2*F_{y_fric} + F_{base_fric} + C_i F_{p_x}}{F_{x_seis} + C_i F_{a_x}} \quad \text{Eq. 3.8-30}$$

$$FOS_{slide_y} = \frac{2*F_{x_fric} + F_{base_fric} + C_i F_{p_y}}{F_{y_seis} + C_i F_{a_y}} \quad \text{Eq. 3.8-31}$$

$$FOS_{overturn_x} = \frac{M_{x_fric} + C_i M_{ax_fric} + M_{x_dead} + C_i M_{p_x}}{F_{v_seis}\frac{L_{NS}}{2} + M_{x_seis} + C_i M_{a_x}} \quad \text{Eq. 3.8-32}$$

$$FOS_{overturn_y} = \frac{M_{y_fric} + C_i M_{ay_fric} + M_{y_dead} + C_i M_{p_y}}{F_{v_seis}\frac{L_{EW}}{2} + M_{y_seis} + C_i M_{a_y}} \quad \text{Eq. 3.8-33}$$

3.8.5.5 Results Compared with Structural Acceptance Criteria

3.8.5.5.1 Reactor Building Stability

The FOS are determined for the RXB and are shown in Table 3.8.5-3. The minimum acceptable factor of safety for flotation, uplift, sliding, and overturning is 1.1.

3.8.5.5.1.1 Reactor Building Uplift

Equation 3.8-29 is used to calculate the uplift FOS. The uplift FOS (flotation) for RXB is shown in Table 3.8.5-3 and found to be acceptable. Uplift is computed for a flooding event acting simultaneously with the maximum vertical seismic force.

Based on the examination of the total vertical reaction force underneath the basemat, all net vertical reactions are in compression.

3.8.5.5.1.2 Reactor Building Sliding

Equation 3.8-30 and Equation 3.8-31 are used to calculate the sliding FOS in both the X and Y directions for each soil type across its corresponding seismic events. The average of the FOS of seismic events within a soil type is then computed and used to evaluate against an acceptable FOS of 1.1. A passive and active pressure factor, ci, of 0.08 is used, resulting in a FOS of 1.28. This calculation is summarized in Table 3.8.5-13.

3.8.5.5.1.3 Reactor Building Overturning

Equation 3.8-32 and Equation 3.8-33 are solved numerically at every time step in each seismic event's time history. Although the numerators in these equations do not change, the denominator does, due to the signed moment and vertical seismic force for each particular time step. Since the signs of the different terms in the denominator may be opposing, each edge of the RXB basemat needs to be evaluated separately. The minimum FOS for every event and soil type with a passive and active pressure factor, c_i, of 0.08 is presented in Table 3.8.5-14. The average of the FOS of seismic events within a soil type is computed and used for evaluation. The resulting FOS is 1.1, exactly matching the acceptable FOS and validating a passive and active pressure factor of 0.08 as the minimum factor needed to have the FOS meet the acceptance criteria.

3.8.5.5.2 Control Building Stability

The resisting force to sliding is the friction that develops due to the compressive normal force between the CRB base and the soil. The normal compressive force is calculated by subtracting the vertical base reaction force (FZ) and buoyancy force from the dead weight of the CRB. To calculate the friction force, static coefficient of friction (μs) is multiplied by the compressive normal force. Driving force is the base reaction force that acts in the lateral directions ($FD=\sqrt{FX^2+FY^2}$) from the hurricane and seismic analyses.

The minimum acceptable factor of safety for flotation, uplift, sliding, and overturning is 1.1. This factor is not achieved for the CRB sliding using a linear analysis.

The uplift, sliding, and overturning stability evaluation of the CRB is performed using a nonlinear analysis to show that sliding, overturning, and uplift are insignificant.

3.8.5.5.2.1 Control Building Uplift

The only resisting force to uplift is the dead weight of the CRB. Driving forces are the buoyancy and the vertical component of the base reaction forces (FZ) that are calculated from the hurricane and seismic analyses.

The base reaction from the seismic harmonic analyses are time histories. For the FOS calculations, only the peak absolute vertical reaction forces are used as the driving forces.

3.8.5.5.2.1.1 Dynamic Control Building Uplift Ratio

The peak seismic base reaction forces for soil libraries and events of interest are calculated by summing nodal forces at the base of the CRB basemat.

As shown in Table 3.8.5-18, the peak vertical seismic reaction forces are always less than the resisting force. The resisting force is calculated by subtracting the buoyancy from the dead weight of the CRB. The dead weight corresponds to the self-weight of the concrete and steel structures, and equipment, based on ANSYS calculations. Thus, the net reactions are always in compression.

The typical total basemat seismic vertical reaction time history and Fourier amplitude spectrum are shown in Figure 3.8.5-17.

3.8.5.5.2.2 Control Building Sliding

The stability analyses performed with the force equilibrium method shows that the hurricane loads do not cause stability issues. However, the calculated FOS values fall below the minimum acceptable level under

seismic loads. The CRB sliding FOS time histories are shown in Figure 3.8.5-16a, Figure 3.8.5-16b, and Figure 3.8.5-16c. Therefore, a set of nonlinear transient analyses is performed for seismic events of interest.

The maximum absolute sliding, taken as the SRSS of the directional sliding values shown in Table 3.8.5-17, is calculated from the nonlinear transient analysis as 1.3 inches. This value is considered to be acceptable given the level of conservatism in the analyses and the distance of the CRB to the nearby SC-II structure.

3.8.5.5.2.3 Control Building Overturning

The minimum overturning FOS values under hurricane load are calculated for the north and south edges of the basemat as 3.71 and 3.72 respectively. As the CRB is wider in X-direction (east-west), the wind loads on the north and south walls lead to higher moments at the bases of these walls compared to east and west edges of the basemat. Since the minimum overturning FOS value is higher than the limit of 1.1, overturning is not a concern under hurricane load for the CRB.

Under seismic load, the overturning stability analyses considering seismic loading show FOS values less than 1.1 for Soil-7 (as shown in Figure 3.8.5-16d), which are instantaneous and therefore are expected to be negligible. Overturning stability is further analyzed through nonlinear transient analyses for the load combination with seismic load.

3.8.5.6 Average Bearing Pressures Approach

Bearing pressure values under the RXB and RWB models are calculated from the SOLID185 elements forming the soil layers under the basemats. Bearing pressure values under the CRB model are calculated similarly, but the nodes at the soil-structure interface beneath the basemat are used instead, as the CRB basemats utilize contact elements directly to connect to the soil in lieu of a transitional layer of backfill elements. Mean bearing pressure values are calculated by dividing the sum of the nodal forces in vertical direction under the basemats by the corresponding areas. Figure 3.8.5-6 shows the areas selected to calculate the toe pressure values under RXB and RWB models and Figure 3.8.5-6a shows the same under the CRB.

Considering the rate of change of the toe pressure with the addition of new rows of element and the vertical stress fluctuations close to the basemat edges, three rows of elements are used to define the areas for the calculation of the toe pressure. Three element rows correspond to around 12, 9, and 4 foot length for the RXB, RWB, and CRB basemats, respectively.

Reactor Building

Initial estimates of the RXB static bearing pressure values are presented. Table 3.8.5-1 presents the mass of each building in DB model and the approximate soil and buoyant pressures under their basemats. A comparison of

pre-excavation and post-construction total soil pressure values shows that there is a drop of around 10 percent and 20 percent in the pressures under RXB and RWB respectively. Although not included in the settlement analysis, buoyant pressure values are listed for comparison.

Control Building

Initial estimates of the CRB static bearing pressure values are presented in Table 3.8.5-2. The CRB model is analyzed to recover the static bearing pressure of the basemat. Table 3.8.5-5 lists these results. These bearing pressures are evaluated against the allowable limit and found to be acceptable. Figure 3.8.5-9 provides a plot of this contact pressure.

3.8.5.6.1 Average Bearing Pressure Results

The mean static normal stress values underneath the basemats are presented in Table 3.8.5-4 and Table 3.8.5-5. These values are similar to the initial estimates based on mass calculations presented in Table 3.8.5-1, for the RXB, and Table 3.8.5-2, for the CRB, which is a verification of the analysis results. The recovered values are 15.9 ksf for the RXB and 5.8 ksf for the CRB.

The mean dynamic bearing pressure values are presented in Table 3.8.5-4 and Table 3.8.5-5. The total maximum dynamic bearing pressure (static+dynamic) values are 33.6 ksf for the RXB and 25.1 ksf for the CRB.

The static-to-dynamic bearing pressure ratios at the RXB, RWB, and CRB bases are 2.38, 2.26, and 2.30 respectively. The ratio approaches 1.0 and falls below it for the toe pressures of the RXB and CRB.

Bearing pressure is used to establish a design parameter for allowable bearing capacity for site selection.

3.8.5.6.2 Bearing Pressure along Edges (Toe Pressure)

Reactor Building

Table 3.8.5-4 lists the RXB average bearing and toe pressure values under the RXB basemat.

Control Building

Toe pressures are calculated for the base area that is covered by four layers of nodes (and the elements attached to them on the side of the building) from the edge of the basemats. Four layers of nodes from the edge to the interior have a width of approximately 4 ft on the CRB basemat. Using multiple layers of nodes smoothens the stress concentration and fluctuations that appear towards the edges of the basemats. These stress concentrations and fluctuations develop due to the sharp stiffness change between the soil and structure media at the interface and the discontinuity of the secondary

variables (stress, strain) in the finite element method that is more pronounced for low order elements such as SOLID185 that is used to represent the soil medium.

The toe pressures are calculated for each edge and presented in Table 3.8.5-5, which also shows the average bearing pressure.

3.8.5.7 Settlement

3.8.5.7.1 Settlement Approach

A large-scale ANSYS finite element model is used to determine the effect of foundation differential movements of the RXB and RWB comprising of uncracked and cracked structural members, referred to as hybrid static double building (DB) model. The cracking state of this model corresponds to the hybrid DB seismic model used for design (Section 3.7.2). The CRB is analyzed independently in a similar ANSYS model. Appendix 3B describes the procedure for deriving the hybrid cracked models. To maximize the effect of the differential movements, the soil is modeled using the softest soil profile, i.e., Soil Type 11. In addition, soil stiffnesses are further reduced by 50 percent to amplify the effect of differential movements or settlements. The 50 percent reduction in soil stiffness includes the areas below the basemats and is extended to the entire free-field soil model. The size of the soil included in the model is so large that the static displacements induced by static loads of the structures become negligible on the edges of the free-field soil model.

The size of the free-field soil block in the DB model is 4000 ft x 4000 ft wide and 521 ft deep. The size of the free-field soil block in the CRB model is approximately 1350 ft x 1350 ft wide and 300 ft deep, as the CRB has a smaller footprint and is not embedded. Figure 3.8.5-7 shows the size of the free-field soil block with the two embedded buildings of the DB model and Figure 3.8.5-8 shows a quarter of the CRB soil.

The hybrid static double building and CRB models are used to analyze service load and settlement demands on the structural members under static load combination.

Service Loads

According to ACI 349-13, differential settlement load demand for the structural members is to be included within the dead load. The estimation of differential settlement is to be based on assessment of the structure in service. The static load combination of dead, live, hydrostatic, and effective earth pressure (which excludes the groundwater pressure) is chosen for differential settlement analysis. Within the differential settlement analysis for the static bearing pressure, buoyancy forces due to water table are ignored. Ignoring the ground water allows the buoyancy forces to disappear making the

settlement analysis more conservative. This load combination is defined as follows:

U= D+F+L+H

where U is total load, D is dead load, F is the hydrostatic loads that stem from the RXB pool, L is live load, and H is the effective earth pressure with surcharge load excluding hydrostatic loads. Since the CRB is surface founded and does not have a pool, the load combination reduces to simply D+L for its analysis.

Lateral effective earth pressure that acts on the RXB is modeled explicitly using the following equation:

$H(z)=K*(\gamma s*(zg-z)+sq)$

where H(z) is the lateral effective earth pressure at elevation z, K is the coefficient of lateral earth pressure, which is taken to be 0.5, γs is the unit weight of the unsaturated soil, zg is the elevation at grade, and sg is the surcharge load.

In modeling the lateral effective earth pressure, surface loads acting at the grade level to the north and east of RWB model are ignored. These surface loads (approximately 227 psf), which are introduced in the RWB model to represent the Seismic Category III structures above grade, are less than the surcharge load of 250 psf; therefore, using surcharge load is conservative.

3.8.5.7.2 Settlement Results

Reactor Building

Figure 3.8.5-10 presents the vertical displacement field (UZ) of the DB model and shows that the RWB tilts towards the RXB. Given the relatively higher weight of the RXB with respect to the RWB, it is an expected result.

The settlement under RXB and RWB models is presented in Figure 3.8.5-11 and Figure 3.8.5-12 respectively. It is visible in these figures that settlement maximizes under the reactor pool with the peak value of 1.48 inches.

Table 3.8.5-7 and Table 3.8.5-8 list displacement, differential settlement values, and tilt in inches per 50 ft for a set of nodes selected on the RXB basemat. The locations of these nodes are shown in Figure 3.8.5-13.

Table 3.8.5-9 and Table 3.8.5-10 list displacement, differential settlement values, and tilt in inches per 50 ft for a set of nodes selected on the RWB basemat. The locations of these nodes are shown in Figure 3.8.5-14. For the RWB, the selected nodes are used to calculate the differential settlement not only for the basemat but also for the tunnels.

Analysis results show that the RWB tilts predominately around the north-south axis towards the RXB. The RXB, on the other hand, experiences a relatively uniform settlement with higher values under the reactor pool. Given the weight of the pool, six NPMs in the pool, and the crane located above the pool, relatively higher settlement below the pool is an expected outcome.

Control Building

Due to the higher mass spread over a smaller area in the SC-I portion, the maximum vertical displacement occurs in this region, as seen in Figure 3.8.5-12a. The maximum vertical displacement is 0.77 inches.

Differential settlement, total tilt, and tilt in a 1 inch per 50 feet threshold are summarized in Table 3.8.5-11. The node locations and displacements used to obtain these values are provided in Table 3.8.5-12. Figure 3.8.5-15 provides a plot of the soil's vertical displacement and Figure 3.8.5-16 depicts an exaggerated plot of the CRB tilt. Since the SC-I portion has more mass spread over a smaller area and is located south of the SC-II region, it was expected to have positive rotation about the west direction for the SC-I basemat and about the east direction for the SC-II basemat.

3.8.5.8 Thermal Loads

In design of RC members including basemat, the design demands from thermal effects due to accident temperature are not directly included in design load combinations. Instead, thermal effects are considered by calculating the capacity of concrete sections by limiting the "usable" axial and bending strains to the allowable strains reduced by the thermal strains.

Loads resulting from thermal effects are self-relieving; that is, thermal forces and moments are greatly reduced or completely relieved with the progress of concrete cracking and reinforcement yielding. In the presence of thermal effects, the concrete design is performed by comparing design demands from mechanical loads to "usable" strength which is calculated based on "usable" strains. The "usable" strains are calculated by subtracting the thermal strains from the allowable strains. This calculation is further discussed in Section 3B.1.

3.8.5.9 Construction Loads

The RXB basemat is poured in a short time. The building is essentially constructed from the bottom up. The main loads (the reactor pool and the NPMs) are not added until the building is complete. Therefore, there are no construction-induced settlement concerns. The CRB basemat is smaller and poured later than the RXB basemat in the construction sequence.

3.8.5.10 Leak Detection

Groundwater has the potential to leak through the RXB exterior walls through microscopic concrete cracks. Due to the exterior concrete wall thickness, these leaks are very slow (<<1 gallon per day (gpd)). This leak rate through the wall is

not enough to cause an interior flood in any rooms with an exterior wall. Leaks of this nature will be discovered and addressed in accordance with plant concrete maintenance specifications. Further reduction of groundwater seepage can be accomplished with a building dewatering system surrounding the RXB.

A leak chase system is provided in the RXB basemat to detect leakage from the reactor pool.

3.8.5.11 Materials, Quality Control, and Special Construction Techniques

Section 3.8.4.6 describes the materials, quality control, and special construction techniques applicable to the RXB and CRB, including the foundations.

3.8.5.12 Testing and Inservice Inspection Requirements

Section 3.8.4.7 identifies the testing and inservice surveillances applicable to the RXB and CRB, including the foundations.

Table 3.8.5-1: Total and Soil Pressure Estimates Based on Building Weights under Dead and Live Loads

	RXB
Dead Load (kip)*	3.26E+05
Live Load (kip)	1.57E+04
D+L (kip)	3.42E+05
Area (ft^2)	3.60E+04
Base Pressure (ksf)	9.50
Buoyant Pressure (ksf)	5.18
Pre-Excavation Soil Pressure (ksf)	9.96
Soil Pressure Change at the base after construction (ksf)	-0.46

*Dead Load includes the weight of the pool water and NPMs.

Table 3.8.5-2: Expected Control Building Bearing Pressure on the Soil

Region	Total Mass	Mass Percent	Area	Calculated Soil Bearing Pres.	
	lbm	%	in^2	psi	ksf
SC-I	5.86E+04	55.4	915,600	24.71	3.56

Table 3.8.5-2a: Parameters Used in Reactor Building Stability Analysis

Symbol	Description	Value	Unit
W_{RXB}	Dead weight of RXB structure including equipment, pool, and concrete self-weight	326,000	kips
L_{EW}	RXB east-west length (between exterior faces of walls)	231.5	ft
L_{NS}	RXB north-south length (between exterior faces of walls)	155.5	ft
H_{embed}	RXB embedment depth	83	ft
Y_W	Water unit weight	62.4	lb/ft^3
Y_{soil}	Soil unit weight	130	lb/ft^3
$P_{surcharge}$	Surcharge	250	lb/ft^2
Φ	Minimum angle of internal friction	30	deg
D_{wtable}	Depth of water table below top of soil	1	ft

Table 3.8.5-3: Summary of Reactor Building Factors of Safety

	Flotation	EW Sliding	NS Sliding	EW Overturning	NS Overturning
	kip	*kip*	*kip*	*kip-ft*	*kip-ft*
Resistance	430,687	161,205	148,814	15,232,251	24,372,941
Demand	330,528	126,328	114,858	12,645,992	22,155,138
FOS	1.30	1.28	1.30	1.20	1.10
Minimum FOS	1.10	1.10	1.10	1.10	1.10

Table 3.8.5-4: Static, Seismic, and Dynamic Bearing Pressure Values and Ratios at the Base of the RXB Model

Areas	Mean Vertical Static Stress (ksf)	Mean Vertical Seismic Stress (ksf)	Mean Vertical Dynamic Stress (ksf) Static + Seismic	Static/Seismic Ratio
Basemat	9.51	3.99	13.50	2.38
North	15.54	17.48	33.02	0.89
South	15.89	17.68	33.57	0.90
East	15.58	17.06	32.64	0.91
West	13.00	15.61	28.61	0.83

Table 3.8.5-5: Static and Dynamic Bearing Pressure Values and Ratios at the Base of the CRB Model

Areas	Mean Vertical Static Stress (ksf)	Mean Vertical Dynamic Stress (ksf)	Mean Vertical Static+Dynamic Stress (ksf)	Static/Dynamic Ratio
Basemat	3.57	1.55	5.12	2.30
North	5.14	17.77	22.91	0.29
South	5.01	20.06	25.07	0.25
East	5.59	12.73	18.32	0.44
West	5.82	13.66	19.48	0.43

Table 3.8.5-6: Control Building Stability Input Evaluation Parameters

Data Description	Value
CRB Seismic Weight (kips)	21,476
Buoyancy Load (kips)(1)	1984
CRB East-West Length (ft)	116.667
CRB North-South Length (ft)	54.5
CRB Height (ft)	50.25
CRB Embedment Depth (ft)	5
Main Foundation Area (ft^2)	6358
Surcharge (psf)	250
Static Coefficient of Friction between Concrete and underlying Soil	0.58
Kinetic Coefficient of Friction between Concrete and underlying Soil	0.5

Note:

(1) Buoyancy load based on the water level at Elevation 100 ft for conservatism.

Table 3.8.5-7: Displacement Values for the Nodes Selected at the Base of RXB Model

Node ID	X (in)	Y (in)	Z (in)	UX (in)	UY (in)	UZ (in)
66	45	-933	12	1.09E-01	-7.27E-02	-1.30E+00
130	2823	-933	12	1.72E-01	-5.52E-02	-1.28E+00
1325	45	39	12	1.09E-01	-1.23E-02	-1.41E+00
1370	2823	45.75	12	1.70E-01	-3.78E-03	-1.38E+00
2519	45	933	12	1.15E-01	3.91E-02	-1.31E+00
2583	2823	933	12	1.66E-01	4.58E-02	-1.27E+00
88196	1980	39	12	1.41E-01	-7.74E-03	-1.49E+00

Table 3.8.5-8: Differential Settlement and Tilt Values Calculated for the Selected Nodes at the Base of RXB Model

Node-I	Node-II	Differential Settlement (in)	TILT (inch per 50 ft)
66	130	0.021	0.005
2519	2583	0.040	0.009
1325	88196	0.085	0.026
1370	88196	0.115	0.082

Table 3.8.5-9: Displacement Values for the Nodes Selected at the Base of RWB Model

Node ID	X (in)	Y (in)	Z (in)	UX (in)	UY (in)	UZ (in)
1000626	-315	-843	636	3.81E-02	-8.34E-03	-1.07E+00
1001344	-2499	-843	636	2.15E-02	-7.80E-03	-3.90E-01
1008575	-315	1155	636	4.65E-02	-4.32E-03	-1.04E+00
1009671	-315	-15	636	3.63E-02	-7.75E-03	-1.19E+00
1009680	27	-15	648	4.56E-02	6.62E-03	-1.33E+00
1009936	-2499	1155	636	2.70E-02	-1.60E-03	-5.44E-01
1011861	27	477	636	4.57E-02	-2.11E-02	-1.30E+00
1011862	-315	477	636	4.15E-02	-8.63E-03	-1.18E+00

Table 3.8.5-10: Differential Settlement and Tilt Values Calculated for the Selected Nodes at the Base of RWB Model

Node-I	Node-II	Differential Settlement (in.)	TILT (inch per 50 ft)
1008575	1009936	0.500	0.137
1000626	1001344	0.684	0.188
1011861	1011862	0.119	0.209
1009680	1009671	0.142	0.249

Table 3.8.5-11: Control Building Differential Settlement and Tilt Results

Region	Differential Settlement Direction	Edge	Differential Settlement	Total Tilt	Tilt (inch per 50')
			in	*in/in*	*in/ft*
SC-I	NS	E	0.1408	-2.15E-04	-0.12920
		W	-0.1421	-2.17E-04	-0.13034
	EW	S	-0.0286	2.05E-05	0.01228
		N	0.0299	2.14E-05	0.01281
Max:			-0.2643	2.41E-04	0.14469

Table 3.8.5-12: Control Building Basemat Corner Nodes and Displacements

Corner	Node #	Coordinate			Displacement		
		X	Y	Z	X	Y	Z
		in	in	in	in	in	in
SW	1	-700	-888.5	-30	4.19E-05	-3.87E-02	-8.14E-01
SE	4899	700	-888.5	-30	1.44E-03	-3.79E-02	-7.85E-01
NE	12659	700	-234.5	-30	2.50E-03	-4.03E-02	-9.26E-01
NW	11217	-700	-234.5	-30	-1.58E-03	-3.92E-02	-9.56E-01

Table 3.8.5-13: Reactor Building Sliding Factors of Safety for Every Seismic/Soil Configuration

Sliding Factor of Safety			
Soil Type	**Seismic Load Case**	**East-West Global FX**	**North-South Global FY**
Soil 7 CSDRS	Capitola	1.20	1.27
	Chi-Chi	1.16	1.49
	El Centro	1.42	1.45
	Izmit	1.19	1.38
	Yermo	1.41	1.34
	Average	**1.28**	**1.39**
Soil 7 Soil Separation CSDRS	Capitola	1.19	1.23
	Chi-Chi	1.15	1.47
	El Centro	1.44	1.38
	Izmit	1.25	1.37
	Yermo	1.38	1.37
	Average	**1.28**	**1.36**
Soil 9 CSDRS-HF	Lucerne	**2.76**	**2.94**
Soil 11 CSDRS	Capitola	1.38	1.31
	Chi-Chi	1.21	1.45
	El Centro	1.61	1.18
	Izmit	1.48	1.16
	Yermo	1.31	1.38
	Average	**1.40**	**1.30**

Table 3.8.5-14: Reactor Building Overturning Moment Factors of Safety for Every Seismic/Soil Configuration

Overturning Factor of Safety					
Soil Type	**Seismic Load Case**	**MX South Wall**	**MX North Wall**	**MY East Wall**	**MY West Wall**
Soil 7 CSDRS	Capitola	1.47	1.17	1.40	1.21
	Chi-Chi	1.32	1.19	1.47	1.22
	El Centro	1.11	1.10	1.54	1.34
	Izmit	1.41	1.21	1.38	1.27
	Yermo	1.28	1.06	1.24	1.16
	Average	**1.32**	**1.15**	**1.41**	**1.24**
Soil 7 Soil Separation CSDRS	Capitola	1.33	1.03	1.36	1.19
	Chi-Chi	1.22	1.01	1.33	1.20
	El Centro	1.01	1.21	1.64	1.31
	Izmit	1.20	1.15	1.40	1.19
	Yermo	1.13	1.09	1.15	1.06
	Average	**1.18**	**1.10**	**1.38**	**1.19**
Soil 9 CSDRS-HF	Lucerne	**1.76**	**1.89**	**2.30**	**1.99**
Soil 11 CSDRS	Capitola	1.75	1.85	2.11	1.77
	Chi-Chi	1.88	1.76	1.89	1.89
	El Centro	1.60	1.75	1.78	1.70
	Izmit	1.73	1.61	1.97	1.73
	Yermo	1.56	1.43	1.71	1.68
	Average	**1.70**	**1.68**	**1.89**	**1.75**

Table 3.8.5-15: Control Building Uplift Factor of Safety Values Calculated under Seismic Load

Soil-ID	Event-ID	F_D (kips)	F_R (kips)	FOS
7	CAP	11227.408	21476	1.913
7	CHI	12595.438	21476	1.705
7	IZM	12261.94	21476	1.751
7	YER	12321.804	21476	1.743
7	ELC	12250.601	21476	1.753
11	CAP	10532.291	21476	2.039
11	CHI	10264.075	21476	2.092
11	IZM	11647.141	21476	1.844
11	YER	10988.117	21476	1.954
11	ELC	11045.232	21476	1.944
9	LCN	11909.099	21476	1.803

Table 3.8.5-16: Control Building Peak Sliding and Rotation around Vertical Axis per Transient Analysis

Soil-ID	Event-ID	Peak Total Accumulated Sliding (in)	Peak Rotation Around Z-dir (deg)
7	CAP	1.90	0.01
7	CHI	2.19	0.02
7	YER	1.98	0.01
7	IZM	1.07	0.01
7	ELC	1.85	0.01
7	CAP	2.99	0.02
11	CHI	3.20	0.05
11	YER	2.49	0.03
11	IZM	1.88	0.01
11	ELC	2.00	0.04
9	LCN	0.32	0.01

Table 3.8.5-17: Control Building Peak Sliding and Rotation around Vertical Axis per Transient Analysis

Soil-ID	Event-ID	Max UX (in.)	Max UY (in.)	Max UZ (in.)	Peak Rotation Around Z-dir (deg)
7	CAP	0.58	0.51	0.02	0.01
7	CHI	0.95	0.48	0.02	0.03
7	YER	0.71	0.68	0.02	0.03
7	IZM	0.25	0.53	0.02	0.02
7	ELC	0.86	1.01	0.03	0.03
11	CAP	0.71	0.58	0.02	0.01
11	CHI	0.58	0.29	0.01	0.01
11	YER	0.47	0.43	0.01	0.01
11	IZM	0.32	0.44	0.01	0.01
11	ELC	0.41	0.95	0.01	0.01
9	LCN	0.26	0.16	0.02	0.02

Table 3.8.5-18: Control Building Uplift Factor of Safety Values Calculated under Seismic Load

Soil-ID	Event-ID	F_D (kips)	F_R (kips)	FOS
7	CAP	11,227.408	21,476	1.913
7	CHI	12,595.438	21,476	1.705
7	IZM	12,261.94	21,476	1.751
7	YER	12,321.804	21,476	1.743
7	ELC	12,250.601	21,476	1.753
11	CAP	10,532.291	21,476	2.039
11	CHI	10,264.075	21,476	2.092
11	IZM	11,647.141	21,476	1.844
11	YER	10,988.117	21,476	1.954
11	ELC	11,045.232	21,476	1.944
9	LCN	11,909.099	21,476	1.803

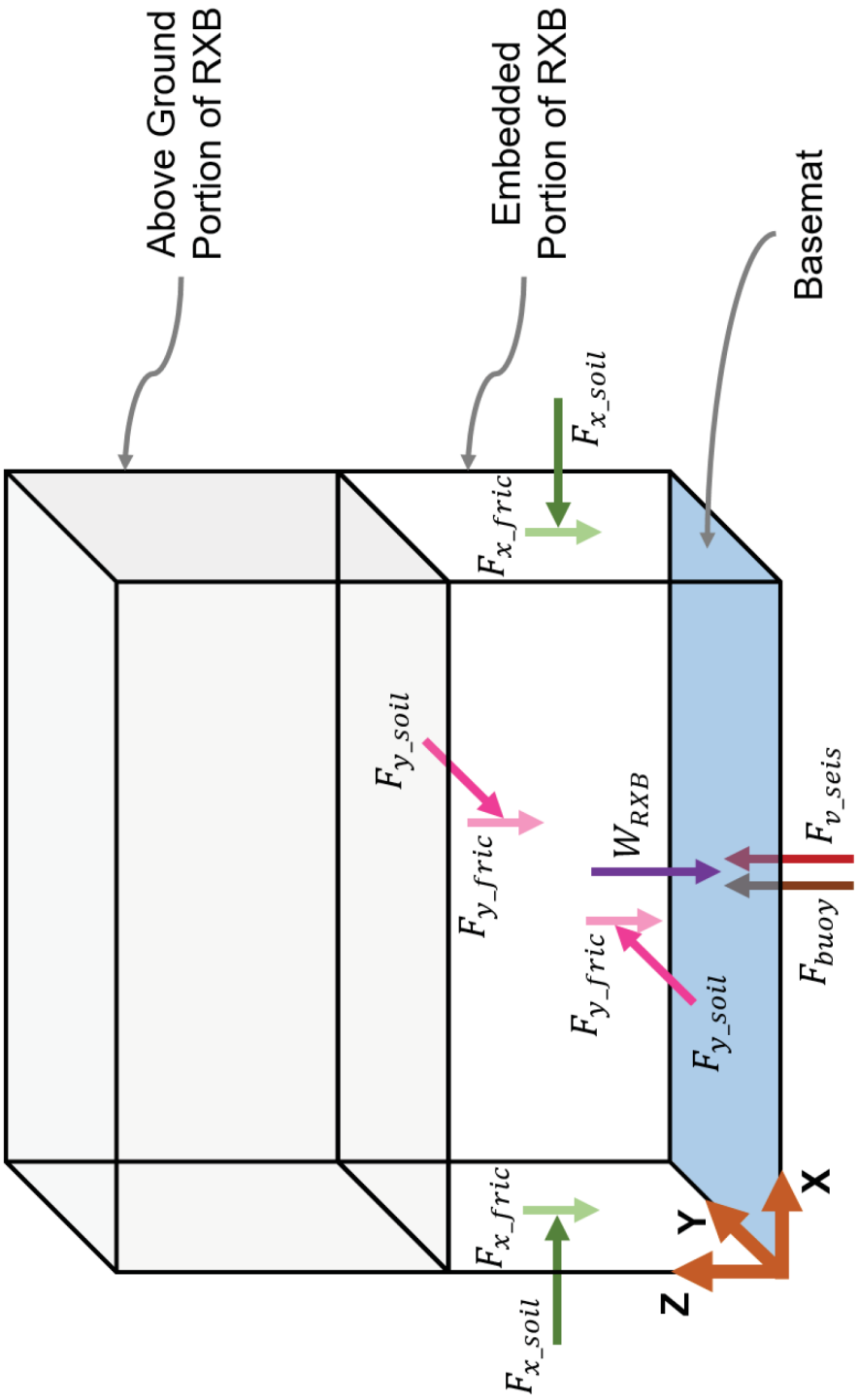


Figure 3.8.5-1: Illustration of Reactor Building Forces for the Vertical (Z) Uplift Case

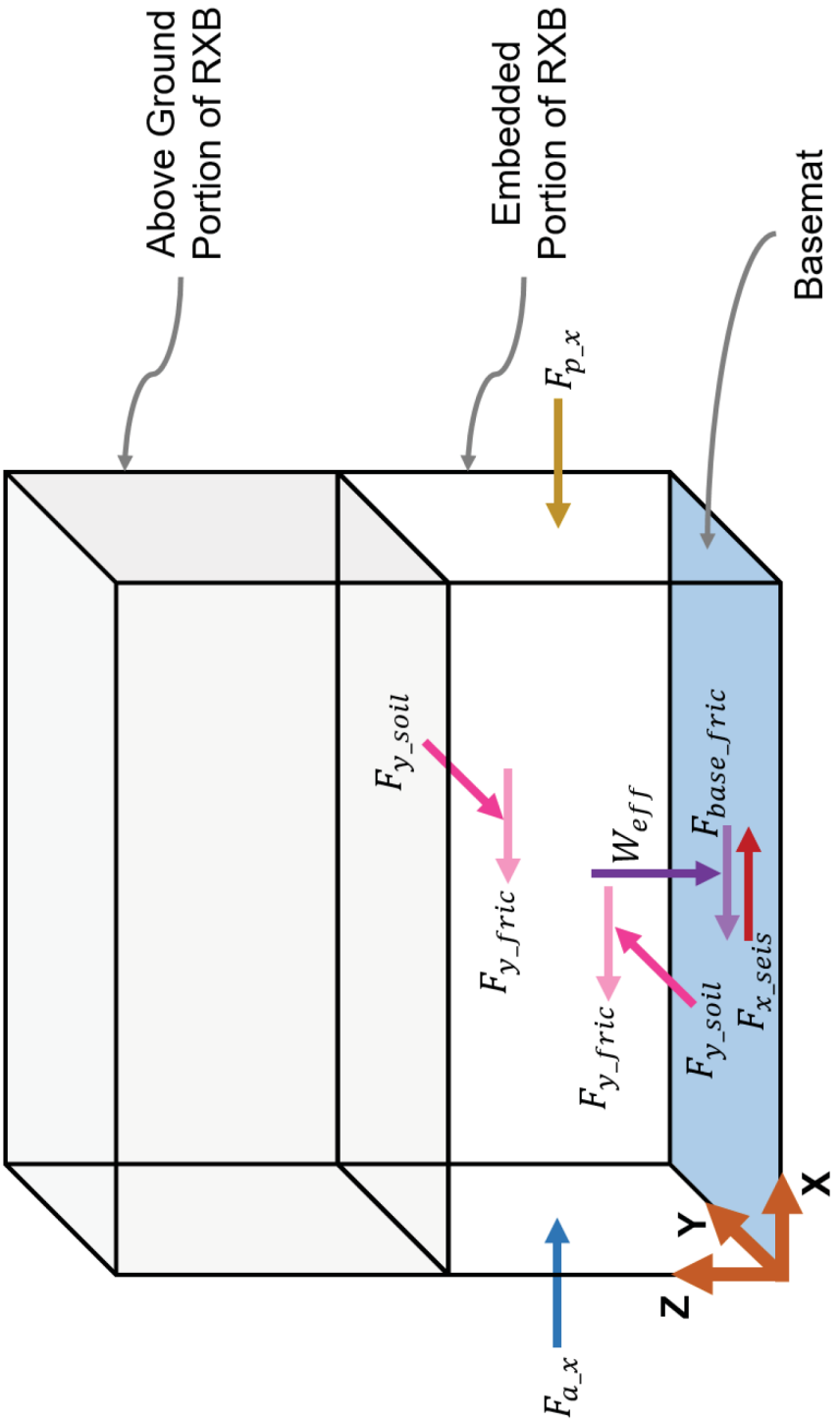


Figure 3.8.5-2: Illustration of Reactor Building Forces for the East-West (X) Sliding Case

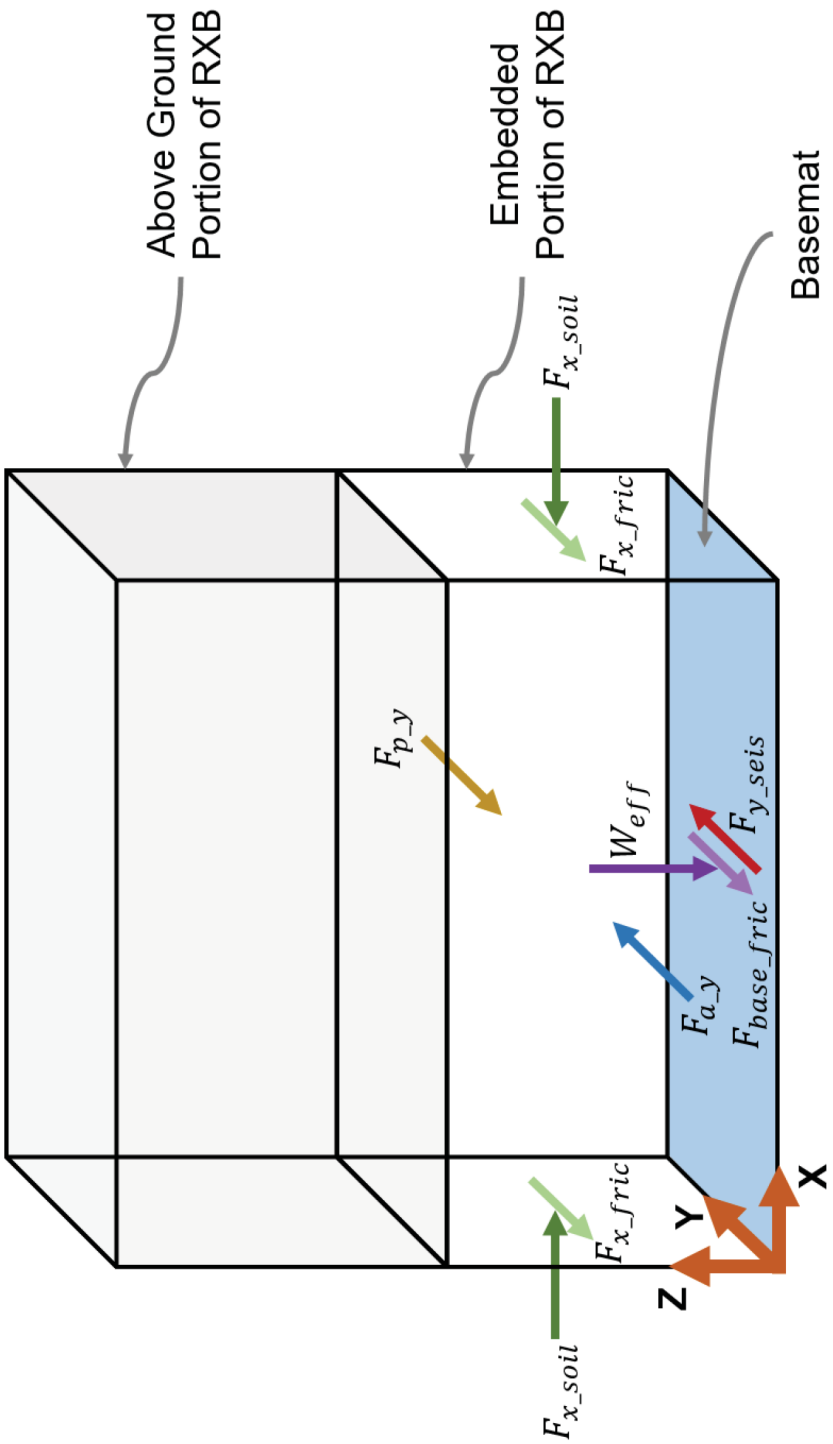


Figure 3.8.5-3: Illustration of Reactor Building Forces for the North-South (Y) Sliding Case

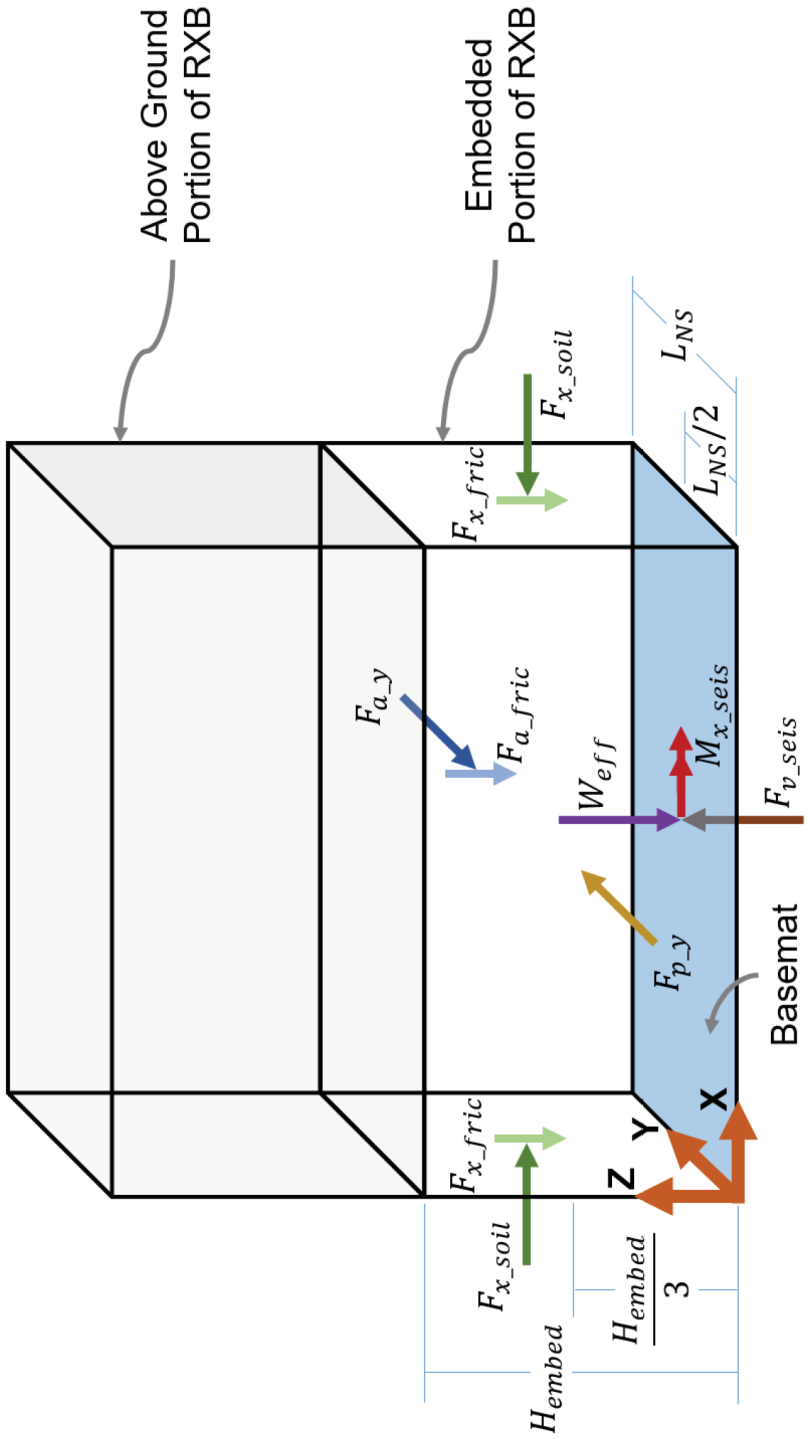


Figure 3.8.5-4: Illustration of Reactor Building Forces for the East-West (X) Overturning Case about the Southern Edge

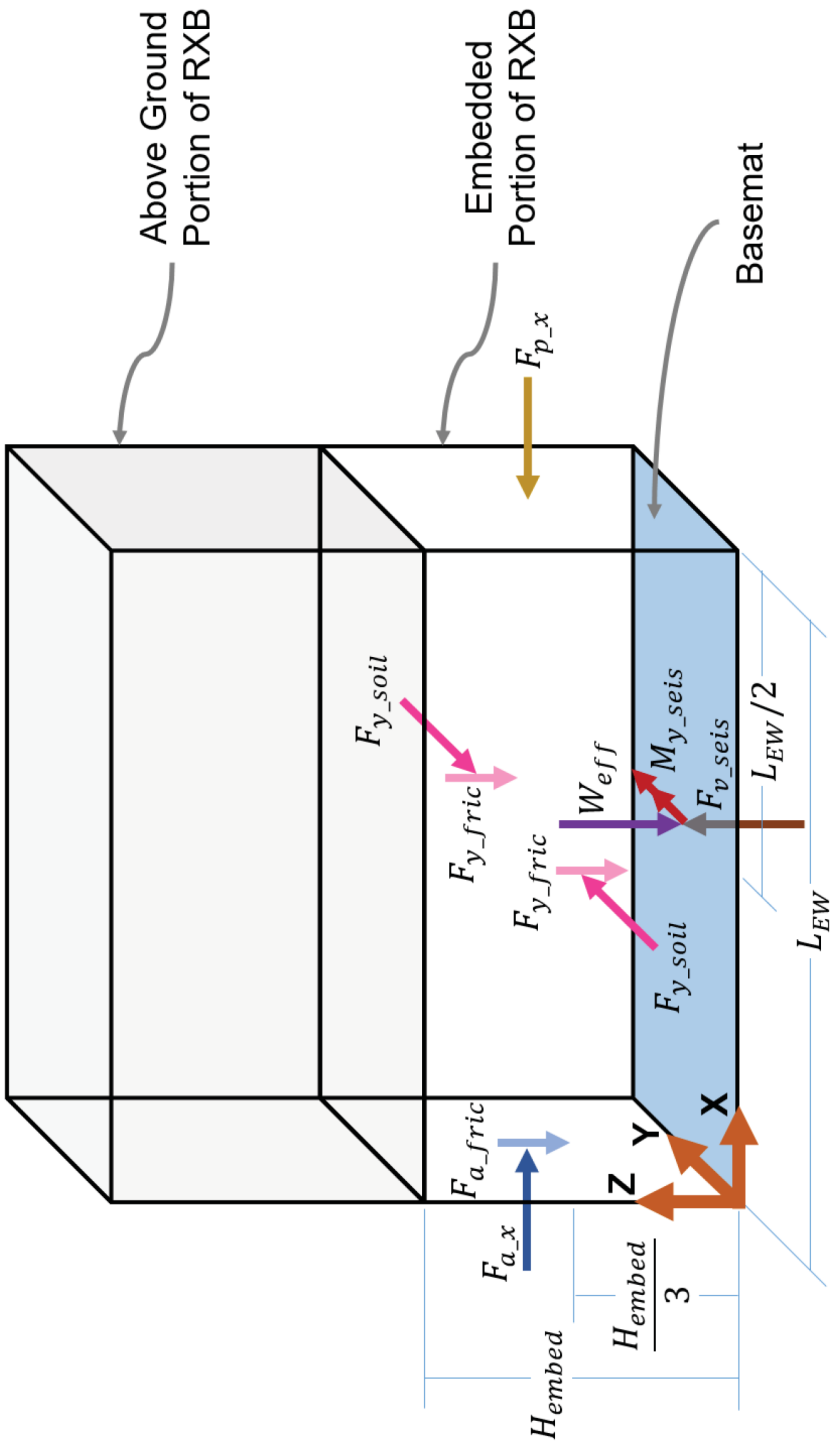
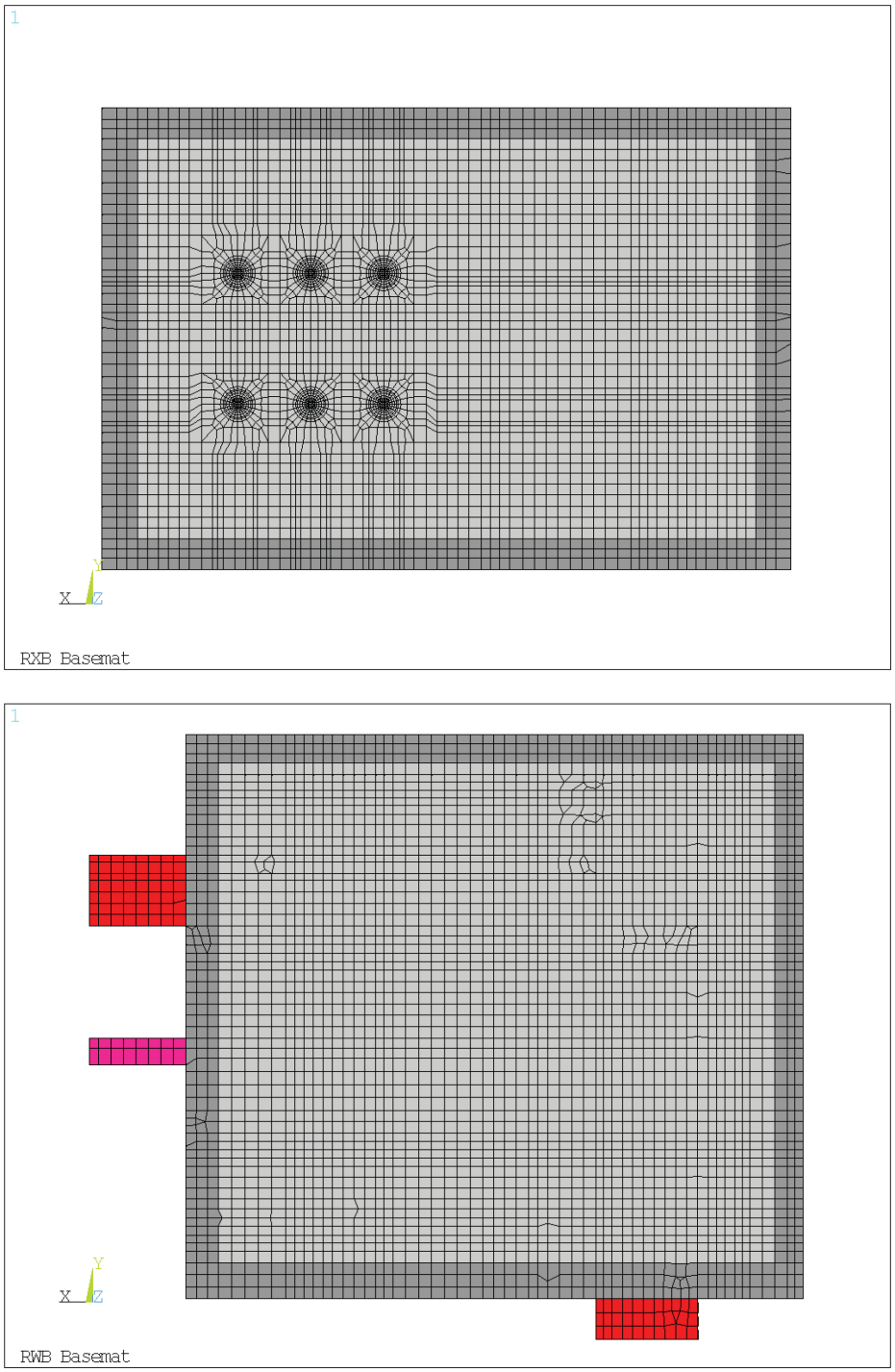


Figure 3.8.5-5: Illustration of Reactor Building Forces for the North-South (Y) Overturning Case about the Eastern Edge

Figure 3.8.5-6: Base Toe Areas (Dark Gray) Defined to Calculate the Toe Pressure for the Reactor Building (Top) and the Radioactive Waste Building (Bottom) Models



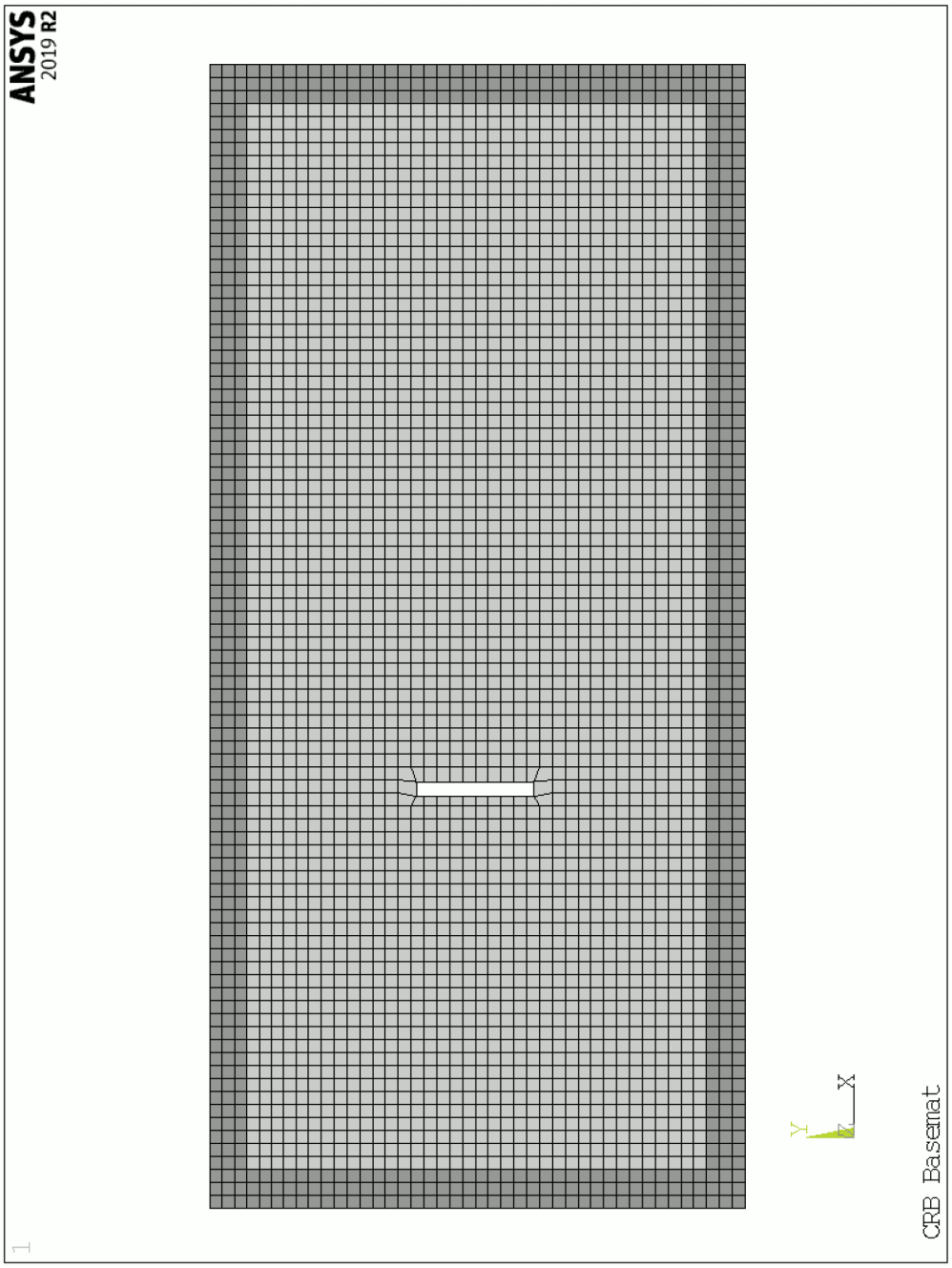


Figure 3.8.5-6a: Base Toe Nodes (Dark Gray) Defined to Calculate the Toe Pressure for the CRB Model

Figure 3.8.5-7: Layered Excavated Half Space Mesh Generation Process for DB Model. Surface Area Model (Top-Left), is Meshed with Plane182 Elements (Top-Right). The Meshed Area is Swept in Depth (Bottom-Left) and Finally the Elements inside Excavation Volume are Deleted (Bottom-Right)

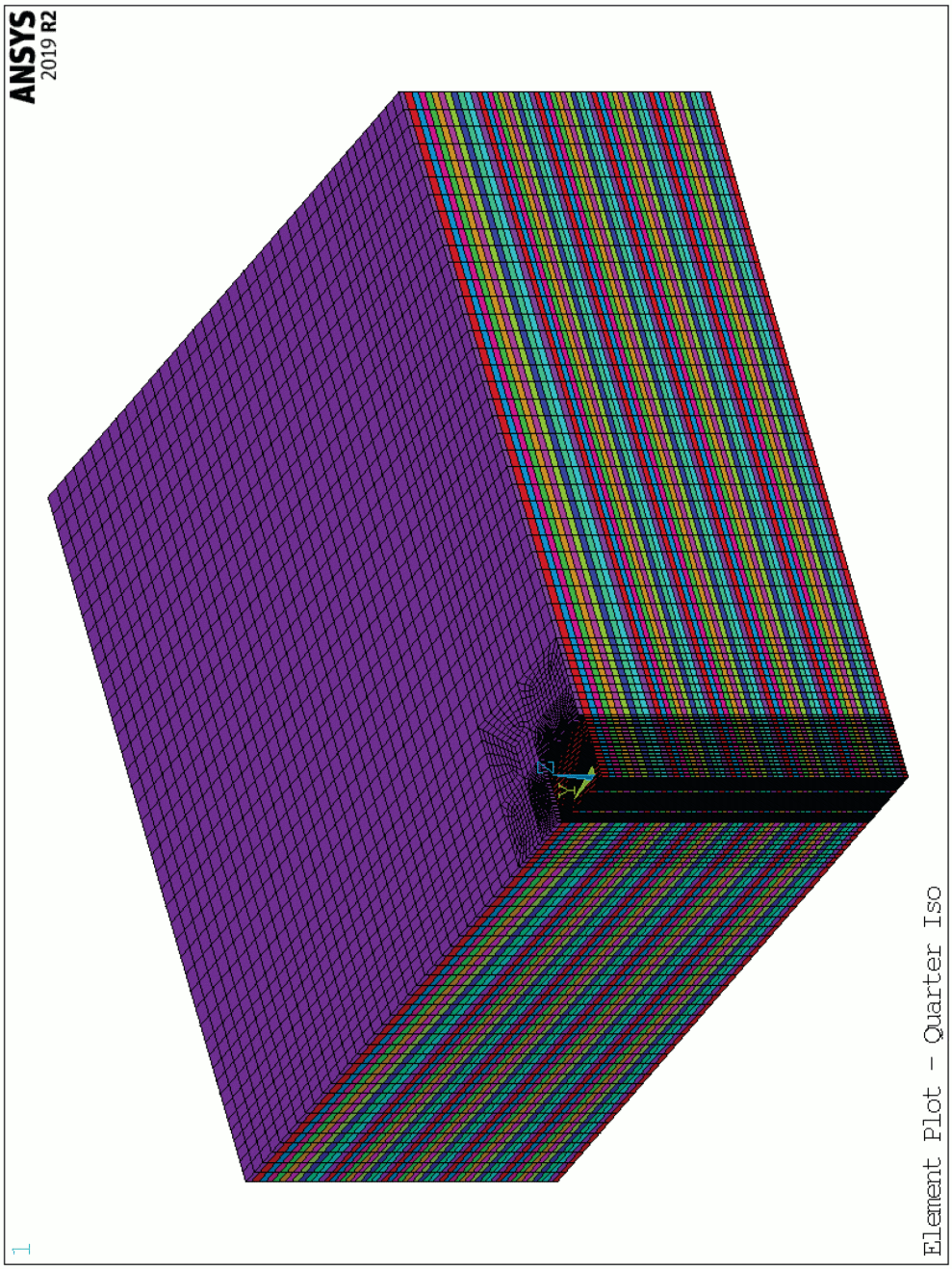


Figure 3.8.5-8: Swept Volume Mesh for One Quarter of the CRB Model, Showing the Soil Layers

Figure 3.8.5-9: Vertical Stress (SZ) Distribution on the Soil Layers under the Control Building Basemat (negative values represent compressive stress)

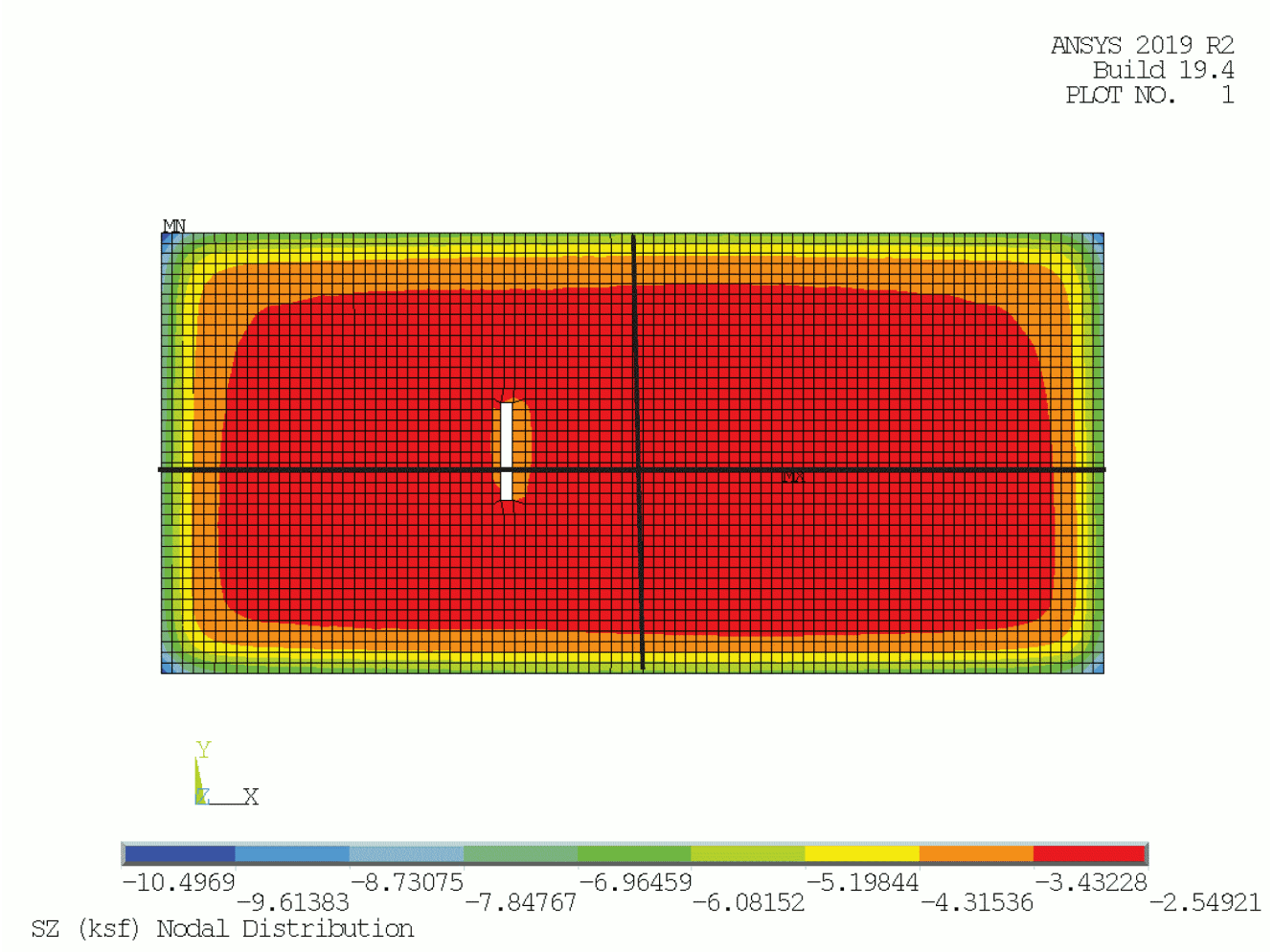
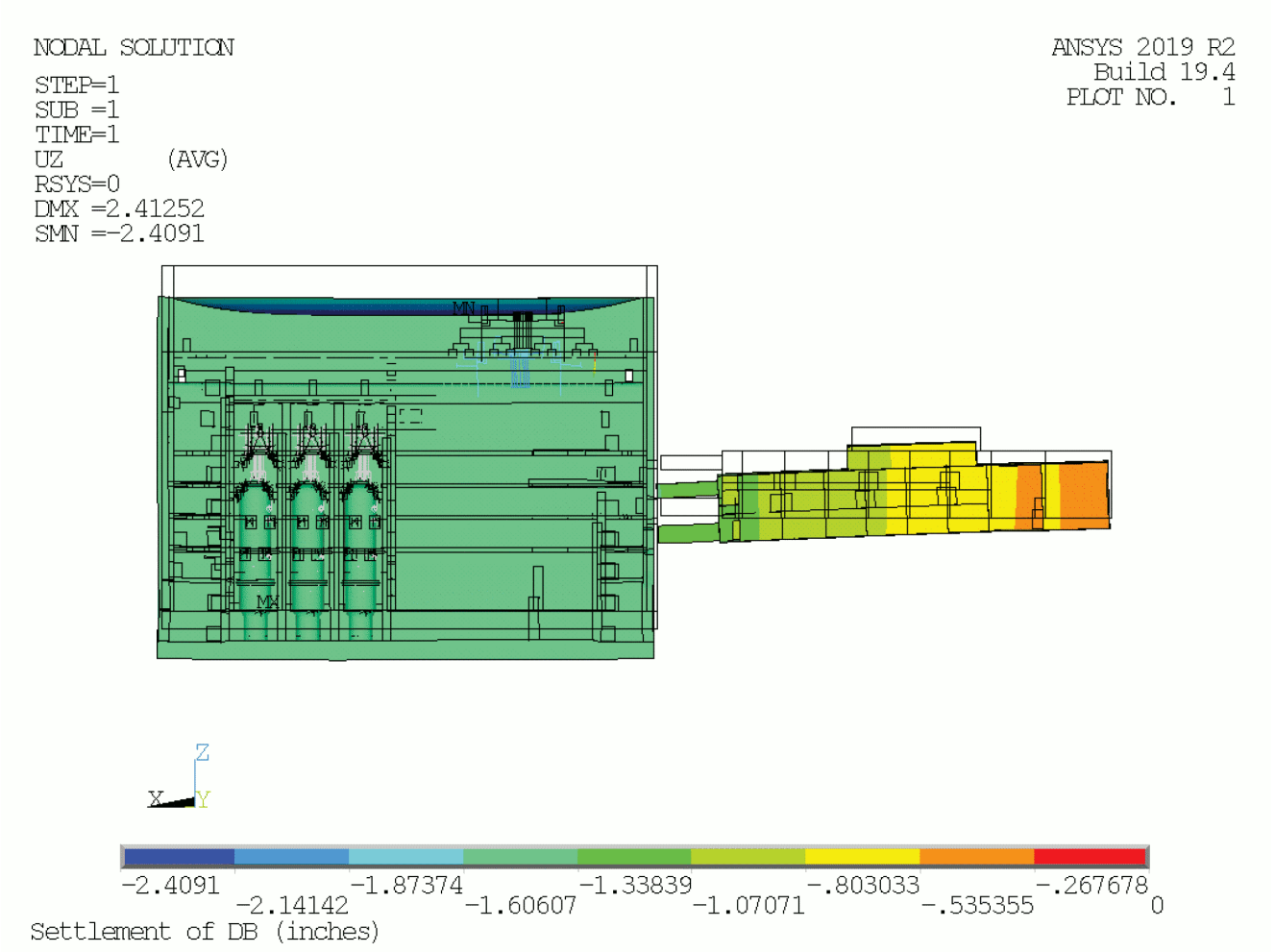
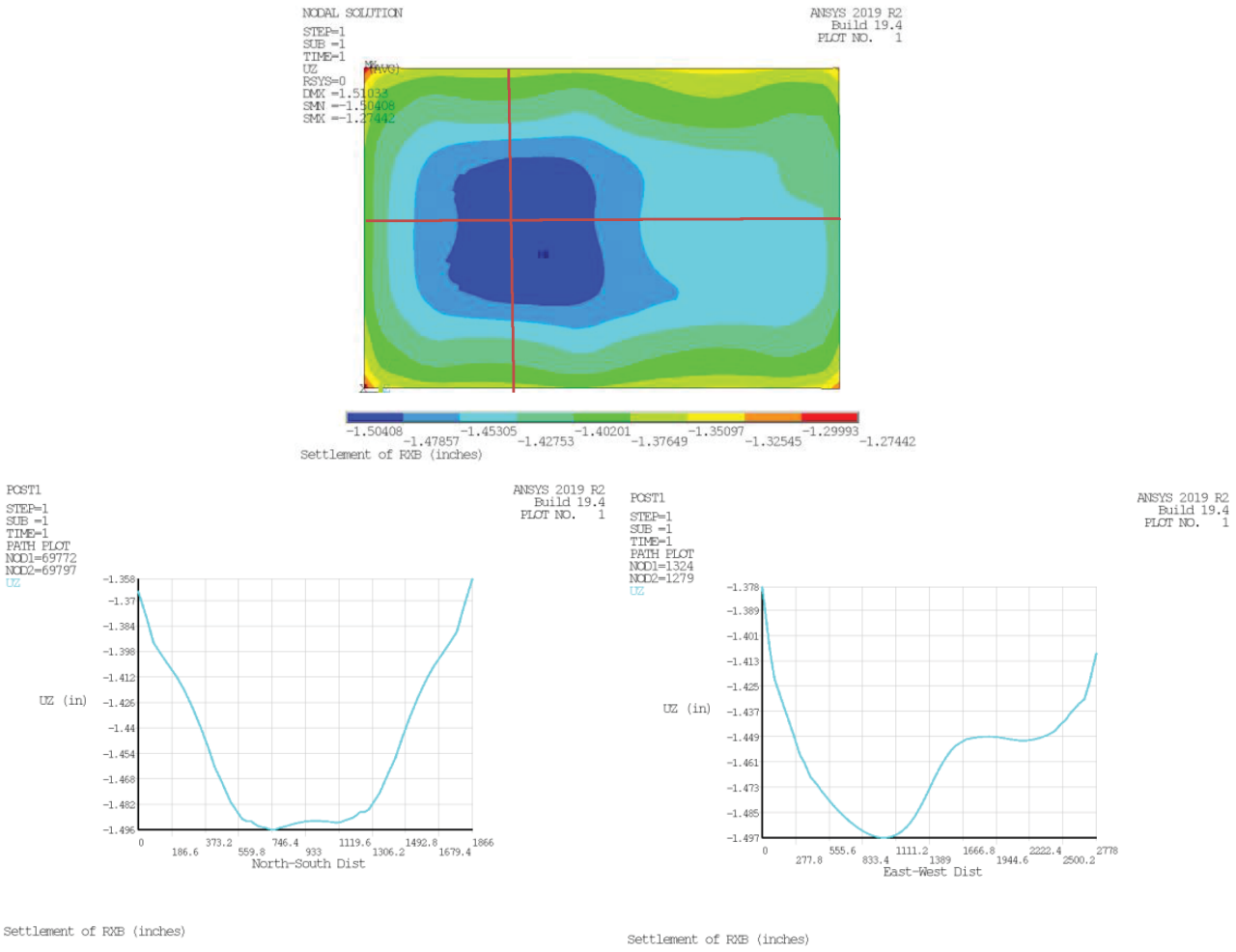


Figure 3.8.5-10: Vertical Displacement Field (UZ) of DB model



Footnote: The image is scaled to make the settlements visible with respect to the original locations which are marked by black lines. The presented values are in inches.

Figure 3.8.5-11: Vertical Displacement (UZ) Field on Reactor Building Base (Top) Together with the Variation of UZ along the Presented Paths on the Basemat: North-South Path (Bottom-Right), East-West Path (Bottom-Left)



Footnote: Presented values are in inches.

Figure 3.8.5-12: Vertical Displacement (UZ) Field on the Radioactive Waste Building Base (Top) Together with the Variation of UZ along the Presented Path on the Basemat

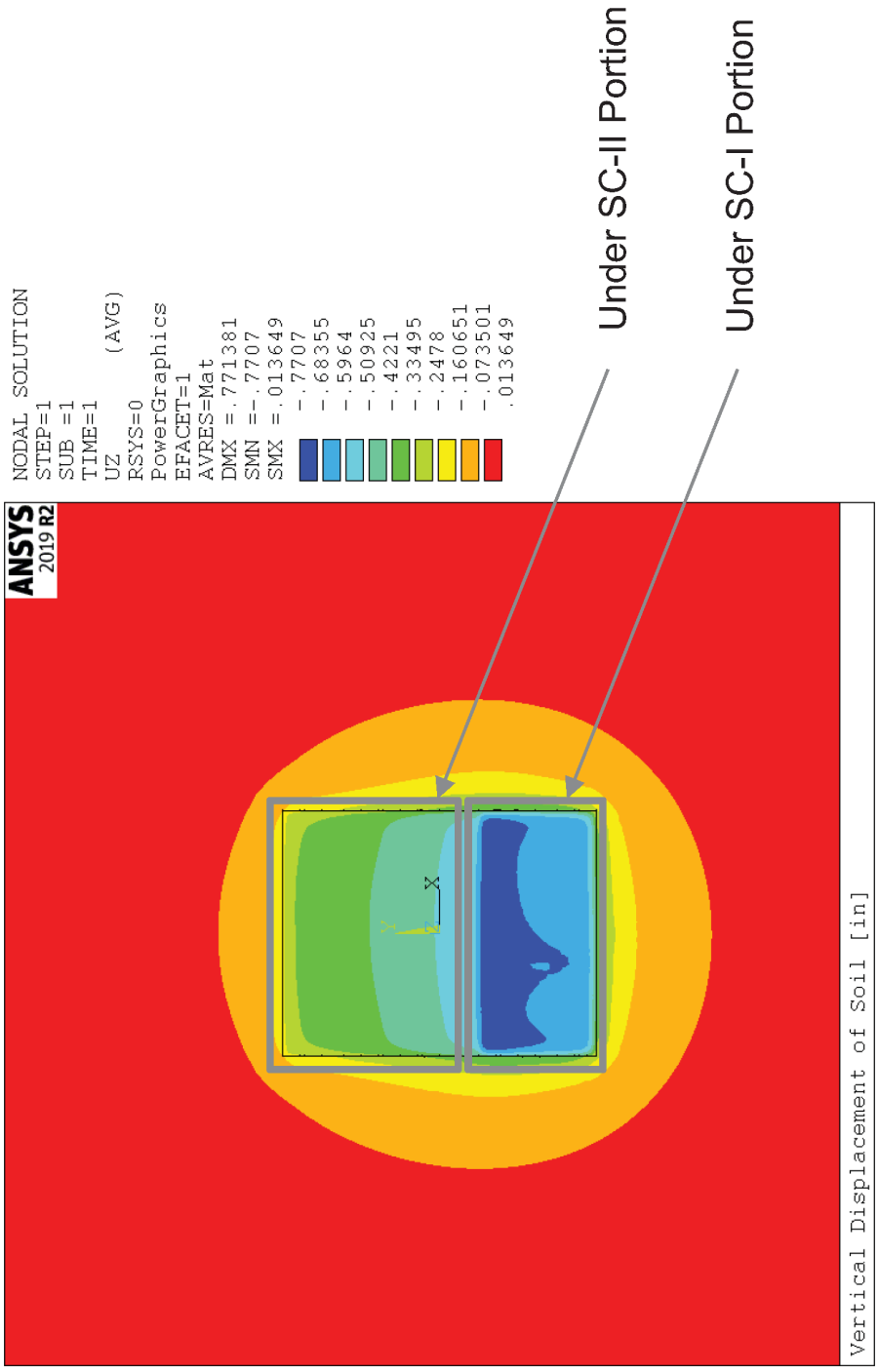


Figure 3.8.5-12a: Contour Plot of Vertical Displacement on the Top Soil Layer under the Control Building

Figure 3.8.5-13: Selected Nodes for Tabular Presentation of the Differential Settlement and Lateral Movement Values at the Base of RXB Model

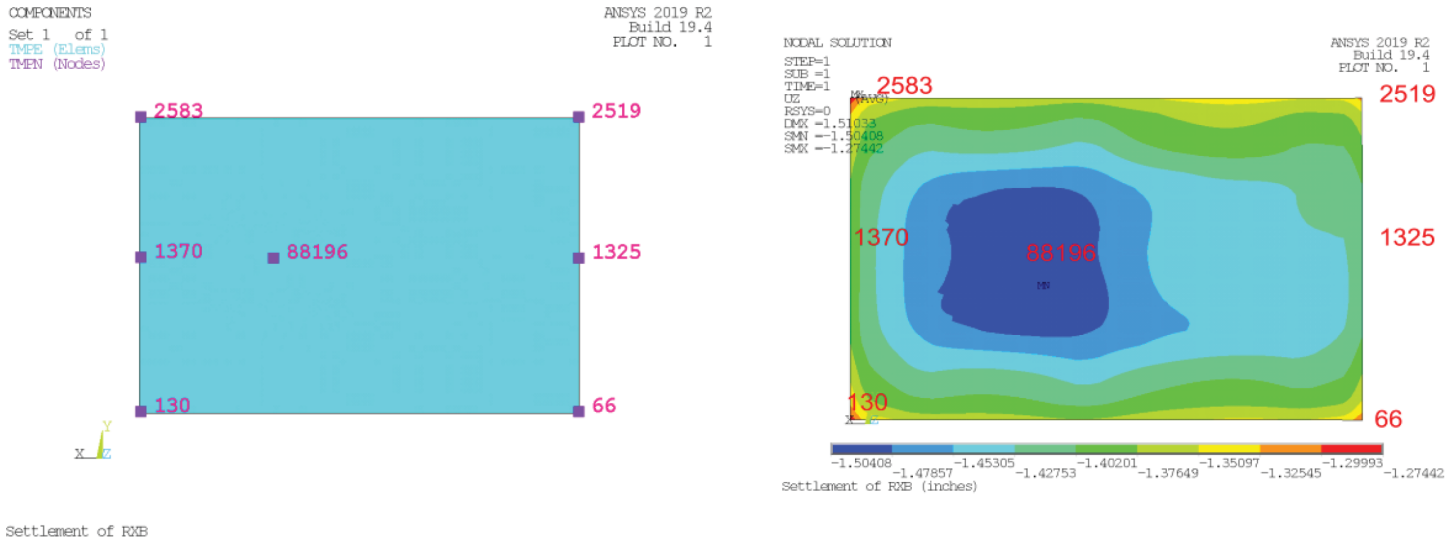
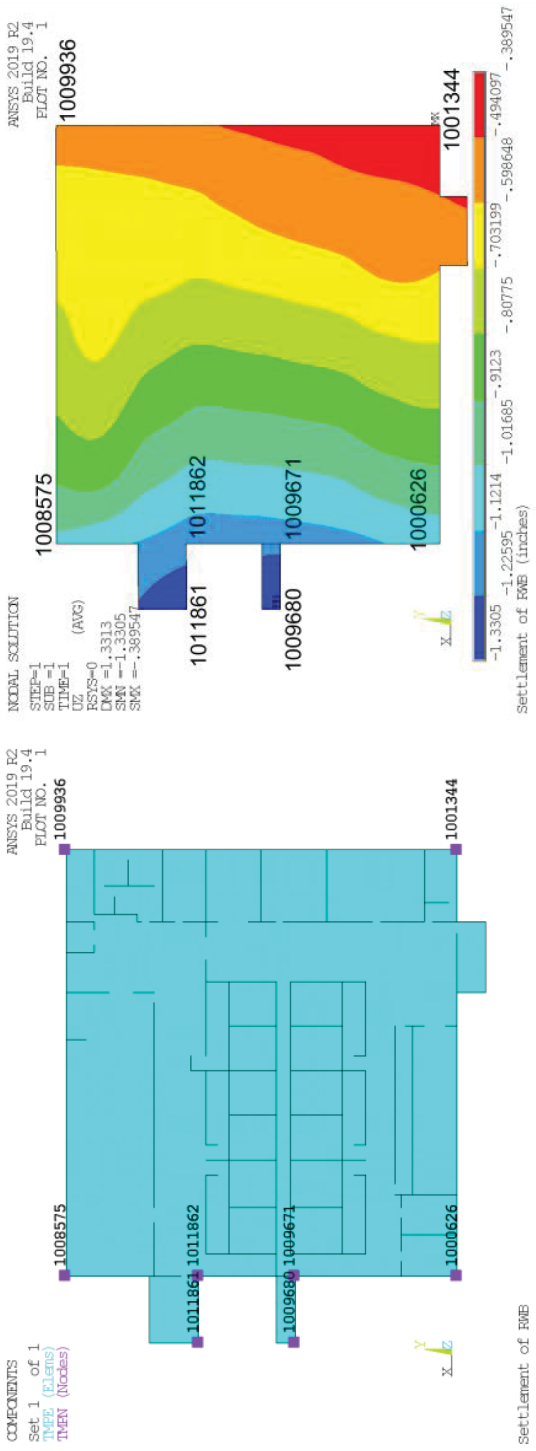


Figure 3.8.5-14: Selected Nodes for Tabular Presentation of the Differential Settlement and Lateral Movement Values at the Base of RWB Model



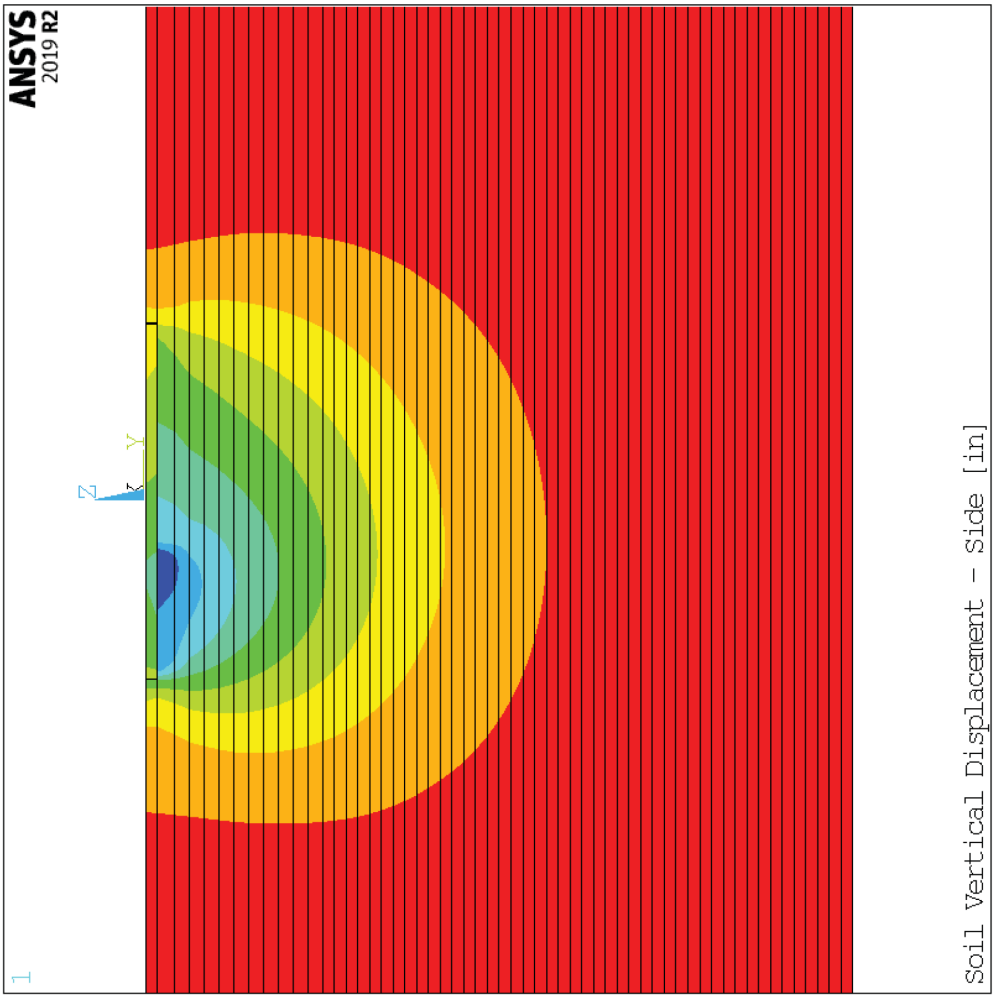


Figure 3.8.5-15: Vertical Displacement of Soil under the Control Building Basemats at Section Cut YY (North-South Path along the Centerline of the Control Building)

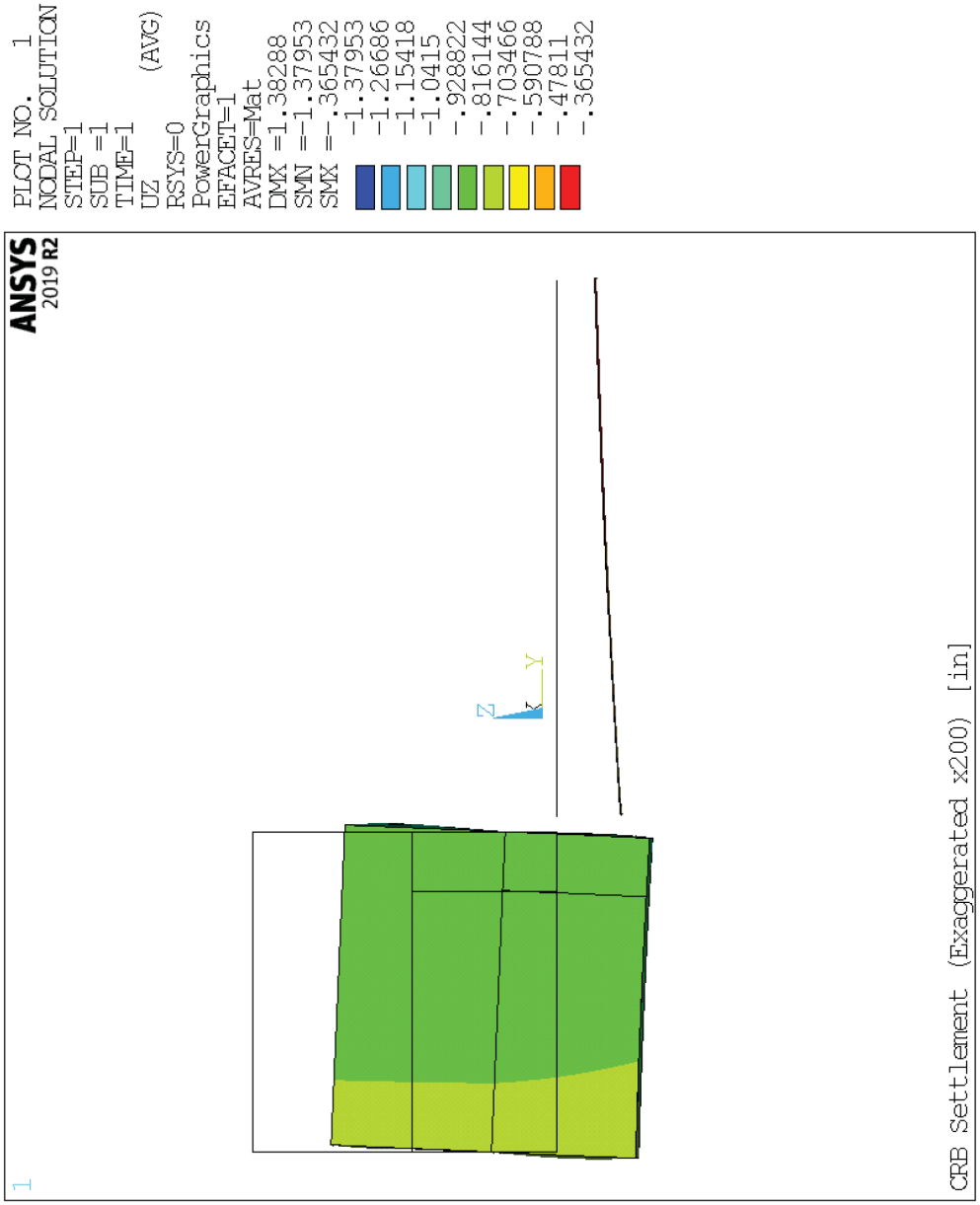
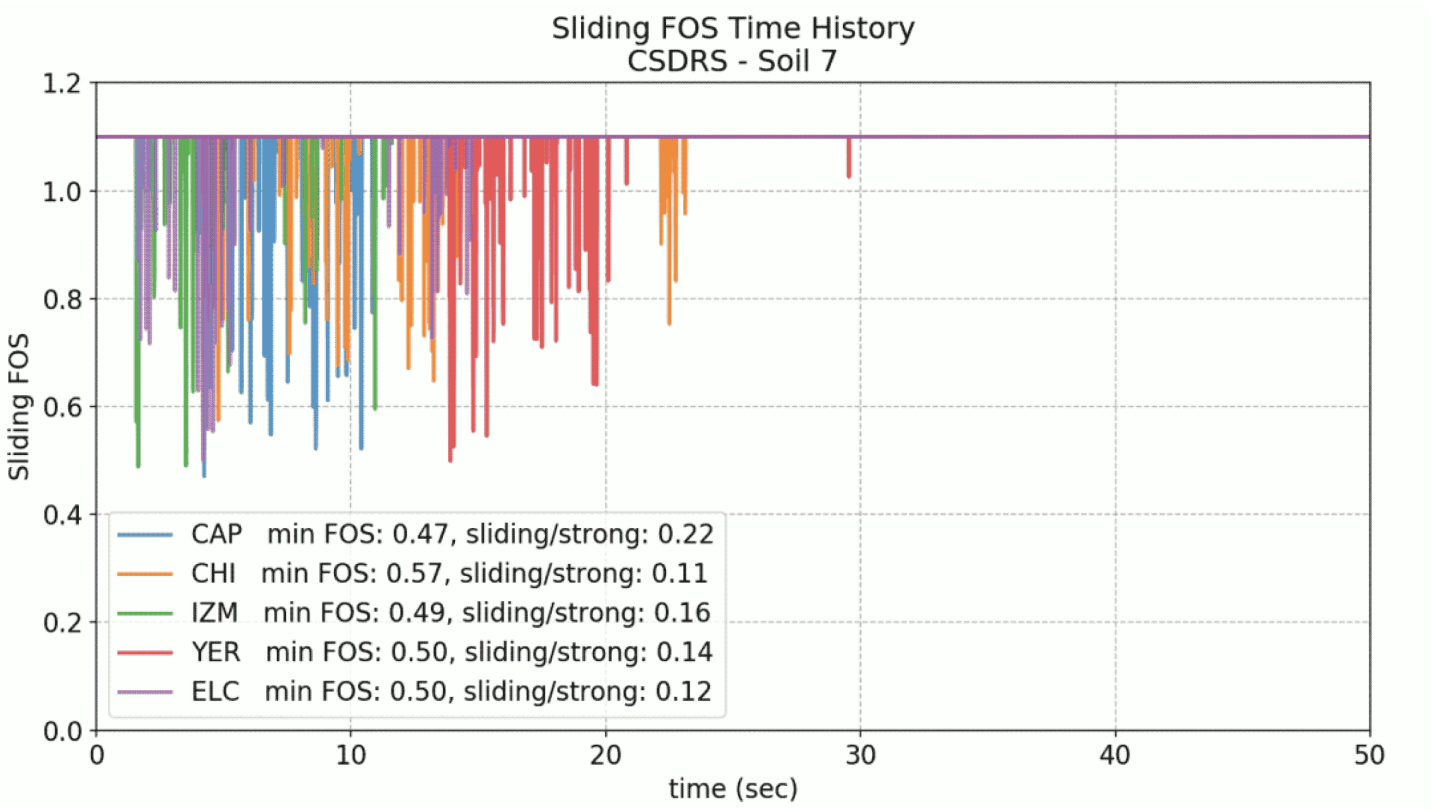


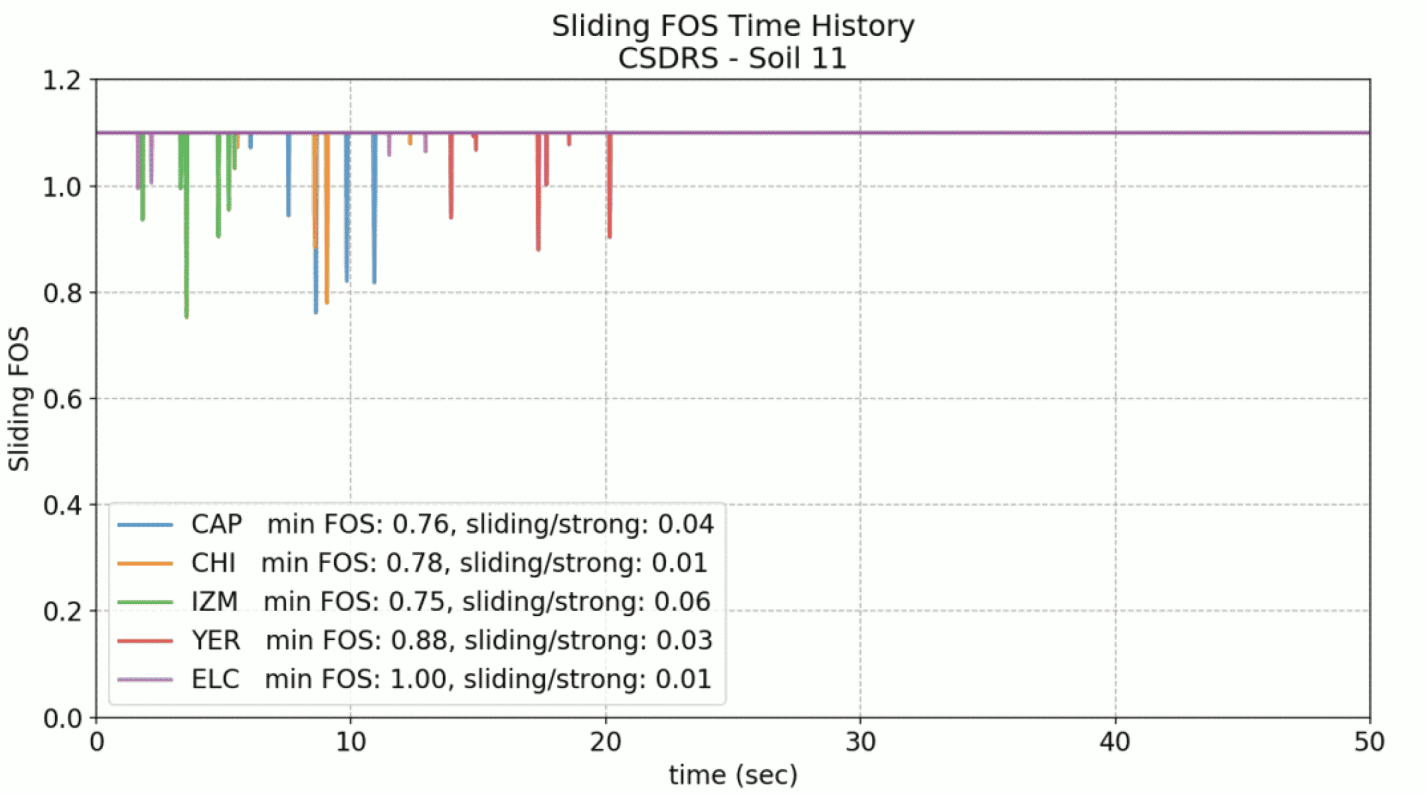
Figure 3.8.5-16: Exaggerated Settlement of the Control Building

Figure 3.8.5-16a: Control Building Sliding FOS Time Histories for Five Input Accelerations Used with Soil-7 Library



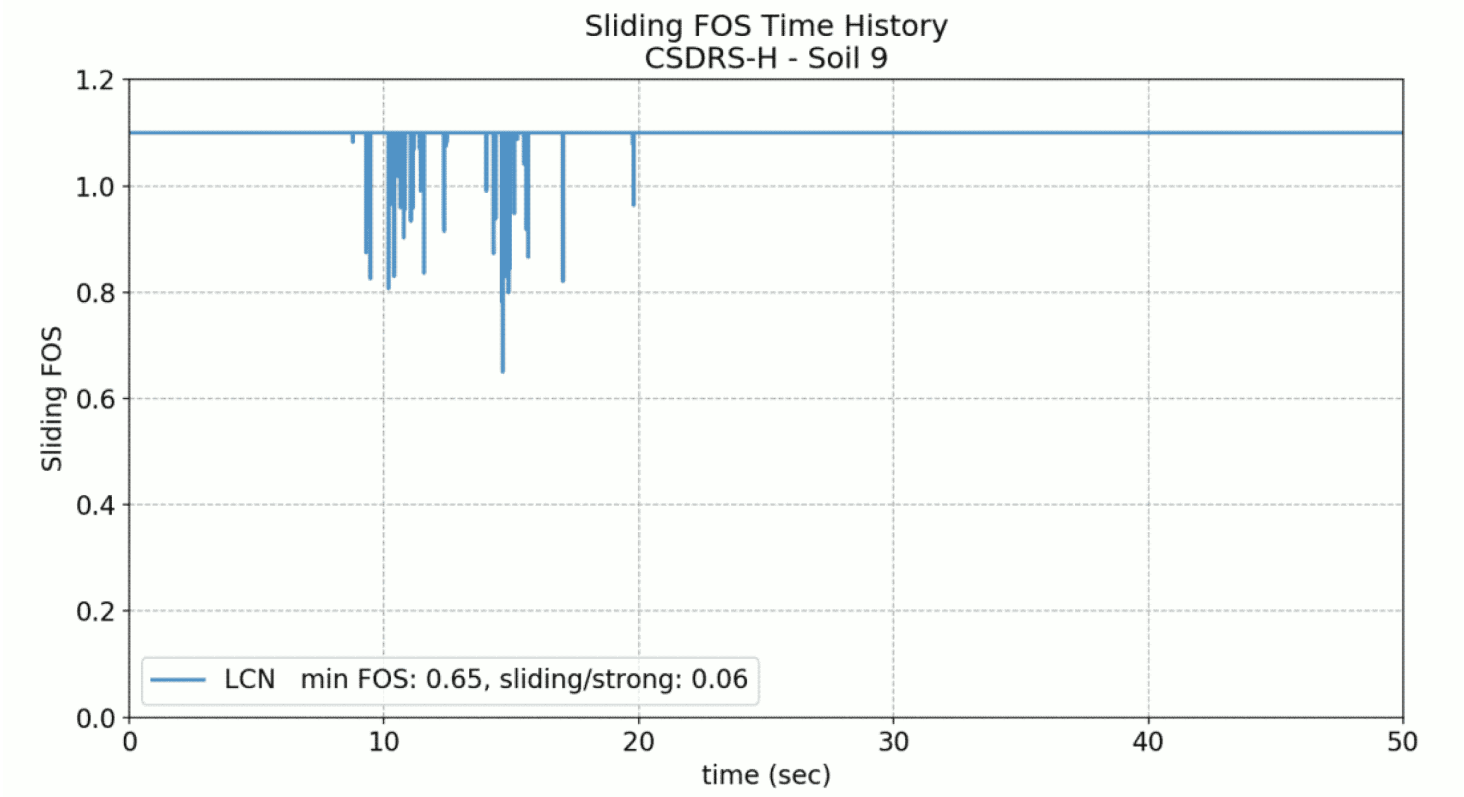
The maximum FOS value is capped at 1.1 to make the lower FOS values more visible. The legend for each event lists the minimum FOS value and the ratio of the total time.

Figure 3.8.5-16b: Control Building Sliding FOS Time Histories for Five Input Accelerations Used with Soil-11 Library



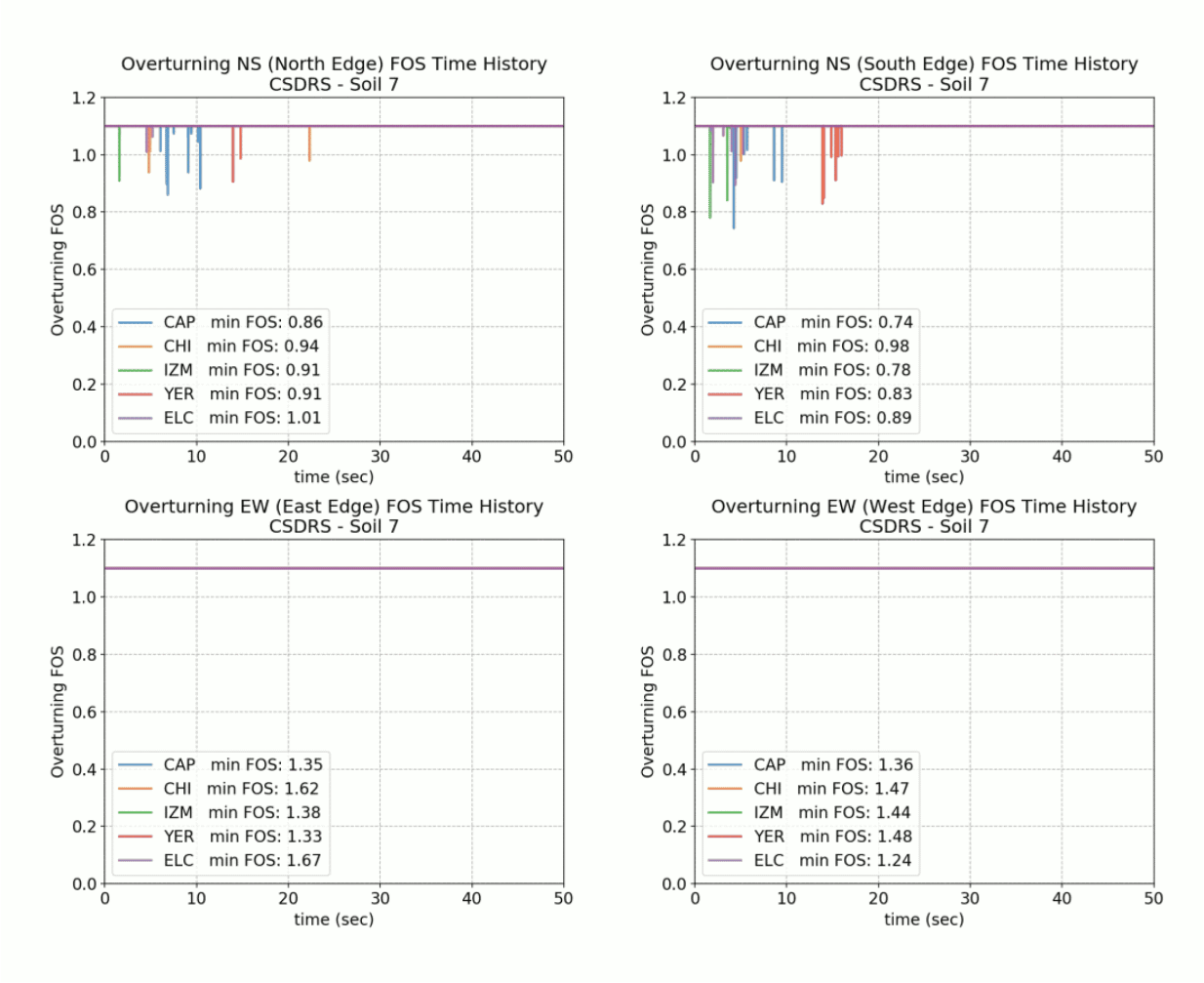
The maximum FOS value is capped at 1.1 to make the lower FOS values more visible. The legend for each event lists the minimum FOS value and the ratio of the total time with FOS less than 1.1 to the strong ground motion time as sliding/strong.

Figure 3.8.5-16c: Control Building Sliding FOS Time History for the Lucerne Input Accelerations Used with Soil-9 Library



The maximum FOS value is capped at 1.1 to make the lower FOS values more visible. The legend lists the minimum FOS value and the ratio of the total time with FOS less than 1.1 to the strong ground motion time as sliding/strong.

Figure 3.8.5-16d: Control Building Overturning FOS Time Histories for Five Input Accelerations Used with Soil-7 Library at Four Edge of the Basemat (North, South, East and West) that are Listed in the Titles of Each Sub-figure.



The maximum FOS value is capped at 1.1 to make the lower FOS values more visible.

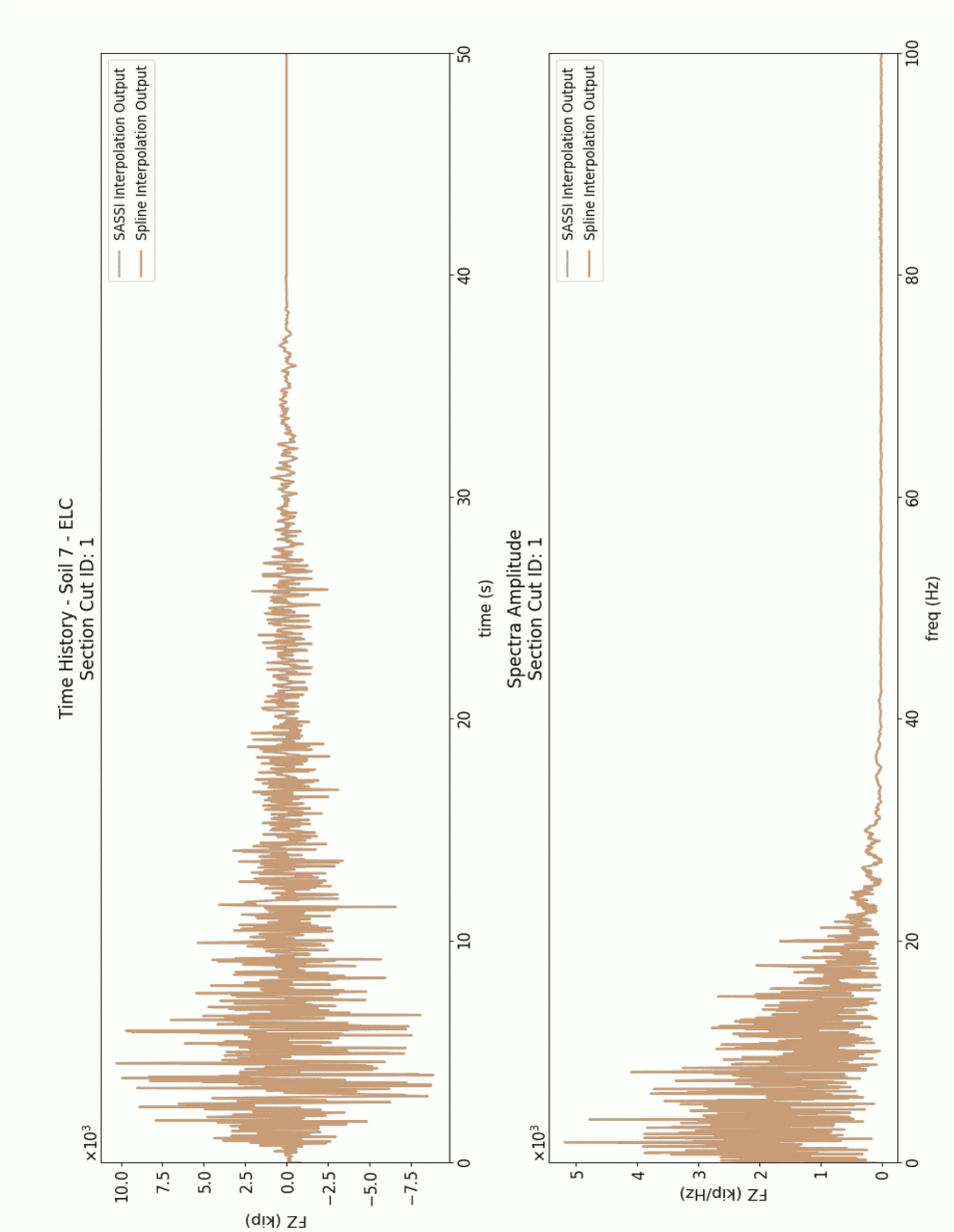


Figure 3.8.5-17: Time History and Fourier Amplitude Spectrum of the Vertical Seismic Base Reaction (FZ) Calculated at the Geometric Center of the Control Building Base with Soil-7 Library with ELC Event Input Acceleration Time History using SASSI and Spline Interpolation Methods for the Transfer Function

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

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

COL Item 3.9-1: An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific seismic analysis in accordance with Section 3.7.2. In addition to the requirements of Section 3.7, for sites where the high frequency portion of the site-specific spectrum is not bounded by the certified seismic design response spectra, the standard design of NuScale Power Module components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demand.

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

- GDC 1, for components designed, fabricated, erected, constructed, tested, and inspected in accordance with the requirements of applicable codes and standards commensurate with the importance of the safety-related functions to be performed. Compliance with GDC 1 is discussed in Section 3.1.
- GDC 2, for mechanical components of systems designed to withstand seismic events without loss of capability to perform their safety-related functions. Pursuant to GDC 2, mechanical components are designed to withstand the loads generated by natural phenomena as discussed Section 3.1.1.
- GDC 14, for the reactor coolant pressure boundary (RCPB) designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
- GDC 15, for mechanical components of the reactor coolant system (RCS) being designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).
- 10 CFR 50, Appendix B, Section III, for quality of design control. Section 17.5 satisfies the requirements of 10 CFR 50, Appendix B to ensure that structures, systems, and components (SSC) are designed, procured, fabricated, inspected, erected, and tested to standards commensurate with their contribution to plant safety.
- 10 CFR 50, Appendix S, for suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics.

3.9.1.1 Design Transients

Design transients define thermal-hydraulic conditions (i.e., pressure, temperature, and flow) for the NuScale Power Module (NPM). Bounding thermal-hydraulic design transients are defined for components of the RCPB. The number of events for each design transient is based on the 60-year design life of the NPM. Transients are defined for safety-related equipment design purposes and provide a bounding representation of NPM operation.

Operating condition categories, as defined in the ASME BPVC, Section III (Reference 3.9-1), apply to Class 1, 2, and 3 components:

- **ASME Service Level A**

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

- **ASME Service Level B**

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

- **ASME Service Level C**

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

- **ASME Service Level D**

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

- **Test Conditions**

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

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

Service Level A and B transients are representative of events that are expected to occur during plant operation. These transients are frequent enough to be evaluated for component cyclic behavior and equipment fatigue life, and analyzed conditions are based on a conservative estimate of the frequency and magnitude of temperature and pressure changes. When used as a basis for component fatigue evaluation, bounding transients provide confidence that the component is

appropriate for its application over the design life of the plant. Service Level C and D conditions are not typically included in fatigue evaluations in accordance with the ASME BPVC, Section III (Reference 3.9-1). However, for select component and transient combinations, Service Level C events may be evaluated against Level B stress limits. This selection is made either because the event contains significant stress cycles or the plant-wide transient is considered a normal service level for that component.

The following sections describe assumptions used in thermal-hydraulic analysis for each Service Level.

3.9.1.1.1 Service Level A Conditions

Service Level A Transient 1 - Reactor Heatup to Hot Shutdown

This transient covers the heatup and pressurization from transition mode to a hot shutdown state. The event begins with a depressurized NPM and containment filled with water at least above the minimum level required for passive cooling.

Containment is pressurized to at or above the minimum pressure required to begin the containment drain process. The RCS is pressurized equivalently by adding nitrogen gas to the pressurizer (PZR). Once pressurizer heaters are actuated to increase RCS pressure, nitrogen is removed through the pressurizer high point degasification line and is replaced with steam. During heatup, the module heatup system heater increases the RCS temperature to the minimum temperature for criticality. Once the steam generator system (SGS) level is boiled off within an acceptable mass inventory, at least one steam generator (SG) is isolated to ensure a single decay heat removal system (DHRS) train is operable. During the temperature increase, pressurizer heaters continuously add energy to the RCS and increase RCS pressure to reach normal operating pressure by the time the RCS reaches the minimum temperature for criticality.

Service Level A Transient 2 - Reactor Cooldown from Hot Shutdown

This transient encompasses cooling from a hot shutdown state to transition mode and is generally the reverse of the reactor heatup to hot shutdown. One SG train is isolated and RCS temperature is continually reduced by controlling the feedwater flow rate. Steam and feedwater flow rates are controlled to keep the cooling rate below allowable cooldown rates for the RCS and pressurizer regions. Boron concentration is increased using the chemical and volume control system (CVCS) to reach the cold shutdown concentration. The RCS makeup is added to account for coolant shrinkage. Once the RCS is cooled to an allowable temperature for containment flooding, the containment flooding and drain system is used to add pool water to containment. This water cools the containment vessel (CNV) and reactor pressure vessel (RPV) and ensures that decay heat can be removed from the RCS without feedwater flow. Once the pressurizer steam bubble is collapsed, nitrogen gas is added to the pressurizer to control primary pressure.

Service Level A Transient 3 - Power Ascent from Hot Shutdown

This transient covers the power ascent from a hot shutdown state to 20 percent of full reactor power, at which point the turbine may be brought online for power generation. Throughout this transient, the steam and feedwater flow rates through the unisolated SG train are controlled to match the demanded load ramp, which is specified as a maximum rate limit of 0.5 percent of full power per minute. During this power transient, the secondary flow in the SG may experience oscillations. The feedwater temperature remains at the condenser hotwell temperature as the feedwater heaters are unavailable.

Service Level A Transient 4 - Power Descent to Hot Shutdown

This transient occurs as the reverse of the power ascent from hot shutdown. The transient covers reactor conditions that span from the power level at which the power descent reactor trip occurs to a hot shutdown state. Since the turbine is offline, one SG is placed in isolation and steam from the unisolated SG train produced by cooling the RCS is diverted through the turbine bypass valve. Feedwater heating is not available when the turbine is offline and after the turbine trips; therefore, the feedwater temperature is equal to the condenser hotwell temperature.

Service Level A Transient 5 - Load Following

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

Service Level A Transient 6 - Load Regulation

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

Service Level A Transient 7 - Steady State Fluctuations

While operating at a steady load, RCS temperature and pressure may fluctuate. These fluctuations could be due to control system setpoints, small power variations, operational dead band, measurement uncertainties, or other unplanned variations. The full-power, normal operating bands for RCS average temperature and pressurizer pressure are assumed to be ±1 degree

Fahrenheit and ±10 psia. There may also be flow oscillations on the secondary side caused by similar parameter variations.

Service Level A Transient 8 - Load Ramp Increase

When the turbine is online for power generation, the reactor can provide a load increase at a rate of 5 percent of full power per minute over the power generation range from 20 to 100 percent of full power. A rate of 5 percent of full power per minute is an upper bound on the load increase rate for power maneuvers and is consistent with other pressurized water reactor (PWR) designs. Extraction steam is available to the feedwater heaters, and feedwater temperature increases as power level increases.

Service Level A Transient 9 - Load Ramp Decrease

When the turbine is online within the power generation range, the reactor can provide a load decrease at a rate of 5 percent of full power per minute over the power generation range of 20 to 100 percent of full power. A rate of 5 percent of full power per minute is an upper bound on the load decrease rate for power maneuvers and is consistent with other PWR designs. Feedwater temperature decreases due to less feedwater heating as the power level decreases.

Service Level A Transient 10 - Step Load Increase

When the turbine is online within the power generation range, the ASME components are designed to withstand the cycles associated with a 10 percent of full power step load increase. This transient could occur due to a disruption in the electrical power grid. As load is increased, the imbalance between load and core power causes the RCS temperature and pressure to decrease. The pressurizer heaters respond to a pressure decrease by increasing proportional heater output and energizing the backup heaters.

Service Level A Transient 11 - Step Load Decrease

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

Service Level A Transient 12 - Large Step Load Decrease

This transient occurs when there is a large decrease in the demanded load by the power grid from full power down to 20 percent of full power. When the load decreases, steam pressure increases and steam flow rate decreases. The RCS temperature and pressure increase due to the decrease in secondary heat removal. Some steam likely needs to bypass the turbine to prevent a reactor trip on high pressurizer pressure or level. The bypass load is ramped down at about 5 percent power per minute to give the reactor time to reduce

power. Two control signals detect a large step load decrease and regulate bypass steam flow. An error signal between the demanded and actual turbine load determines the need for steam bypass, and a signal provides the expected bypass valve position as a function of demanded load.

Service Level A Transient 13 - Refueling

During refueling, the containment vessel flange and reactor vessel flange are opened and the upper portion of the NPM is lifted away from the lower portion, exposing the reactor core for refueling. This operation takes place in the refueling pool. There is a minimal thermal cycle on the RPV as the flanges are unbolted and pool water intermixes. For module handling operations to begin, the RCS must be in transition mode. There are minimal thermal cycles for cold unbolting and re-bolting of the RPV flanges for reactor startup, because the RPV temperature is near equilibrium with the surrounding reactor pool water following the duration of a refueling outage.

Service Level A Transient 14 - Reactor Coolant System Makeup

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

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

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

This transient occurs while leaving a hold at hot shutdown. Normally, continuous feedwater and steam flows are used for SG inventory control when transitioning to and from hot shutdown. If there is an extended hold at hot shutdown, decay heat generation may drop below the minimum capability of secondary heat removal systems, thus securing the feedwater and steam flows. Breaking the hold initiates the feedwater flow again, which provides cold feedwater to components that reached equilibrium with the hot RCS. The main steam isolation valves (MSIVs) are opened to the desired position and the feedwater and steam flows are operated continuously to achieve the SG inventory objective.

During a hold at hot shutdown with little decay heat, the CVCS module heatup system may be used to maintain primary coolant temperature, which reduces the need to turn off feedwater flow and reduces thermal cycles on RPV components.

During a reactor cooldown due to DHRS actuation, cooldown may be interrupted by re-establishing feedwater flow to the SG. The feedwater flow is a continuous flow, providing inventory to the SG. The thermal impact of reestablishing feedwater flow is less severe for components such as the feedwater plenum, because the plenum is at a cooler temperature due to the flow of DHRS condensate during reactor cooldown.

Service Level A Transient 16 - High-Point Degasification

Transients for the high-point degasification line are the normal operation venting of the pressurizer and shutdown degasification of the RCS. The addition of gaseous nitrogen to the pressurizer through the degas line is included in Service Level A Transient 1, Reactor Heatup to Hot Shutdown. The normal operation venting transient involves periodically opening valves in the high-point degasification line to remove non-condensable gases that collected in the vapor space of the pressurizer. Prior to and during shutdown operations, the high-point degasification line is used to mechanically degas the RCS to remove noncondensable gases from the pressurizer vapor space and to dilute the concentration of hydrogen in the reactor coolant by venting and providing makeup from the CVCS.

Service Level A Transient 17 - Containment Evacuation

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

Service Level A Transient 18 - Containment Flooding and Drain

The containment flooding and drain system connects to a CNV nozzle with piping extending from the top head of the CNV to the bottom of containment. The piping is used to add and remove water from the CNV. This transient is split into two events: containment flooding operations after shutdown and containment drain operations prior to startup. After shutdown, the containment flooding and drain containment isolation valves are opened and the pump

transfers water from the reactor pool to the CNV. Prior to NPM startup, the containment flooding and drain system containment isolation valves are opened and containment is pressurized through the CES penetration to the minimum pressure required, to provide adequate net positive suction head to the pump, which helps drain the CNV of water.

Service Level A Transient 19 - Secondary Leakage Tests

A leakage test is performed after each opening of the secondary systems of the NPM. The secondary side consists of the main steam (MS), feedwater (FW), and DHRS lines inboard of the secondary containment isolation valves as well as the SGs and associated plena. The secondary leakage test initiates from a hot shutdown condition. During this leakage test secondary fluid pressure is raised to the maximum steam pressure across the range of power generation. While the secondary side is held at the maximum steam pressure, the system is checked for leaks according to methods in Section XI of the ASME Code (Reference 3.9-2).

Service Level A Transient 20 - Initial Test Program

The NPM undergoes pre-service start-up tests prior to initial plant start-up after completion of construction. Some tests occur at elevated temperatures and pressures prior to loading nuclear fuel. For these pre-service hot functional tests, the module heatup system is used to heat the primary fluid to the hot shutdown condition and produce secondary steam. Pre-service tests are independent of other operating transients as they occur prior to operation. Some tests may be performed after initial criticality is achieved.

3.9.1.1.2 Service Level B Conditions

Service Level B Transient 1 - Decrease in Feedwater Temperature

A decrease in feedwater temperature could occur due to malfunctions in the secondary side system. The bounding malfunction is loss of feedwater heating. Such a failure at full power lowers feedwater temperature, which reduces the RCS temperature and adds reactivity due to the negative moderator temperature coefficient. The secondary control system compensates for lower feedwater temperature by adjusting feedwater flow rate to reach the load demand setpoint. Reactivity feedback allows reactor power to re-adjust to match demanded load.

Service Level B Transient 2 - Increase in Secondary Flow

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

feedwater pump speed increases the feedwater pressure. The increase in feedwater flow rate may reduce feedwater temperature.

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

There is a control system response for the secondary side. Steam superheat falls below the setpoint and the SG load is larger than the setpoint. The feedwater regulating valve and turbine governor valve both close to try to match the steam pressure and load setpoints leading to a turbine trip due to low superheat. A reactor trip occurs on low superheat, low pressurizer level, low steam pressure, or high reactor power. When the turbine trips, feedwater heating is lost and feedwater temperature matches the temperature of the condenser hotwell.

Service Level B Transient 3 - Turbine Trip without Bypass

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

The RCS pressure and temperature increase due to the loss of heat removal, and the pressurizer level rises due to expanding RCS fluid. The reactor trips and actuates both trains of the DHRS to remove decay heat and cool the RCS.

Service Level B Transient 4 - Turbine Trip with Bypass

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

Service Level B Transient 5 - Loss of Normal Alternating Current Power

A loss of normal AC power consists of a loss of AC power with no credit taken for the backup power supply system or operation with an NPM in island mode. Under these circumstances the reactor trips, containment isolation valves fail closed, and the DHRS actuation valves fail open. The module reaches a safe shutdown state by dissipating heat through the DHRS condensers. Batteries supply power to the four emergency core cooling system (ECCS) valves (two

reactor vent valves (RVVs) and two reactor recirculation valves (RRVs)) that hold the valves closed. Once battery power is supplied to the ECCS valves a 24 hour timer begins. If there is no operator action or automatic actuation of the valves, after 24 hours battery power is removed and RVVs and RRVs fail open. Actuation of the ECCS establishes a two-phase natural circulation loop. Steam generated in the RPV exits through the RVVs and condenses on the walls of the CNV. The condensed water returns to the RPV through the RRVs. Coincident losses of the DC power systems: the augmented DC power system (EDAS) or the normal DC power system (EDNS), as well as delays in module protection system (MPS) actuations, are considered to determine bounding pressure and temperature responses for mechanical design.

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

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

Both trains of the DHRS are actuated. The DHRS removes heat through the two SG trains and rejects the heat to the reactor pool. Components of the DHRS are sized to remove decay heat and cool the RCS.

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

Inadvertent operation of the DHRS could occur in two ways. The first is inadvertent opening of one of the DHRS actuation valves. Opening an actuation valve allows flow between the DHRS condenser and the steam line as steam and feedwater pressures equalize. Initial pressure equalization in the secondary side disrupts the primary temperature. Inadvertent opening of the single DHRS valve would result in a reactor trip and eventual actuation of the second DHRS train. The second way to inadvertent DHRS actuation is by the MPS sending a signal to actuate the DHRS by closing the MSIVs and feedwater isolation valves (FWIVs) and opening the DHRS actuation valves, which results in the full-power operation of both trains of the DHRS. The DHRS actuation signal causes a reactor trip. The RSVs do not lift for either occurrence.

Service Level B Transient 8 - Reactor Trip from Full Power

A reactor trip from full power could be caused by multiple spurious sensor signals to the MPS, or a spurious trip signal from the MPS, or miscellaneous failures that cause a reactor trip setpoint to be reached, and are not already included in other transients. Once the trip begins, control rods drop into the core to take the core subcritical, reducing core thermal power to decay heat and causing hot and cold RCS temperatures to converge close to the average RCS temperature. Cooling is then initiated by one of two methods: either normal feedwater, or actuating the DHRS. If the DHRS is actuated, then a

containment isolation signal may also be generated. When circulating feedwater through the SGs, the steam produced is directed through the turbine bypass valve to the condenser. Steam and feedwater flow rates are controlled to keep the cooling rate below allowable cooldown rates for the RCS and pressurizer regions. This transient ends once the reactor reaches the minimum hot shutdown temperature. Cooldown is accounted for in the cycles of the cooldown from hot shutdown transient. If the DHRS is actuated for a more severe failure, heat is removed through the DHRS condenser to the pool. An eight-hour ECCS actuation timer starts at an automatic or manual reactor trip and may be bypassed after completing a calculation demonstrating margin to subcriticality at cold temperature (e.g., pool temperature).

Service Level B Transient 9 - Control Rod Misoperation

This transient includes misoperations of the control rod assemblies (CRAs), such as the drop of a single CRA, the drop of a bank of CRAs, withdrawal of a single CRA, or withdrawal of a CRA bank. The CRA adds negative reactivity to the core that quickly reduces reactor power. Such a reduction in power leads to a decrease in RCS temperature and pressure. The decreasing temperature leads to a reactivity insertion due to the negative moderator temperature coefficient. The reactor trips on low pressure or pressurizer level. Removal of decay heat is by feedwater flow or DHRS.

A withdrawal of a single control rod assembly, or the entire bank, is initiated from full or partial power (whichever is more limiting) with the control bank at its limiting insertion configuration. The rods are withdrawn at a maximum speed until fully withdrawn or until a reactor trip signal is reached.

Service Level B Transient 10 - Inadvertent Pressurizer Spray

The inadvertent pressurizer spray transient entails, either through equipment failure or operator error, actuation of continuous pressurizer spray. With the spray control valve fully open, spray flow at the maximum design flow and the minimum expected temperature is provided to the pressurizer. The pressurizer heaters energize to counteract the decrease in pressurizer pressure. A reactor trip on low pressurizer pressure occurs. The low pressurizer pressure triggers secondary system containment isolation and eventually actuates both trains of the DHRS. Removal of decay heat is by the DHRS.

Service Level B Transient 11 - Cold Overpressure Protection

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

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

Service Level B Transient 12 - Chemical and Volume Control System Malfunctions

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

3.9.1.1.3 Service Level C Conditions

Service Level C Transient 1 - Spurious Emergency Core Cooling System Valve Actuation

The ECCS consists of two RVVs and two RRVs. In an inadvertent actuation of an RRV, the inadvertent actuation block (IAB) feature provides mechanical pressure-locking to prevent opening of the valve when the RCS and CNV are at normal operating pressure. The opening of a single RVV or RRV could be caused by equipment malfunction. This event causes a decrease of RCS inventory due to the blowdown of RCS fluid to the CNV. The bounding operating condition for the opening of an RVV or RRV maximizes the mass and energy release from the valve within the range of power generation. When the ECCS valve opens, a reactor trip signal is generated on either high containment pressure or low pressurizer pressure. The high containment pressure signal causes a containment system isolation, secondary system isolation, and DHRS actuation. The open ECCS valve allows reactor coolant to blow down into the CNV. As the hot steam contacts the CNV walls, it condenses to liquid and accumulates in the bottom of the CNV. The CNV wall

is cooled by convection to the surrounding reactor pool. The ECCS actuates when the water level in the RPV drops and the low RPV level setpoint is reached. The other RRVs and RVVs open, and this configuration establishes a two-phase natural recirculation loop that provides cooling for the RCS through the RVVs and keeps the core covered by returning liquid to the RPV through the RRVs.

The second scenario considered is that both RVVs open simultaneously due to operator error or a failure of the control system. This transient develops in the same sequence as the first scenario, but the depressurization occurs sooner than the first scenario.

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

Inadvertent opening of one of the RSVs causes the RCS to quickly depressurize as primary coolant blows down to containment. The reactor trips likely due to high containment pressure or low pressurizer pressure. High containment pressure causes a containment system isolation, secondary system isolation, and DHRS actuation. Hot vapor entering containment condenses on CNV walls and fall to the bottom of containment. When the low RPV liquid level setpoint is reached, the four ECCS valves open. The open valves establish the ECCS two-phase natural recirculation loop. Decay heat is removed by the vapor moving through the RVVs to the CNV and the core is kept covered by liquid returning to the RPV through the RRVs. Removal of decay heat is expected through the containment wall and peak pressure in the CNV is kept below design pressure.

Service Level C Transient 3 - Chemical and Volume Control System Pipe Break

The CVCS pipe break is characterized by a rupture of a pipe carrying reactor coolant to or from the CVCS. The break could occur inside containment in the RCS piping or outside of containment in the CVCS piping. Breaks are excluded from occurring in the containment penetration areas. A non-isolable break inside containment maximizes the dynamic response of the RPV and reactor vessel internals and captures a pressure and thermal cycle for the CNV and components inside containment. A break outside of containment could cause greater stresses on components just outside of containment. In this transient, the RCS depressurizes through the break and the level in the pressurizer decreases. The reactor trips due either to low pressurizer pressure or to level or high containment pressure, and the DHRS actuates. The ECCS actuates on low RPV water level. Removal of decay heat is through the containment wall and peak pressure in the CNV is kept below design pressure.

Service Level C Transient 4 - Steam Generator Tube Failure

The steam generator tube failure (SGTF) transient is bounded by the double-ended failure of a SG tube. The term “failure” includes both a tube

collapsing due to higher external pressure, which is the expected failure for the design, and a tube bursting due to higher inner pressure, which is a traditional failure when the high-pressure primary fluid is located inside the tubes. Multiple simultaneous SGTFs are considered beyond-design-basis events, because the NPM steam generator tubes fail by collapse under a higher external pressure. In this transient, the RCS blows down into the SG. A reactor trip would occur quickly due to high steam pressure, low pressurizer pressure, or low pressurizer level. Both trains of the DHRS will actuate to remove the decay heat, as normal cooldown using feedwater flow is not possible with SGTF. A SGTF incapacitates one train of the DHRS, and cooldown is accomplished with the other train. Components within the RPV experiences a decrease in pressure when the SG tube fails and the RCS blows down to the SG. Once the MSIVs and FWIVs close and the DHRS actuates, the pressure decrease slows to be a function of the RCS cooldown rate. The cooldown rate is determined by the performance of the single DHRS train.

3.9.1.1.4 Service Level D Conditions

Service Level D Transient 1 - Steam Piping Failures

A main steam line break causes an increase in steam flow rate and reduces SG inventory. A break inside containment is not postulated to occur because of break exclusion designations on these lines. The RCS temperature and pressure briefly decrease due to excess heat removal by the steam line blowdown. A break causes a reactor trip on low steam pressure. Once the reactor trips, both trains of the DHRS activates. If the break compromises water inventory inside one DHRS train, the remaining train of the DHRS removes decay heat from the reactor. The RSVs do not lift and there is no ECCS actuation. Removal of decay heat is by the DHRS.

Service Level D Transient 2 - Feedwater Piping Failures

Due to the interaction of the DHRS and feedwater system, the spectrum of feedwater piping breaks includes breaks in the DHRS. A feedwater or DHRS piping break inside containment is not postulated to occur because of break exclusion designation on these lines. A break outside of containment could cause stresses on nearby components. The RCS temperature and pressure briefly decrease due to the excess cooling provided by the feedwater line blowdown. Once the quick blowdown phase is over, the transient results in heating and pressurization of the RCS. A break causes a reactor trip on low steam pressure. Once the reactor trips, both trains of the DHRS activate. If the break compromises water inventory inside one DHRS train, the remaining train of the DHRS removes decay heat from the reactor. The RSVs do not lift. Peak pressure in the CNV is kept below design pressure.

Service Level D Transient 3 - Control Rod Assembly Ejection

This transient covers a spectrum of possible control rod ejection scenarios to find the most limiting case. Scenarios are considered at different power levels,

fuel burnups, and rod configurations. The most reactive control rod for a given scenario is postulated to be ejected from the core. Removing the control rod causes a local reactivity insertion that leads to a pressure increase. Once the rod is ejected, there is a delay before the MPS trips the reactor. The trip could be caused by high reactor power or high-rate power change.

3.9.1.1.5 Test Conditions

Primary Side Hydrostatic Test

The primary side hydrostatic test consists of pressurizing the RPV to a minimum of 125 percent of design pressure. The testing complies with ASME BPVC Section III (Reference 3.9-1), Article NB-6000.

Secondary Side Hydrostatic Test

The secondary-side hydrostatic test consists of pressurizing the secondary side to a minimum of 125 percent of the SGS design pressure. The testing complies with ASME BPVC Section III (Reference 3.9-1), Article NB-6000.

Containment Hydrostatic Test

The CNV hydrostatic test consists of pressurizing the CNV to a minimum of 125 percent of design pressure. The testing complies with ASME BPVC Section III (Reference 3.9-1), Article NB-6000.

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

3.9.1.2 Computer Programs Used in Analyses

Development, procurement, testing, and maintenance of computer programs used in the dynamic and static analyses of mechanical loads, stresses, and deformations, and in the hydraulic transient load analysis of seismic Category I components and supports are completed in compliance with the Quality Assurance Program described in Chapter 17.

Delegated responsibilities may be performed under an approved supplier's or principal contractor's Quality Assurance Program, in which case the supplier would be responsible for the control of computer programs used.

NuScale uses the following computer programs.

ANSYS - The ANSYS, Inc., ANSYS software package is a pre-verified and configuration-managed finite element analysis program used in the design and analysis of safety related components. Section 3.7.2 and Section 3.8.4 discuss use of this program in structural and seismic analyses, and Section 3.12.4 discusses its use in piping stress analyses.

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

NRELAP5 - NRELAP5 is NuScale's proprietary system thermal-hydraulics code for use in safety-related design and analysis calculations, and is pre-verified and configuration-managed. The NRELAP5 system is based on RELAP5-3D, a product of Idaho National Laboratory. The code permits simulation of single-phase or two-phase systems and includes many generic component models that can be used in transient dynamic analyses. Development, use, verification, validation, and code limitations of this program are discussed in Section 15.0.2 for application to transient and accident analyses, and in the NuScale Power Module Short-Term Transient Analysis technical report TR-121517 (Reference 3.9-8) for application to short-term transient dynamic mechanical loads such as pipe breaks and valve actuations. The NRELAP5 system is also used for analysis of secondary-side instabilities in the steam generator design. The Methodology for the Determination of the Onset of Density Wave Oscillations (DWO) topical report (Reference 3.9-9) discusses development and use of NRELAP5 for identification of density wave oscillations in steam generator tubes.

Section 3.7 describes computer programs used in seismic design.

3.9.1.3 Experimental Stress Analysis

Experimental stress analysis is not used for the design.

3.9.1.4 Considerations for the Evaluation of Service Level D Condition

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

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

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

Section 3.1 discusses the design's general compliance with GDCs. Descriptions below describe compliance for dynamic testing and analysis of systems, components, and equipment. The design complies with relevant requirements of the following regulations, including the General Design Criteria (GDC) of 10 CFR 50, Appendix A:

- GDC 1 and 10 CFR 50.55a, for testing systems and components to quality standards commensurate with the importance of the safety-related functions to be performed. Section 17.5 describes the Quality Assurance Program Description.
- GDC 2 and 10 CFR 50, Appendix S, for SSC designed to withstand appropriate combinations of effects of normal and accident conditions with effects of natural phenomena without losing the ability to perform their safety functions. Pursuant to GDC 2, mechanical components are designed to withstand loads generated by natural phenomena as discussed in Section 3.1.1.
- GDC 4 as it relates to SSC being protected against dynamic effects of discharging fluids. As discussed in FSAR Section 3.6, the design protects SSC against dynamic effects, including effects from missiles, pipe whipping, and discharging fluids, which may result from equipment failures and from events and conditions outside the NPM.
- GDC 14 as it relates to SSC of the RCPB being designed to have an extremely low probability of rapidly propagating failure or of gross rupture. Section 3.9.2 addresses dynamic testing of components of the RCPB to ensure that they withstand applicable design-basis seismic and dynamic loads in combination with other environmental and natural phenomena loads without leakage, rapidly propagating failure, or gross rupture.
- GDC 15 as it relates to the RCS being designed with sufficient margin to ensure that the RCPB is not breached during normal operating conditions and AOOs. The RCPB is designed to resist seismic, loss-of-coolant accident (LOCA), and other environmental loads. Dynamic analyses are described to confirm the structural design adequacy of the RCPB. Vibration, thermal expansion, and dynamic effects testing are also described to verify the design.
- 10 CFR 50, Appendix B, as it relates to quality assurance in dynamic testing and analysis of SSC. The Quality Assurance Program Description discussed in Section 17.5 satisfies the requirements of 10 CFR 50, Appendix B to ensure that SSC are designed, procured, fabricated, inspected, erected, and tested to standards commensurate with their contribution to plant safety.

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

This section addresses pre-operational and initial startup testing performed to verify that vibrations and thermal expansion and contraction of as-built piping systems are bounded by design requirements. Piping systems in testing scope include:

- ASME BPVC, Section III (Reference 3.9-1), Class 1, 2, and 3 piping systems.
- High-energy piping systems inside Seismic Category I structures or those whose failure would reduce function of a Seismic Category I plant feature to an unacceptable level identified in Section 3.6.1.
- Seismic Category I portions of moderate-energy piping systems located outside of containment.

COL Item 3.9-2: An applicant that references the NuScale Power Plant US460 standard design will complete an assessment of piping systems inside the Reactor Building to

determine the portions of piping to be tested for vibration, thermal expansion, and dynamic effects. Piping systems within the scope of this testing include American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Class 1, 2, and 3 piping systems, other high-energy piping systems inside Seismic Category I structures or those whose failure would reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and Seismic Category I portions of moderate-energy piping systems located outside of containment. The applicant may select the portions of piping in the design for which vibration testing is performed while considering the piping system design and analysis, including the vibration screening and analysis results and scope of testing as identified by the Comprehensive Vibration Assessment Program.

The test program, described in Section 14.2, verifies that Class 1, Class 2, Class 3, and other high-energy and Seismic Category I piping systems meet functional design requirements, that piping vibrations and thermal expansions are within acceptable levels, and that these piping systems withstand dynamic effects due to operating transients.

The vibration, thermal expansion, and dynamic effect elements of this test program, summarized below, are performed during pre-operational testing and initial startup testing.

Pre-operational Testing

Preoperational tests are performed to demonstrate that piping system components meet functional design requirements, and that piping dynamic effects are acceptable. If test acceptance criteria are not met, then corrective actions (e.g., reanalyzing with as-built values) are implemented and systems are retested if necessary.

Initial Startup Testing

Initial startup testing is performed after the reactor core is loaded into an NPM. These tests establish that vibration levels, thermal expansions and contractions, and dynamic effects of transient conditions are acceptable and bounded by analyses. If test results are not bounded, evaluations use the results from testing as input and verify that the design is acceptable.

3.9.2.1.1 Piping Vibration Details

Vibration test specifications are developed in accordance with ASME OM-2017, Division 2 (OM Standards), Part 3 (Reference 3.9-3). Piping vibration testing and assessment are performed in accordance with ASME OM-2017, Division 2 (OM Standards), Part 3 (Reference 3.9-3).

Preoperational tests and initial startup tests demonstrate that piping systems withstand vibrations resulting from normal operation, including anticipated operational occurrences. If excessive vibration is observed that is outside the

bounds of the analyses, a re-analysis to determine the cause and to identify the corrective action is performed.

The Initial Test Program does not address vibrations resulting from abnormal events (e.g., accidents).

Selection of portions of piping in the design for which vibration testing is performed as described in this section may consider the piping system design and analysis, including vibration screening and analysis results and scope of testing as identified by the Comprehensive Vibration Assessment Program (CVAP) (References 3.9-5 and 3.9-7). The ASME Code Class 1, 2, and 3 piping systems that are part of the NPM are included in the scope of the CVAP. Piping systems that meet screening criteria for applicable flow-induced vibration mechanisms are evaluated in the analysis program. If analysis shows less than 100 percent safety margin, validation testing is required in accordance with Reference 3.9-7 and the requirements of Part 3 of ASME OM-2017, Division 2 (OM Standards).

3.9.2.1.1.1 Main Steam Line Branch Piping Acoustic Resonance

Evaluations are performed during the detailed design of MS lines downstream of the NPM disconnect flanges using acoustic resonance screening criteria and additional calculations as necessary (e.g., Strouhal number) to determine if there is a concern. The methodology in "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report," TR-121353 is acceptable for this purpose.

COL Item 3.9-3: An applicant that references the NuScale Power Plant US460 standard design will verify that evaluations are performed during detailed design of the main steam lines, using acoustic resonance screening criteria and additional calculations as necessary (e.g., Strouhal number) to determine if there is a concern. The methodology in "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report," TR-121353 is acceptable for this purpose. The applicant will update Section 3.9.2.1.1.1 to describe the results of this evaluation.

3.9.2.1.2 Piping Thermal Expansion Details

Thermal expansion testing verifies that the design of piping systems tested prevents constrained thermal contraction and expansion during normal operation. The tests also provide verification that component supports can accommodate expansion of piping during normal operation. Section 14.2 describes selected planned piping thermal expansion measurement tests. Test specifications for thermal expansion testing of piping systems during preoperational and start-up testing meet ASME OM Code (Reference 3.9-3) Division 3, Part 7.

3.9.2.2 Seismic Analysis and Qualification of Seismic Category I Mechanical Equipment

This section describes the seismic system analysis and qualification of Seismic Category I SSC identified in, Table 3.2-2, performed to confirm functional integrity and operability during and after a postulated seismic event. Appendix 3A addresses seismic design criteria for the NPM.

3.9.2.2.1 Seismic Qualification Testing

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

3.9.2.2.2 Seismic System Analysis Methods

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

3.9.2.2.3 Determination of Number of Earthquake Cycles

This topic is addressed in Section 3.7.3.

3.9.2.2.4 Basis for Selection of Frequencies

This topic is addressed in Section 3.7.3.

3.9.2.2.5 Three Components of Earthquake Motion

This topic is addressed in Section 3.7.2 and Section 3.12.

3.9.2.2.6 Combination of Modal Responses

This topic is addressed in Section 3.7.2, Section 3.12, and Appendix 3A.

3.9.2.2.7 Analytical Procedures for Piping

This topic is addressed in Section 3.12.

3.9.2.2.8 Multiple-Supported Equipment Components with Distinct Inputs

This topic is addressed in Section 3.7.3 and Section 3.12.

3.9.2.2.9 Use of Constant Vertical Static Factors

This topic is addressed in Section 3.7.3.

3.9.2.2.10 Torsional Effects of Eccentric Masses

This topic is addressed in Section 3.12 and Section 3.7.3.

3.9.2.2.11 Buried Seismic Category I Piping

This topic is addressed in Section 3.12.

3.9.2.2.12 Interaction of Other Piping with Seismic Category I Piping

This topic is addressed in Section 3.12.

3.9.2.2.13 Analysis Procedure for Damping

This topic is addressed in Section 3.7.3.

3.9.2.2.14 Test and Analysis Results

This topic is addressed in Section 3.9.2.2.1 and Section 3.9.2.2.2.

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

NuScale developed a CVAP Analysis Technical Report (Reference 3.9-5) and CVAP Measurement and Inspection Plan Technical Report (Reference 3.9-7) to verify structural integrity of the NPM components subject to flow-induced vibration (FIV).

Classification of particular NPMs, in accordance with RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing,” is described in Reference 3.9-5.

The NPM components are screened against FIV mechanisms and analysis is performed to determine component susceptibility. Due to the first-of-a-kind NPM design, component screening analysis errs on the side of including potentially susceptible components, even when they could be excluded based on engineering judgment or precedent. This methodology minimizes the risk of failing to analyze a significant component. Compared to the existing PWR and BWR designs, the natural circulation design of the NPM is inherently less susceptible to FIV due to the lower primary coolant velocities. Based on these two factors, FIV analysis results demonstrate that many components have very large margins of safety.

Testing is performed to quantitatively validate analytical methods and results. Together, Reference 3.9-5 and Reference 3.9-7 demonstrate that the NPM components are not expected to be subject to unacceptable FIV.

3.9.2.4 Flow-Induced Vibration Testing of Reactor Internals Before NuScale Power Module Operation

The CVAP establishes the scope of analyses, testing, and inspections required to ensure that components of the NPM are not subject to unacceptable vibratory degradation.

A vibration test program for the NPM is described in Reference 3.9-5 and Reference 3.9-7.

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

Based on acceptable completion of the CVAP analysis and Measurement and Inspection Program, in accordance with RG 1.20, for the prototype NPM, subsequent NPMs are classified as non-prototype.

COL Item 3.9-4: An applicant that references the NuScale Power Plant US460 standard design will provide applicable test procedures before the start of testing and will submit test and inspection results from the Comprehensive Vibration Assessment Program for the NuScale Power Module in accordance with Regulatory Guide 1.20.

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

Appendix 3A describes the dynamic system analysis of reactor internals under Service Level D conditions.

Appendix 3A provides details of the structural and dynamic analysis. The dynamic analysis for Level D service condition events considers safe shutdown earthquake (SSE) events, pipe rupture, and valve actuation conditions. Section 3.9.3 defines loads and loading combinations for ASME Code Class 1, 2, and 3 components, component supports, and core support structures.

The dynamic model used for the blowdown analysis includes the CNV, RPV, RVI, and control rod drive mechanisms (CRDMs). Appendix 3A has a representative diagram of the model and additional information regarding dynamic loading analysis of this model.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

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

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

This section discusses the structural integrity of pressure-retaining components, their supports, and core support structures that are designed in accordance with ASME BPVC, Section III (Reference 3.9-1), Division 1. Section 3.2.2 discusses system quality group classifications.

Section 3.1 discusses the design's general compliance with GDCs. Descriptions below describe compliance for ASME Code Class 1, 2, and 3 components, component supports, and core support structures. The design complies with relevant requirements of the following regulations including General Design Criteria of 10 CFR 50, Appendix A:

- GDC 1 and 10 CFR 50.55a, for structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety-related function to be performed.
- GDC 2 and 10 CFR 50, Appendix S, for safety-related structures and components being designed to withstand effects of earthquakes combined with effects of normal or accident conditions without loss of capability to perform their safety functions.
- GDC 4, for structures and components being designed to accommodate effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.
- GDC 14, for the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.
- GDC 15, for the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to assure that design conditions of the RCPB are not exceeded during conditions of normal operation, including AOOs.

The design is consistent with the 2017 ASME Code, Section III (Reference 3.9-1), Division 1, subject to limitations and modification in 10 CFR 50.55a(b)(1). Section 3.12 describes piping analysis criteria and methods, modeling techniques, and pipe support criteria.

3.9.3.1 Loading Combinations, System Operating Transients, and Stress Limits

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

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

Sub-subarticle NCA-3550 of Reference 3.9-1 defines the requirements for design output documents for Design Reports (Subparagraph NCA-3551.1), Load Capacity Data Sheets (Subparagraph NCA-3551.2), and Certified Design Report Summaries (Subparagraph NCA-3551.3). Nonmandatory Appendix C provides guidance on the contents and organization of Design Reports. The documentation requirements of Subsubarticle NCA-3550 will be followed for each applicable component.

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

This section describes loads considered in the design of components, component supports, and core support structures. Loads used for piping analysis are described in Section 3.12.

Pressure

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

Hydrostatic Test

A hydrostatic test pressure at a minimum of 125 percent times the design pressure is performed, as specified by ASME Code, Section III, paragraph NB-6221.

Deadweight

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

Thermal Expansion

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

Seismic

Appendix 3A describes analyses of seismic loads on ASME Class 1, Class 2, and Class 3 components and supports. The number of SSE stress cycles included in the fatigue analysis is identified in Section 3.7.3. Fatigue effects of

such cyclic events are also considered in the design of Class 1 components, piping, and supports. Reactor pool sloshing as a result of a seismic event is discussed in TR-121515.

System Operating Transients

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

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

Wind

All Class 1, Class 2, and Class 3 components and piping are located in the Reactor Building (RXB), which is designed to withstand the effects of natural phenomena; no wind or missile loading due to hurricanes or tornadoes is applicable.

Pipe Break

Loads due to high-energy pipe breaks can take the form of pipe whip, jet impingement, elevated ambient temperatures, thrust forces, dynamic pressure transients associated with blowdown of the system, and asymmetric cavity pressurization occurring simultaneously with blowdown. Pipe whip and jet impingement are mitigated using restraints, and barriers, or if unmitigated they are evaluated and included in the applicable load combinations. Section 3.6 discusses methods used to mitigate and evaluate dynamic effects of pipe whip and jet impingement. Section 3.12 addresses pressure transients in piping. Loading on components due to thrust forces, dynamic pressure transients associated with blowdown, and asymmetric cavity pressurization are discussed in Appendix 3A.

Thermal Stratification, Cycling, and Striping

Section 3.12 discusses thermal stratification, cycling, and striping (including applicable NRC Bulletins 79-13, 88-08, and 88-11).

Friction

Frictional forces induced by the pipe on the support develop when sliding occurs across the surface of a support member in unrestrained direction(s) due to thermal expansion and contraction. Since friction is due to the gradual movement of the pipe, loads from friction are calculated using only the deadweight and thermal loads normal to the applicable support member.

Environmentally Assisted Fatigue

A fatigue analysis is performed in accordance with ASME BPVC Section III (Reference 3.9-1), Subsections NB-3200, or NG-3200 considering the effects of the light-water reactor environment in accordance with RG 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors," and NUREG/CR-6909.

Section 3.12 addresses effects of the environment on fatigue for Class 1 piping and supports.

SCRAM

The SCRAM load (Load SCR in Table 3.9-2) is the mechanical load produced by the sudden shutting down of the reactor by rapid insertion of the control rods, either automatically or manually by the reactor operator. As control rods are quickly inserted, the spring in the control rod spider hub becomes compressed and transfers load through the fuel assembly into the lower core plate, and into the NPM. The SCRAM load produces a single cycle load each time the reactor trips.

Load Test

Each path in a dual-load-path lifting device for radioactive materials weighing 10,000 pounds or more is tested to a lift load equal to 150 percent times the maximum service load for a period of 10 minutes, in accordance with ANSI 14.6-1993.

Lifting, Handling, and Transportation

Lifting, handling, and transportation loads are not required to meet ASME stress limits. However, Service Level B primary limits are used as the allowable limits for lifting, handling, and transportation loads. Platform mounting assemblies are analyzed to ensure minimum safety factors of five for material ultimate strength and three for material yield strength, and are maintained for dual-load-path loading conditions considering the dynamic load factor specified.

3.9.3.1.2 Load Combinations and Stress Limits

The RPV is a Seismic Category I, ASME Section III, Class 1 component. Load combinations and stress limits for the RPV shell and head are presented in Table 3.9-3. Load combinations for RPV piping and valve nozzles are in Table 3.9-4. Load combinations for RPV bolted flange connections are in Table 3.9-5. Load combinations for Class 1 supports are in Table 3.9-6.

The CNV is a Seismic Category I component. The ASME classification of the CNV and its supports is described in Section 3.8.2. Load combinations and stress limits for the CNV and its supports are in Section 3.8.2.

The RVI are Seismic Category I components. Portions of the RVI that perform a core support function are Class CS components in accordance with ASME Section III, Subsection NG. Remaining portions of the RVI are designated as internal structures; however, they are designed using NG-3000 as a guide and constructed to ASME Subsection NG. Load combinations and stress limit are in Table 3.9-7.

The SG supports and SG tube supports are Seismic Category I components. The SG supports and SG tube supports are designated as internal structures in accordance with ASME Section III, Subsection NG. The load combinations and stress limit are consistent with those in Table 3.9-7.

Portions of the CRDM providing a RCPB function are ASME Code Class 1, Seismic Category I components. The CRDM pressure housing is a Class 1 appurtenance per ASME BPVC, Section III, NCA-1271. The load combinations and stress limits are in Table 3.9-8. The CRDM seismic supports located on the RPV and CNV head are ASME Code Class 1, Seismic Category I component supports.

The DHRS condensers are Seismic Category I components and are classified as ASME Section III, Class 2 components. The condenser supports are classified as ASME Section III, Subsection NF, Class 2 supports. Load combinations and stress limits are presented in Table 3.9-9 and Table 3.9-10.

Load combinations for the DHRS actuation valves, RSVs, ECCS valves, secondary system containment isolation valves (SSCIVs), primary system containment isolation valves (PSCIVs), and thermal relief valves are in Table 3.9-11 through Table 3.9-15.

ASME Class 1 Piping

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

ASME Class 2 and 3 Piping

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

Core Support Structures

Core support structures are designed to ASME BPVC, Section III, Subsection NG. The loading combinations and corresponding stress design criteria per ASME service level for ASME core support structures are consistent with the RVI load combinations and acceptance criteria (Table 3.9-7).

ASME Class 1, 2, and 3 Component Supports

The ASME Class 1, Class 2, and Class 3 components and piping supports are designed in accordance with ASME BPVC, Section III, Subsection NF. Supports include the CNV support skirt, CNV lugs, the top support structure, the CNV to RPV support ledges, DHRS condenser supports, and the CRDM seismic support structure and frame. Load combinations are in Table 3.9-6, Table 3.9-10, and in Section 3.8.2. The allowable stress criteria are supplemented by RG 1.124, “Service Limits and Loading Combinations for Class 1 Linear-Type Supports,” and RG 1.130, “Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports,” for Class 1 linear-type and plate-and-shell-type support structures.

The top support structure is mounted to the CNV top head; it interfaces with the RXB crane for NPM lifting, and supports piping systems attached to penetrations in the CNV top head and electrical cables and conduit for equipment in the NPM. It is a Seismic Category I component and classified as an ASME III, Subsection NF, Class 2 support. The ASME BPVC analysis of the top support structure is in accordance with NF-3350 and it is designed to withstand service loads and loading combinations in Table 3.8.2-5.

ASME Class 1, 2, and 3 Pipe Supports

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

3.9.3.2 Design and Installation of Pressure Relief Devices

ASME Class 1 Pressure Relief Valves

The RCS reactor safety valves, described in Section 5.2.2.4, are designed as ASME BPVC, Section III, Class 1 pressure-relief devices. They are part of the RCPB and are located on the RPV head. There are two RSVs and vent directly into the CNV without piping connected to their outlets. Load combinations for the RSVs are in Table 3.9-12.

The ECCS valves, described in Section 6.3.2.2, are located on the RPV and are part of the RCPB. As described in Section 5.2.2, the ECCS valves are credited for low-temperature overpressure protection. The ECCS valves are Seismic Category I components and designed as ASME BPVC, Section III, Class 1 components.

ASME Class 2 Thermal Relief Valves

The SG thermal relief valves, addressed in Section 5.4.1.2, are classified as Seismic Category I, ASME III, Class 2 relief valves per Reference 3.9-1.

The SG thermal relief valves are designed to maintain structural integrity and to function under ASME Service Level A, B, C, and D loading combinations as shown in Table 3.9-15.

Pressure Relief Device Discharge System Design and Analysis

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

3.9.3.3 Pump and Valve Operability Assurance

The design does not rely on pumps to perform safety-related functions. Section 3.9.6 lists active safety-related valves.

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

A description of the functional and operability design and qualification provisions and IST programs for safety-related valves is provided in Section 3.9.6. Environmental qualification of safety-related valves is discussed in Section 3.11. The seismic qualification of safety-related valves is performed in accordance with ASME QME-1 (Reference 3.9-4) as endorsed by RG 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” and as discussed in Section 3.10.

The stress limits are discussed in Section 3.9.3.1.

3.9.3.4 Component Supports

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

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

3.9.4 Control Rod Drive System

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

The control rods and their drive mechanisms are capable of reliably controlling reactivity, including the safety-related function of shutting down the reactor, under conditions of normal operation, including AOOs, or under postulated accident conditions. The CRDM internal moving components, consisting of the latch mechanism and control rod drive shaft are safety-related. A positive means of insertion of the control rods is maintained and, combined with the design of the CRDS, provides margin for malfunctions such as a stuck rod (Section 4.3.1.5).

The CRDM internals that ensure positive CRA insertion consist of the latch mechanism and control rod drive shaft and are classified as safety-related and non-risk-significant. Portions of the CRDS are a part of the RCPB (specifically, the pressure housings of the CRDMs) and are safety-related. The system is designed, fabricated, and tested to quality standards commensurate with the safety-related functions to be performed. The design, fabrication, and construction complies with the ASME codes in accordance with 10 CFR 50.55a (Section 3.9.4.2), providing assurance the CRDS is capable of performing its safety-related functions by withstanding the effects of AOOs, postulated accidents, and natural phenomena such as earthquakes, as discussed in GDC 1, 2, 14, 26, 27, and 29.

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

Section 3.1 discusses the design's general compliance with GDCs. Descriptions below describe compliance for the CRDS. The design complies with the relevant requirements of the following General Design Criteria of 10 CFR 50, Appendix A:

- GDC 1 (as further specified in 10 CFR 50.55a), for the CRDS being designed to quality standards commensurate with the importance of the safety functions to be

performed. CRDS, that are required to perform safety functions, performs the functions satisfactorily.

- GDC 2, for the CRDS being designed to withstand the effects of an earthquake without loss of capability to perform its safety-related functions. Section 3.2 provides seismic classification of the CRDS in accordance with RG 1.29 “Seismic Design Classification for Nuclear Power Plants.” Seismic analysis is performed for the CRDM to ensure that components can withstand the effects of natural phenomena without loss of capability to perform their safety functions. Dynamic analysis of the CRDM is performed for the SSE event to ensure that pressure integrity is maintained during and after the SSE and the capability to rapidly insert the CRA connected to the control rod drive shaft is not compromised.

  Protection against the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, and tsunamis is provided by locating the CRDS components inside the RXB.
- GDC 14, for the RCPB portion of the CRDS being designed, constructed, and tested for the extremely low probability of leakage or gross rupture. The pressure-retaining components are seismically and environmentally qualified, ensuring RCPB integrity is maintained.
- GDC 26, for the CRDS being one of the independent reactivity-control systems that is designed with appropriate margin to assure its reactivity control function under conditions of normal operation including AOOs. The CRDS facilitates reliable operator control by performing a safe shutdown (i.e., reactor trip) by gravity dropping of the CRA on a reactor trip signal or loss of power. The CRDS is designed such that core reactivity can be safely controlled and that sufficient negative reactivity exists to maintain the core subcritical under cold conditions.
- GDC 27, as it relates to reactivity control systems, including the CRDS, being designed with appropriate margin for reliably controlling reactivity under postulated accident conditions. The CRDS and the CVCS, along with the ECCS supplemental boron addition, have the combined capability to reliably control reactivity changes and maintain the core cooling capability under postulated accident conditions with appropriate margin for a stuck rod. The CRDS rapidly inserts the CRAs by gravity upon a reactor trip signal or loss of power. The safety analyses demonstrate that, with a stuck rod, the capability to cool the core is maintained.
- GDC 29, as it relates to the CRDS, in conjunction with reactor protection systems, being designed to assure an extremely high probability of accomplishing its safety-related functions in the event of AOOs. The CRDS fulfills its functions to control the reactor within fuel and plant limits during AOOs despite a single failure of the system. The CRDS accomplishes safe shutdown (i.e., reactor shutdown via gravity-dropping of the CRAs) on a reactor trip signal or loss of power. The CRDS pressure housing is an ASME Class 1 pressure boundary.

3.9.4.1 Descriptive Information of Control Rod Drive System

The CRDS is composed of a pressure-retaining housing enclosing the working mechanism, a control rod drive shaft with a coupling for attaching to the CRA hub,

external electromagnetic coils with water cooling, the power/control system, and the rod position indication system. Two support structures are provided for the CRDMs outside of the RPV: the CRDM support structure on the top of the RPV, and CRDM support frame in the top dome of the CNV head. The CRDM seismic support plates interface with adjacent CRDMs and act together with the two support structures to laterally support the CRDM pressure housings near their tops and midpoints. Figure 4.6-1 depicts the CRDM support structure and the CRDM support frame.

Internal to the upper riser, control rod drive shaft lateral supports are provided for the control rod drive shafts that extend down from the drive mechanisms to the CRAs. In addition to these dedicated control rod drive shaft supports, the pressurizer baffle plate provides a lateral support point for the shafts. The control rod drive shaft supports are depicted in Figure 3.9-1, Figure 3.9-2, and Figure 5.1-1. The CRDS provides the rod control, reactor trip, and control rod position indication necessary for operation of the NPM. The CRDS includes the CRDM, the control and indication cabinets and cables, and supporting SSC described below and in Section 4.6. Information about the CRA and its interface with the fuel system design is in Section 4.2.

The CRDS functional testing program is discussed in Section 3.9.4.4.

3.9.4.1.1 Control Rod Drive Mechanism

The CRDM assembly is an electro-mechanical device that moves the CRA in and out of the nuclear reactor core or may hold the CRA at elevations within the range of CRA travel. If electrical power is interrupted to the CRDM, the CRA (connected to the control rod drive shaft) is released and inserted by gravity into the core. Figure 4.6-1 through Figure 4.6-6 depict the CRDM assemblies mounted above the pressurizer steam space on the RPV. The structural materials of construction for the non-pressure boundary portions of the CRDM are discussed in Section 4.5.1. Materials for the pressure boundary portions of the CRDM are discussed in Section 5.2.3. The materials for the CRA are provided in Section 4.2. Additional characteristics of the CRDMs are provided in Section 4.6.

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

The CRDM is capable of a continuous full-height withdrawal and insertion and holding a position during normal operating conditions.

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

Control Rod Drive Shaft

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

The control rod drive shaft is analyzed to the guidance of ASME, Section III, Nonmandatory Appendix F for linear type supports and is evaluated to not adversely affect the integrity of the core support structures in accordance with NG-1122(c).

Martensitic stainless steel materials used in the control rod drive shafts are Cv tested in accordance with NB-2331.

Drive Coil Assembly

The drive coil assembly has four main coils: the lift coil, the upper gripper coil, the lower gripper coil, and the load transfer coil. The drive coil assembly is a part of the external assembly that slides over the pressure housing and sits on a ledge at the base of the pressure housing. The drive coil assembly is depicted by Figure 4.6-3.The direct current generated by the control cabinets is sent through a coil that generates a magnetic field; this magnetic field engages the flat-face plunger magnet of the latch arm assembly, which moves the latch arm to engage the control rod drive shaft. The rate at which the upper gripper coil, the load transfer coil, the lower gripper coil, and the lift coil are energized determines the speed of the control rod drive shaft. The motive power supply from the alternating current distribution system to the CRDM control cabinet is interrupted when the reactor trip breakers open, causing the control rods to be inserted via gravity. The CRDS safety function of rapid insertion of the control rods is accomplished when power is removed from the CRDM. Rod movement logic tracks the speed of the control rods, which utilizes direct rod position indication.

The remote disconnect mechanism coils together with the magnets and internal components of the control rod drive shaft are capable of remotely connecting and disconnecting the control rod drive shaft from the CRA, as the control rod drive shafts are not accessible during NPM disassembly.

Pressure Housing

The pressure housing includes components of the CRDM that form the pressure boundary for the reactor coolant. The pressure housing is an ASME

BPVC, Section III, Subsection NB component. The pressure housing consists of the single-piece pressure housing (bolted to the reactor vessel head), and the top plug assembly. The removable top plug assembly is threaded onto the top of the pressure housing to complete the RCPB seal.

Latch Mechanism Assembly

The basic functions of the latch mechanism assembly are to grip, release, lift, and lower the CRA. The lifting and lowering functions are referred to as "stepping," and these steps are in 0.375-inch increments. The latch mechanism assembly contains two sets of latches, the upper gripper and lower gripper latches, as shown in Figure 4.6-5. The latches grip the control rod drive shaft when the teeth of the latch arms are engaged within the grooves in the upper segment of the control rod drive shaft.

The latch assembly is secured into the bottom of the pressure housing.

Rod Position Indicator Assembly

The rod position indicator assembly contains the rod position indication coils and interfaces with the CRDM seismic support plates. The coil assembly is a part of the CRDM external assembly and slides over the pressure housing and sits on the rod disconnect mechanism coil housing. The sensor coil assembly is shown in Figure 4.6-4.

3.9.4.1.2 Operation of the Control Rod Drive Mechanisms

The CRDM mechanical and operational requirements are discussed in Section 4.6. The following describes the different modes of CRDM operation. Reactor trip, consisting of full insertion of the CRAs into the core at design conditions, is achievable during the CRDM operating modes described below.

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

Control Rod Insertion

The control rod insertion sequence begins with only the lower gripper coil energized and the lower gripper supporting the control rod drive shaft. The load transfer coil is energized, closing the load transfer gap in the latch assembly and raising the drive shaft to prepare for engagement of the upper gripper below the drive shaft tooth. The lift coil is energized and the upper gripper is raised 0.375 inches by the magnetic force acting on the armature. The upper gripper coil is energized and the upper gripper engages the drive shaft below the tooth. The load transfer and lower gripper coils are de-energized and the load of the control rod is transferred to the upper gripper by the force of gravity and the lower gripper is released. The lift coil is de-energized and the upper gripper and the drive shaft and CRA move down

0.375 inches under the force of gravity. The lower gripper coil is re-energized and the lower gripper reengages the drive shaft below the tooth. The upper gripper coil is de-energized and the load is transferred to the lower gripper by the force of gravity. The insertion sequence for one step is complete. The sequence is repeated for additional insertion steps.

Control Rod Withdrawal

The control rod withdrawal sequence begins with only the lower gripper coil energized and the lower gripper supporting the control rod drive shaft. The load transfer coil is energized, closing the load transfer gap in the latch assembly and raising the drive shaft to prepare for engagement of the upper gripper below the drive shaft tooth. The upper gripper coil is energized and the upper gripper engages the drive shaft below the tooth. The load transfer and lower gripper coils are de-energized and the load of the control rod is transferred to the upper gripper by the force of gravity and the lower gripper is released. The lift coil is energized and the drive shaft and CRA are lifted 0.375 inches by the magnetic force acting on the lift armature. The lower gripper coil is re-energized and the lower gripper reengages the drive shaft below the tooth. The upper gripper coil is de-energized and the load is transferred to the lower gripper by the force of gravity. The lift coil is de-energized, lowering the upper gripper armature by 0.375 inches in preparation for the next stepping sequence. The withdraw sequence for one step is complete. The sequence is repeated for additional withdrawal steps.

Control Rod Holding

During most of the plant operating time, the CRDMs hold the CRAs withdrawn from the core in a static position (i.e., holding position). The latches of the latch mechanism assembly grip the drive rod when the teeth of the latch arms are engaged within the grooves in the drive rod. The three latch positions are referred to as "in-contact" (engaged and loaded, holding, closed), "in-clear" (engaged and unloaded, closed), and "out" (disengaged, open).

In normal steady state operation in which stepping is not occurring and the CRA is being maintained at a particular elevation (i.e., holding position), the lower gripper latches are in the in-contact position, and the upper gripper latches are out.

Control Rod Stepping

During normal stepping operations, the interface between the latch arms and drive rod alternates among three distinct positions. The in-contact position is the position in which the drive shaft and CRA weight are supported by the latch arms. In the normal stepping sequence, the gripper latches cycle through the three positions, but the latches do not move in or out when supporting the drive rod. When changing from in-contact to out, or vice versa, the latch/control rod drive shaft interface passes through the in-clear position. This design feature minimizes wear at the latch/control rod drive shaft

interface. When the gripper latch moves into or out of the in-clear position, the weight of the drive rod is supported by the other latch.

The main control of the stepping cycle is the voltage profile that is imposed on the four drive coils (lift, upper gripper, load transfer, and lower gripper).

3.9.4.2 Applicable Control Rod Drive System Design Specifications

The design, fabrication, construction, examination, testing, inspection, and documentation of the RCPB parts of the CRDS meet requirements of ASME BPVC, 2017 Edition, Section III (Reference 3.9-1), Division I, Subsection NB. Classification of the pressure retaining portions of the CRDS is addressed in Table 3.2-2.

The design, fabrication, examination, testing, inspection and documentation for the CRDM cooling water piping and associated valves and connectors meet the requirements of ASME B31.1.

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

The CRDM, which is part of the RCPB, is designed in accordance with 10 CFR 50.55a. The pressure boundary components are designed to meet the stress limits and design and transient conditions specified in Table 3.9-8. The preservice and inservice inspection requirements of ASME Code, Section XI (Reference 3.9-2) apply to the CRDM. Pressure boundary welds are eliminated on the CRDM due to the use of a single-piece pressure housing and threaded top plug. The requirements to prevent brittle fracture in ASME BPVC, Section III, Division I, Subsection NB apply to the CRDM. The CRDM threaded connections are designed in accordance with ASME BPVC, Section III. Additional information on compliance with codes and code cases for the RCPB is provided in Section 5.2.1.

The design, fabrication, inspection, and testing of non-pressure-retaining components typically are not covered by the ASME Code, with the exception of the CRDM seismic supports that fall under ASME BPVC, Section III, Division I, Subsection NF. For materials that do not have established stress limits, the limits are based in the material specification mechanical property requirements. The latch assembly and drive shaft are safety-related Seismic Category I components inside the RCPB, and as such are designed to not adversely affect the integrity of the core support structures and to ensure they not fail in a manner that prevents CRA insertion.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

The CRDM internal design and normal operating conditions are

- design pressure (RCS) - 2200 psia
- normal operating pressure (RCS) - 2000 psia

- design temperature (RCS) - 650 degrees Fahrenheit
- normal operating temperature (RCS) - 540 degrees Fahrenheit
- pressurizer operating temperature - 636 degrees Fahrenheit

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

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

3.9.4.4 Control Rod Drive System Operability Assurance Program

The ability of the CRDM reactor coolant pressure boundary components to perform throughout the 60-year design life of the system is confirmed by the design report required by the ASME BPVC, Section III (Reference 3.9-1).

Although the NuScale CRDS is similar to the CRDS of the operating fleet of PWRs, it has unique features that include a longer control rod drive shaft (due to the presence of an integral SG and a pressurizer volume between the top of the core and the top of the RPV) and a remote-disconnect mechanism. A proof-of-concept testing program integrated a representative CRDM, control rod drive-shaft, CRA, and fuel assembly to demonstrate the acceptable mechanical functioning of a prototype CRDS. Rod drops under various conditions were tested and measured. The testing of the prototype included CRA drop time and misalignment testing and wear susceptibility assessment is described in Section 4.2.4.2.3.

The CRDS design has an Operability Assurance Program. The CRDS Operability Assurance Program testing is a series of tests designed to demonstrate the life cycle performance and endurance, including wear characteristics, of the CRDS in a representative operational environment.

The testing verifies the performance of the CRDS components under a broad range of conditions of temperature, pressure, and flow representative of design conditions. The tests also demonstrate the acceptability of the design to meet the seismic and dynamic conditions that are expected based on the seismic and dynamic analyses. The life cycle testing is conducted under simulated normal operating conditions to demonstrate acceptable mechanical operation of the CRDM and ensure that anticipated mechanical actuations (steps, reactor trips, CRA, coupling/decoupling cycles) representative of the 60-year design life of the system can be achieved without a failure that would prevent the CRDM from performing its safety-related function.

This series of tests is intended to demonstrate acceptable performance of the CRDS with respect to wear, functioning times, latching, and the ability to overcome a stuck rod, meeting system design requirements.

COL Item 3.9-5: An applicant that references the NuScale Power Plant US460 standard design will implement a control rod drive system Operability Assurance Program that meets the requirements described in Section 3.9.4.4 and provide a summary of the testing program and results.

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

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

In accordance with the technical specifications, the CRDMs are subjected periodically to partial-movement checks to demonstrate the operation of the CRDM and acceptable core power distribution. In addition, drop tests of the CRA are performed as specified in Technical Specification Surveillance Requirement 3.1.4.3 to verify the ability to meet trip time requirements.

3.9.5 Reactor Vessel Internals

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

The RVI primary functions are to:

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

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

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

- ICI and riser level sensor guide tubes
- CSA mounting brackets
- SG tube supports
- SG supports

The design and construction of the core support structures comply with ASME BPVC, Section III, Division 1, Subsection NG. For internal structures, conformance with ASME Section III, Subsection NG is not mandatory; however, internal structures are designed using NG-3000 as a guide and are constructed to not adversely affect the integrity of the core support structures, in accordance with NG-1122(c). The RVI materials including base materials and weld filler materials are discussed in Section 4.5.2 and are designed to minimize the number of welds and bolted interfaces within the high neutron flux regions.

Section 3.1 discusses the design's general compliance with GDCs. Descriptions below describe compliance for reactor vessel internals. The design complies with 10 CFR 50.55a and the relevant requirements of the following General Design Criteria of 10 CFR 50, Appendix A:

- GDC 1 and 10 CFR 50.55a, as they relate to reactor internals; the reactor internals are designed to quality standards commensurate with the importance of the safety-related functions to be performed. The RVI components are Seismic Category I and designed to meet ASME BPVC, Section III, Division 1, Subsection NG Code requirements.
- GDC 2, as it relates to reactor internals; the reactor internals are designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety-related functions for core cooling and control rod insertion. Pursuant to GDC 2, mechanical components are designed to withstand the loads generated by natural phenomena as discussed Section 3.1.1.
- GDC 4, as it relates to reactor internals; reactor internals are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operations, maintenance, testing, and postulated pipe ruptures, including LOCA. Dynamic effects associated with postulated pipe ruptures, such as guillotine breaks of primary piping that cause asymmetric loading effects, are excluded from the design basis when analyses demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.
- GDC 10, as it relates to reactor internals; reactor internals are designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.

3.9.5.1 Design Arrangements

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

Each of the RVI sub-assemblies is described in more detail below.

During all operating conditions, the reactor core is supported by the core support structures of the CSA (core barrel and lower core plate) and upper core plate. The deadweight and other mechanical loads from the fuel are transferred to the upper and lower core plates. Under seismic and other accident conditions, the CSA transfers loads to the RPV through the CSA mounting brackets at the bottom of the RPV.

The CSA includes the core barrel, upper support blocks, lower core plate, shared fuel pins and nuts, and reflector blocks (Figure 3.9-4).

The core barrel is a continuous cylinder with no welds. The upper support blocks are fastened to the core barrel. The lower core plate, which is welded to the bottom of the core barrel, supports and aligns the bottom end of the fuel assemblies. The CSA mounting brackets are attached to the RPV bottom head.

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

The fuel is surrounded by a heavy neutron reflector. The heavy reflector, termed reflector blocks, reflects neutrons back into the core to improve fuel performance. The heavy reflector provides the core envelope and directs the flow through the core. The heavy reflector blocks do not provide support to the core and is classified as an internal structure. During seismic and other accident events the heavy reflector limits lateral movement of the fuel assemblies and transfers potential loads to the core barrel assembly.

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

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

The lower riser channels the reactor coolant flow exiting the reactor core upward toward the upper riser, and separates this flow from the flow outside the lower riser that is returning from the SGs, with the exception of flow paths in the riser shell that permit a small amount of reactor coolant to bypass the top of the riser and flow into the downcomer region.

The upper core plate, which is attached to the upper support blocks, supports and aligns the top end of the fuel assemblies. The CRA guide tubes are fastened to the upper core plate, extend upward, where they are fastened to the CRA guide

tube support plate. These guide tubes house the portion of the CRAs that extend above the top of the reactor core.

The ICI guide tubes are supported at the top by the CRA guide tube support plate, and at the bottom by the upper core plate, both of which maintain alignment of the ICI guide tubes with their respective fuel assemblies.

The upper riser assembly includes the upper riser shell, a series of CRDS support plates, a transition shell, a hanger plate, control rod drive shaft sleeves and a bellows assembly. The upper riser assembly also accepts and positions the RCS injection piping. The CRDS support plates provide support for the control rod drive shafts, the ICI guide tubes, riser level sensor guide tube, and the upper riser shell. The transition shell provides the flow interface between the upper riser assembly and lower riser assembly. The ICI guide tubes extend from their respective penetrations in the RPV top head downward through the PZR space, the upper riser, and the lower riser to their respective fuel assemblies. The portion of the riser level sensor and ICI guide tubes extend from the RPV upper head penetration to the bottom of the upper riser assembly. The portion of the ICI guide tubes extending from the RPV upper head penetrations to the bottom of the upper riser assembly is depicted in Figure 3.9-2. There is a bellows assembly in the lower portion of the upper riser (Figure 3.9-2). The bellows assembly allows for differential vertical thermal expansion between the URA and RPV and additionally provides a downward compressive load on the lower riser assembly to minimize leakage flow between the assemblies.

The upper riser assembly is located immediately above the lower riser assembly and extends upward to the pressurizer (PZR) baffle plate. It channels the reactor coolant exiting the lower riser upwards and into the space above the top of the upper riser shell and below the PZR baffle plate. Reactor coolant flow then turns downward through the annular space outside of the upper riser and inside of the RPV where the SG helical tube bundles are located. Flow paths are located in the upper riser to permit a small amount of reactor coolant to bypass the top of the riser and flow into the SG tube bundle region. These flow paths ensure sufficient boron concentration remains in the reactor coolant during DHRS-driven riser uncovery conditions following non-LOCA transients, while not introducing structural integrity and fatigue concerns.

NuScale evaluated the potential for acoustic noise from vortices that may be formed at the upper riser holes, and other potential flow-induced vibration effects of flow through the holes onto the SG tubes. The riser flow holes are not expected to produce vortices due to the flow through the holes. If, however, vortices form, they do not coincide with a relevant acoustic mode of the riser. Additionally, the fluid passing through the riser holes produces minimal forces on the nearest SG tube column, and the frequency of the normal operation flow through the holes does not coincide with a predominant structural mode of the adjacent SG tubes. Therefore, the riser flow holes do not introduce structural integrity concerns due to FIV.

The RVI assembly is fastened to the pressurizer baffle plate at the hanger plate. The CRDS shaft sleeves connect the hanger plate to the rest of the upper riser assembly.

During refueling and maintenance outages the upper riser assembly stays attached to the upper section of the NPM (upper CNV, upper RPV, and SG) while providing physical access for inspection to the RPV, SG, and upper riser assembly components.

The set of upper CRDS support plates in the upper riser assembly, in conjunction with the CRA guide tube support plate, CRA guide tubes, and upper core plate in the lower riser assembly align and provide lateral support for the control rod drive shafts. These component geometries ensure adequate alignment of the CRDS with the fuel assemblies and permit full insertion of control rods under design-basis events (DBEs).

3.9.5.2 Loading Conditions

Section 3.9.1 describes acceptable analytical methods for Seismic Category I components and supports designated ASME BPVC, Section III, Division 1, Class CS. The plant and system operating transient conditions, including postulated seismic events and DBE, that provide the basis for the design of the RVI are provided in Section 3.9.3. Section 3.9.2 addresses the CVAP including the preoperational vibration test program plan for the RVI that is consistent with RG 1.20.

COL Item 3.9-6: An applicant that references the NuScale Power Plant US460 standard design will develop a Reactor Vessel Internals Reliability Program to address industry identified aging degradation mechanism issues.

COL Item 3.9-7: An applicant that references the NuScale Power Plant US460 standard design will provide a summary of reactor core support structure American Society of Mechanical Engineers (ASME) service level stresses, deformation, and cumulative usage factor values for each component and each operating condition in conformance with ASME Boiler and Pressure Vessel Code Section III Subsection NG.

3.9.5.3 Design Bases

The RVI core support structures and internal structures are designed for the service loadings and load combinations shown in Table 3.9-7. The method of combining loads for ASME Service Level A, B, C, D, and test conditions is addressed in Section 3.9.3.

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

Section 3.9.2 provides the dynamic analyses of the RVI design under steady-state and operational transient conditions, and the proposed program for FIV startup testing.

Section 3.9.3 provides information about structural integrity evaluation for the structural design adequacy and ability, with no loss of safety function, of the RVI to withstand the loads from breaches in high-energy pressure boundaries in combination with the safe shutdown earthquake.

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

This section describes the functional design and qualification provisions for preservice testing (PST) and inservice testing (IST) of valves that are designated as Class 1, 2, or 3 under Section III of the ASME BPV Code and meet the criteria of the ASME Operation and Maintenance Code (OM Code), Paragraph ISTA-1100. This section also includes valves not categorized as ASME BPV Code Class 1, 2, or 3, but that meet the criteria of OM Code Paragraph ISTA-1100. Inservice testing of valves is performed in accordance with the OM Code (Reference 3.9-3) and applicable addenda, as endorsed by 10 CFR 50.55a(f), or where relief has been granted by the NRC in accordance with 10 CFR 50.55a(f).

Testing requirements for pumps, valves, and dynamic restraints are specified in the OM Code. NuScale used the ASME OM Code, 2017 Edition to develop the IST program for the design. The IST program applies to valves in all six NPMs. Valves are grouped as required by Mandatory Appendices I, II, and IV, as specified in Note 4 of Table 3.9-17. The IST program includes augmented testing for a limited number of valves that are relied upon in some safety analyses, but do not meet the criteria of OM Code Paragraph ISTA-1100 (Table 3.9-18). The program also considers the guidance in NUREG-1482, Revision 3.

The OM Code, Subsection ISTC specifies requirements for functional testing of valves. The functional tests are required for valves that have an active inservice test function.

The IST program includes augmented testing of valves that are relied upon in safety analyses, but do not meet the criteria of OM Code, Paragraph ISTA-1100. These valves provide a beyond-design-basis event (BDBE) function, and an ASME code class / non-code class break to meet Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.” Table 3.9-18 provides more information.

The design does not have pumps or dynamic restraints that perform a function that meets the criteria of OM Code Paragraph ISTA-1100 (Reference 3.9-3).

Components subject to the preservice and the IST program are identified in Table 3.9-17.

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

The functional design and qualification of safety-related valves is performed in accordance with ASME QME-1-2017 (Reference 3.9-4). Qualification for the electrical components of valves is described in Section 3.11.

In accordance with 10 CFR 50.55a(f)(3), Class 1, 2 and 3 valves are designed and provided with access to enable the performance of IST to assess operational readiness in accordance with the OM Code and as defined in the IST program (Section 3.9.6.3.4).

Access requirements are incorporated into the engineering design and construction documents, as specified by 10 CFR 50.55a(f)(3). The quality assurance requirements for design, fabrication, construction, and testing of safety-related valves is controlled by the plant Quality Assurance program as described in Chapter 17. These requirements are in accordance with 10 CFR 50 Appendix B.

3.9.6.2 Inservice Testing of Pumps

Pumps that meet the criteria of Paragraph ISTA-1100 are subject to the IST requirements of Subsection ISTB. The design contains no pumps that meet the criteria of OM Code Paragraph ISTA-1100. Therefore, the IST program does not include pumps.

3.9.6.3 Inservice Testing of Valves

Valves that meet the criteria of OM Code Paragraph ISTA-1100 are subject to the IST requirements of Subsection ISTC. A valve has an inservice test function if it performs a specific function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident. Inservice testing of valves verifies operational readiness, including actuating, leakage, and position verification. In accordance with 10 CFR 50.55a(b)(3)(iii)(A), power-operated valves are periodically tested to verify the capability to perform their design basis safety functions. The inservice test function of pressure relief devices is to protect systems or portions of systems that perform a function in shutting down the reactor to a safe shutdown condition, in maintaining a safe shutdown condition, or in mitigating the consequences of an accident. Safe shutdown is defined for the NuScale Power Plant when the NPM is passively cooled with $k_{eff} < 1$.

The IST program is summarized in Table 3.9-17 and includes information regarding the scope of the valve preservice and IST program, valve functions, valve categories, and test frequencies. Augmented testing of valves is summarized in Table 3.9-18. The IST program adheres to the requirements of OM Code, Subsection ISTC. Lessons learned from operating experience at nuclear power plants were used to develop the IST program. The NRC Generic Letters, NUREG-1482, and industry, and utility guidelines were considered in developing the IST program and are reflected in the requirements identified in Table 3.9-17. The testing of power-operated valves uses guidance from 10 CFR 50.55a(b)(3)(iii)(A) and the Joint Owners Group on air-operated valve (AOV) testing. The lessons learned from this guidance are reflected in the IST program and valve qualification testing requirements for AOVs, hydraulic-operated valves (HOVs), and ECCs valves.

Inservice testing may incorporate the use of nonintrusive techniques to periodically assess performance and degradation of selected check valves. The OM Code, Subsection ISTC requires that check valves be exercise tested in both the open and closed direction, regardless of the inservice test function position.

Power-operated valves that have an active function require an exercise test and a performance assessment test. The IST program applies OM Code Mandatory Appendix IV to AOVs, HOVs, and ECCS valves. Performance assessment testing is discussed further in Section 3.9.6.3.2 (3).

The design does not use safety-related

- motor-operated valves.
- manual valves.
- valves actuated by an energy source capable of only one operation, such as an explosively actuated valve.

COL Item 3.9-8: An applicant that references the NuScale Power Plant US460 standard design will establish Preservice and Inservice Testing Programs. These programs are to be consistent with the requirements in the latest edition and addenda of the American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code incorporated by reference in 10 CFR 50.55a.

3.9.6.3.1 Valve Functions Tested

The IST program identifies the intended inservice test functions for valves in systems. Open (active function) and closed (active function) safety-related functions are identified in Table 3.9-17.

Inservice tests confirm the ability of the valve to perform its intended function(s). An active valve is defined as a valve that is required to change obturator position to reach its inservice test function position. Active valves are listed in Table 3.9-16.

There are no passive valves in the design that meet the criteria of Paragraph ISTA-1100. Therefore, no passive valves are included in the IST program.

The design does not rely on safety-related electrical power to position valves to their inservice test function position. Valves classified as active in Table 3.9-17 are designed to fail to their safety position. These valves have a fail-safe test specifically identified, and this function will be tested as part of the exercise test.

Valve inservice test functions and actuator characteristics are used to determine the type of IST required. These valve functions include

- active movement to meet their inservice test function(s). All active valves fail to their inservice test position.
- RCPB isolation.

- containment isolation.
- limiting seat leakage. Seat leakage is limited to a specific maximum amount when required to meet the inservice test function.
- remote position indication.

Valve inservice test functions and valve characteristics were used to determine ASME inservice testing categories. The following criteria are used in assigning the ASME OM Code categories to the valves.

Category A - valves with specific seat leakage requirements (valves for which seat leakage is limited to a specific maximum amount in the closed position to fulfill the required function)

Category B - valves requiring inservice testing, but without specific seat leakage requirements (valves for which seat leakage in the closed position is inconsequential for fulfillment of the required function)

Category C - self-actuated valves, such as check valves and pressure relief devices, and valves that are self-actuating in response to some system characteristic, such as pressure (relief valves) or flow direction (check valves) for fulfillment of the required function.

Category D - valves that are actuated by an energy source capable of only one operation, such as rupture disks or explosively actuated valves.

3.9.6.3.2 Valve Testing

Valve testing is specified in ASME OM Code, Subsection ISTC and Mandatory Appendices I, II and IV.

(1) Valve Position Verification Tests

Valves that are included in the IST program that have position indication are observed locally or by a change in system parameter (typically flow or pressure) during valve exercising to verify proper operation of the position indication. The frequency for this position indication test is once every two years, unless otherwise justified.

(2) Valve Leakage Tests

Valves with specific seat leakage limits are tested to verify their seat leakage is within limits. These valves include:

- containment isolation valves - valves that provide isolation for fluid penetrations into the containment and must meet the requirements of 10 CFR 50 Appendix J.

- decay heat removal system (DHRS) boundary - active secondary system valves that close to establish the DHRS boundary so that closed-loop, natural circulation heat removal can be established between the decay heat removal condensers and the steam generators.

Containment Isolation

Containment isolation valves (CIVs) are leak tested in accordance with 10 CFR 50, Appendix J and Paragraph ISTC-3620. These valves are tested individually as part of the Type C local leak rate testing. Containment leak rate testing is discussed in Section 6.2.6 and in Technical Report TR-123952, “NuScale Containment Leakage Integrity Assurance” (Reference 3.9-6). Containment isolation valves referenced in the IST program shall meet the corrective action requirements of the ASME OM Code, Subsection ISTC if the CIV fails to meet its leakage criteria (Table 3.9-17).

Decay Heat Removal System Boundary

The DHRS boundary is established when the active secondary system containment isolation valves (SSCIVs) close automatically to isolate steam lines and feedwater lines to create a natural circulation flow path. Backup SSCIVs provide a nonsafety-related backup to a safety-related function to isolate steam lines and feed lines and to establish the DHRS boundary. Both the SSCIVs and backup SSCIVs have specific leakage criteria to fulfill their required function as specified in Paragraph ISTA-1100. The leakage criteria are selected to maintain DHRS inventory within acceptable limits. The SSCIVs and backup SSCIVs are leak tested in accordance with Paragraph ISTC-3630 (Table 3.9-17 and Table 3.9-18).

Pressure Isolation Valves

The design does not use pressure isolation valves that provide isolation between high- and low-pressure systems. Instead, eight safety-related chemical volume and control system (CVCS) CIVs perform the following RCS isolation functions:

- isolate RCS makeup to prevent overfilling of the pressurizer during non-LOCA transients. This function is provided by closing the CVCS makeup line and spray line isolation valves.
- isolate postulated CVCS breaks outside containment, thereby maintaining RCS inventory. This function is provided by closing the CIVs on all four CVCS lines.
- protect against reverse RCS flow during low-power startup conditions. The protection function against RCS reverse flow is achieved by closing the CVCS makeup line CIVs.

This pressure isolation function is controlled by Technical Specification 3.4.6, Chemical Volume and Control System (CVCS) Isolation Valves. The design does not incorporate dedicated pressure isolation valves.

(3) Power-Operated Valve Tests

Power-Operated Valve Exercise Tests - Valves identified in Table 3.9-17 as having an active inservice test function are exercised periodically in accordance with the OM Code, Subsection ISTC. Power-operated valves (POVs) in the IST program consist of AOVs and HOVs.

The OM Code, Subsection ISTC requires quarterly exercise testing of POVs. If quarterly full-stroke exercise testing of a valve is not practicable, then full-stroke testing is performed during cold shutdowns. If cold shutdown testing is not practicable, then the full-stroke testing can be performed each refueling cycle.

The IST requirement for measuring stroke time for valves can be completed in conjunction with a valve exercise test. The exercise test identified in Table 3.9-17 includes the stroke time test.

All POVs fail to their safe position and are subject to a valve exercise test and a fail-safe test.

The valve exercise and fail-safe tests are intended to verify that the valve repositions to its safe position on loss of actuator power.

The fail-safe test is identified as a separate test in Table 3.9-17 for clarity; however, the fail-safe and exercise test can be the same inservice test.

Valves that operate during normal plant operation at a frequency that satisfies the exercising requirement do not require an additional exercise test, provided that the observations (and measurements) required of IST are made and recorded at the frequency specified by ASME OM Code, Subsection ISTC and that fail-safe requirements are met.

Power-Operated Valve Skid-Mounted Components - The HOVs are hydraulically powered from a common central hydraulic power unit. There are two hydraulic skids per NPM that provide hydraulic power to the safety-related HOVs. These HOVs include PSCIVs, MSIVs (and bypass valves), FWIVs, and DHRS actuation valves. Valve actuator subcomponents are located on the central hydraulic power units and are treated as skid-mounted components. The components that support the HOV closing function are those in the hydraulic vent path. This path includes solenoid valves, dump valves, and a hydraulic relief valve. These subcomponents meet the criteria of Paragraph ISTC-1200(b) and are tested as part of each valve exercise test. Additional testing and

maintenance on pressure relief device subcomponents is performed as part of performance assessment testing of HOVs.

Power-Operated Valve Performance Assessment Tests - NuScale used lessons learned from operating experience at nuclear power plants to develop the design. The results are a simplified design that relies on passive safety systems and fewer components than in a typical IST program. The active inservice test functions of the high safety significant valves in the IST program include containment isolation and emergency core cooling.

High safety significant POV groups include ECCS reactor recirculation valves and the RVVs, as well as certain small actuator containment isolation HOVs. Risk-significant components are identified pursuant to Section 19.1, which evaluates the NPM for full power, low power, and shutdown modes of operation for internal and external events.

Performance assessment testing to ensure that AOVs, HOVs, and ECCS valves perform their intended safety function(s) when called upon shall consider 10 CFR 50.55a(b)(3)(iii)(A) and OM Code Mandatory Appendix IV. The requirements for OM Code Mandatory Appendix IV are applied to AOVs, HOVs, and ECCS valves. Valves are grouped plant-wide (multi-module) to optimize testing, examination, and preventative maintenance activities. Lessons learned and recommendations from the AOV Joint Owners Group are considered in the development of the specific on-site performance assessment test procedures for AOVs, HOVs, and ECCS valves.

The HOV performance assessment testing contains the following attributes:

- fail safe, exercise test, and stroke time measurement
- verifying the integrity of the nitrogen cylinder by visual inspection
- recording of as-found and as-left nitrogen pressure and temperatures when performing stroke time measurements
- comparing nitrogen pressure and temperature with previous valve tests to determine cylinder leakage rate over the test period
- testing the two redundant, fail-safe hydraulic vent paths on each valve to ensure that each vent path is fully functional
  - flow device downstream of each solenoid valve verifies both safety-related solenoid valves open on valve stroke to safe position
- periodic inspection and replacement of subcomponent relief devices
  - gas bottle relief valve
  - actuator housing relief valve
  - hydraulic line relief valve

- measuring and trending obturator torque periodically to verify and monitor valve friction degradation
- leakage testing, as required (Appendix J, Type C, DHRS boundary)

The ECCS valve performance assessment testing contains the following attributes:

- fail safe, exercise test, and stroke time measurement during NPM shutdown
- testing or inspection to ensure minimum flow capacity Cv(min) is confirmed
- testing of the IAB function (RRV only)
- testing of any ECCS valve not opened during exercise testing during NPM shutdown to demonstrate that the valve will open on low RCS pressure while the trip valve remains energized (closed)
- leak testing, leakage requirement for main valve and block valve

Preservice Performance Assessment Testing and QME-1 - Power operated valves that meet the criteria of Paragraph ISTA-1100 are qualified in accordance with ASME QME-1-2017 as endorsed in RG 1.100. Each POV design is qualified to QME-1, Paragraph QV-7400. Qualification results are used to meet the requirements of Paragraph QV-7463, Demonstration of Functional Capability of Production Valve Assemblies. Physical attributes, application, and diagnostic test data from qualification test valves are used to develop performance assessment test parameters for the power-operated valves' design. Preservice performance assessment testing verifies the functional capability of the production valve to its qualified valve assembly. This testing determines that the valve is operating acceptably and baseline test data are established, meeting the requirements for QME-1 and ASME OM Code, Subsection ISTC for demonstrating the functional capability of production valve assemblies.

COL Item 3.9-9: An applicant that references the NuScale Power Plant US460 standard design will develop specific test procedures to allow detection and monitoring of power-operated valve assembly performance sufficient to satisfy periodic verification design basis capability requirements.

COL Item 3.9-10: An applicant that references the NuScale Power Plant US460 standard design will develop specific test procedures to allow detection and monitoring of emergency core cooling system valve assembly performance sufficient to satisfy periodic verification of design-basis capability requirements.

(4) Check Valve Tests

Check Valve Exercise Tests - Check valves identified with specific inservice test functions to maintain or change state to a closed position are periodically tested.

There are no check valves with an open safety function in the design.

There are four check valves per NPM in the IST program. Two feedwater check valves are normally closed nozzle check valves located in the feedwater line, and two backup feedwater check valves are located in the feedwater header in the RXB. Feedwater check valves and backup feedwater check valves close rapidly on a feedwater line breach to preserve DHRS inventory until the FWIV and the feedwater regulating valves close. There are two CVCS check valves per NPM in the Augmented Valve Testing Program, located outboard of containment isolation valves, that have an augmented quality function.

The valves are tested during cold shutdown as detailed in Table 3.9-17 and Table 3.9-18. The PZR spray check valve and CVCS injection check valve have augmented test requirements specified in Table 3.9-18. Paragraph ISTC-5222 allows an alternative to Subsection ISTC check valve testing by establishing a condition monitoring program in accordance with OM Code Mandatory Appendix II. Paragraph II-2000 specifies grouping criteria. Check valves in the IST and augmented testing programs are nozzle check valves that are 2 to 4-inch. Valves are grouped per module, but program results are tracked and compared plant-wide (multi-module) to optimize trending, condition monitoring, and preventive maintenance activities.

The ASME OM Code, Subsection ISTC requires that check valves be exercised to the open and closed positions regardless of their inservice test function position. The exercise test is intended to show that the check valve will open in response to flow and close when flow is stopped. The open exercise test is in the nonsafety-related function position. Sufficient flow shall be provided to demonstrate that the valve obturator fully opens. This test may be performed during normal operation in accordance with Paragraph ISTC-3550, Valves in Regular Use. During the closed exercise test, valve obturator position is verified by direct measurements using nonintrusive devices or by other positive means (i.e., seat leakage or other system parameters). The acceptance criteria for assessing individual valve performance are based on full open testing (achieving design minimum flowrates) and valve closure verification using backflow tests. Valves that cannot be verified using a flow test may use other means to exercise the valve to the open and closed position as described in Subsection ISTC.

The check valve test frequencies are identified in Table 3.9-17 and Table 3.9-18. Test frequencies are pursuant to Subsection ISTC.

(5) Pressure Relief Device Tests

Pressure relief devices that meet the criteria of Paragraph ISTA-1100 are required to have periodic inservice testing. The inservice tests for these valves are identified in OM Code Mandatory Appendix I.

Preservice and periodic inservice tests are specified according to pressure relief device type and ASME Code Class.

The design has two ASME Code Class 1 RSVs per NPM. The frequency for inservice testing of ASME Class 1 safety valves is every five years. Twenty percent of the valves from this valve group are tested within any 24-month interval for Class 1 safety valves.

The design has two ASME Code Class 2 SGS thermal relief valves per NPM. The replacement frequency for ASME Class 2 SGS thermal relief valves is every 10 years.

There are no Class 3 pressure relief devices.

There are no other ASME Code Class pressure relief devices in the design. The design does not contain safety-related main steam safety valves.

The RXB contains non-ASME Code Class rupture disks and non-ASME Code Class blowout panels that protect the building from overpressure during design-basis accidents and external events. These non-reclosing pressure relief devices meet the criteria of Paragraph ISTA-1100 and are included in Table 3.9-17.

3.9.6.3.3 Valve Disassembly and Inspection

There are four check valves per NPM in the IST program and two check valves that require augmented tests. These valves are normally closed nozzle check valves in the feedwater system and in the CVCS. These valves can be leak tested to satisfy the closed exercise test. Valve disassembly and inspection may be permitted if nonintrusive techniques do not prove to be reliable.

Disassembly and inspection of other types of valves is performed based on information from qualification testing, inservice testing, or other program requirements, such as:

- PRA importance measures.
- historical performance of power-operated valves (identify valve types that experience unacceptable degradation in service).
- basic design of valves including the use of components subject to aging and requiring periodic replacement.
- analysis of valve test results during valve qualification tests.
- analysis of trends of valve test parameters during valve inservice tests.
- nonintrusive techniques applied where possible. Nonintrusive techniques are preferable to disassembly and inspection if both methods sufficiently detect valve degradation.

The program for periodic check valve disassembly and inspection includes evaluation to determine which valves in the IST program in Table 3.9-17 require disassembly and inspection and the frequency of the inspection. If the test methods in Paragraph ISTC-5221(a) and ISTC-5521(b) are impractical for certain check valves, a sample disassembly examination program is used to verify valve obturator movement. The sample disassembly examination program groups check valves of similar design, application, and service condition, and requires a periodic examination of one valve from each group.

3.9.6.3.4 Valve Accessibility

The design allows for the ability to access valves for the performance of preservice and inservice testing as required by 10 CFR 50.55a and the OM Code. Neither relief from Code requirements nor application of any approved ASME Code Case is expected to be implemented as part of the inservice testing program. The valves in the inservice testing program are located in the following areas.

Inside Containment

The ECCS valves, RSVs and SGS thermal relief valves are located inside the CNV. The ECCS valves are exercise tested remotely as the NPM is being shut down for refueling (Mode 5). The ECCS performance assessment testing can be performed remotely or directly removed from containment and bench-tested. The RSVs and SG thermal relief valves are removed for bench testing or replacement at the frequency specified by OM Code Mandatory Appendix I.

Containment Vessel Head

The containment design ensures sufficient valve accessibility for testing and maintenance. This accessibility is achieved byCNV head penetration spacing and a compact CIV design. To meet the GDC 55, 56, and 57 criteria that "isolation valves outside containment shall be located as close to the containment as practical," all CIVs are located on top of the CNV head. The feedwater check valves (located in the FWIV body) are located on the CNV head. Figure 6.2-2b shows CNV head penetrations. These valves are located on top of the CNV head and underneath the top support structure. The POVs on the CNV head are of a hydraulic-operator design with nitrogen gas closure. This actuator design is more compact than that of a hydraulic operator with spring closure. Valve bodies, except the MSIVs, main steam isolation bypass valves (MSIBVs), and DHRS actuation valves are welded directly to the CNV nozzle safe-ends. The MSIVs are welded to an ASME Code Class 2, NPS 12 main steam pipe as described in Section 6.2.4. The MSIBVs are integral to the MSIV body (Figure 6.2-5a). The DHRS actuation valves are located near the MSIVs off of a branch line from the same NPS 12 main steam pipe.

Inservice testing that requires local access at the CNV head is containment leak rate testing, DHRS boundary valve leak rate testing, and feedwater check valve closed exercise testing (leak test). Except for MSIVs and MSIBVs, each

valve is designed with a containment isolation test fixture (CITF). This design allows for Appendix J Type C, Technical Specification surveillance for DHRS or a check valve exercise test to be performed locally. The test fixture is a manual device, exempt from IST, and is designed to allow Appendix J Type B local leak rate testing.

Performance assessment testing will be performed remotely by temporary or permanently installed local diagnostic sensors.

Reactor Building Valves

The CVCS demineralized water supply isolation valves and the backup SSCIVs are outside the NPM bioshield and located in the RXB. These valves are accessible for testing and maintenance.

3.9.6.4 Relief Requests and Alternative Authorizations to the Code

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

If any OM Code Cases are implemented as part of the IST program, they shall either be previously accepted by Regulatory Guide 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code,” as incorporated by reference in 10 CFR 50.55a, or submitted as a separate alternative authorization pursuant to 10 CFR 50.55a(z).

The design requires the following relief requests and alternative authorizations to the OM Code.

3.9.6.4.1 Cold Shutdown Definition Relief Request

REQUIREMENT Paragraph ISTC-3520, Exercising Requirements, refers to full-stroke exercise testing at cold shutdowns if testing during operation at power is not practicable.

ALTERNATIVE NuScale Mode 3, “safe shutdown with all reactor coolant temperatures < 200 °F,” meets the definition of “cold shutdown outage” as defined in the OM Code Paragraph ISTA-2000.

BASIS FOR RELIEF The Technical Specifications do not have a Mode defined as “cold shutdown” as utilized in the OM Code. The term “cold shutdown” is used in this subsection for clarity with OM Code requirements. NuScale Power Plant modes of operation differ from other PWR standard technical specifications. The Mode 3 “safe shutdown” reactivity

condition is k_{eff} < 0.99 and all reactor coolant temperatures < 345 degrees F. Containment and containment isolation operability is required at temperatures ≥ 200 degrees F. To meet the intent of the OM Code, the definition for "cold shutdown outage, safe shutdown with reactor coolant temperatures < 200 °F" is an equivalent condition where the NPM is stable, important safety systems are not required, and cold shutdown testing can commence per OM Code requirements.

"Refueling outage" as defined in OM Code Paragraph ISTA-2000 is Mode 5, "refueling" in the Technical Specifications. The term "refueling" is used in this section.

3.9.6.4.2 Inadvertent Actuation Block Test Frequency Alternate Authorization

REQUIREMENT The ECCS valves are required to meet Paragraph ISTC-3100, Preservice Testing, at conditions as near as practicable to those expected during subsequent inservice testing. The ECCS valves are required to meet Paragraph ISTC-3200, Inservice Testing, when the valves are required to be operable to fulfill their required function(s).

ALTERNATIVE Preservice testing for IABs shall meet the requirements of Paragraph ISTC-3100. Inservice test frequency for IABs for the initial and subsequent NPMs are established below. This test frequency provides an equivalent level of safety by performing comprehensive testing of initial IAB performance and adjusting the test frequency pursuant to OM Code Mandatory Appendix IV.

SCOPE <u>Preservice testing of IABs shall be performed as follows</u>:

Function:

- The IAB minimum analyzed closing threshold pressure shall be verified.
- The IAB opening release pressure shall be verified to be within the analyzed range.

Scope:

- All IABs shall be tested.

<u>Inservice testing of IABs shall be performed as follows</u>:

Function:

- The IAB minimum analyzed closing threshold pressure shall be verified.

- The IAB opening release pressure shall be verified to be within the analyzed range.

Scope:

- First NPM [initial NPM of the initial NuScale Power Plant], first refueling outage: All IABs tested.
- First NPM, second refueling outage: one IAB tested.
- Subsequent NPMs, first refueling (if before the second refueling of first NPM): one IAB tested.
- All NPMs, after second refueling of the first NPM: The IAB test frequency shall be established per the requirements of OM Code Mandatory Appendix IV, Paragraph IV-3410, Performance Assessment Testing.
- Performance Assessment Testing to be performed as an alternative to OM Code Mandatory Appendix IV, Paragraph IV-3420, Stroke Testing.

ALTERNATIVE AUTHORIZATION

During plant shutdown, the emergency core cooling system RVVs and RRVs are exercise tested, fail-safe tested, and position verification tested. This testing demonstrates ECCS main and trip valve inservice test functions, but does not verify RRV inadvertent actuation block threshold and release pressures. The IAB valve is treated as a skid-mounted component with respect to IST. Skid-mounted components can be excluded from IST testing requirements when their functions are demonstrated with the assembly, but in this case, the IAB function is not demonstrated with the assembly during ECCS valve exercise testing. It is not practicable to test the IAB during normal operation. The IAB is required to be bench-tested to verify threshold and release pressure to demonstrate IAB functionality. Alternative test criteria are established pursuant to 10 CFR 50.55a(z) for preservice and inservice testing of IABs.

The ECCS valves are grouped plant-wide (multi-module) to optimize testing, examination, and maintenance activities. This test frequency, combined with the population of valves in the valve group, ensures that there is sufficient data to confirm IAB performance. When six NPMs are installed and operating, three refueling outages occur annually. The frequencies established by the criteria of OM Code Mandatory Appendix IV provide reasonable assurance for the satisfactory performance of the IAB and an equivalent level of safety to Paragraph ISTC-3510.

3.9.6.4.3 Active Hydraulically Operated Valve Alternate Authorization

Comprehensive valve testing is required for active POVs to provide reasonable assurance that these components can perform their inservice functions during all design-basis conditions. ASME OM Code Mandatory Appendix IV provides a method for accomplishing comprehensive testing active HOVs; however, Subsection ISTC does not provide a means for utilizing this method. Thus, an alternate to the Code is required.

REQUIREMENT HOVs are subject to testing pursuant to the requirements of Subsection ISTC. Unlike AOVs or MOVs, no Mandatory Appendix exists to assess the operational readiness of certain active HOVs in water-cooled reactor nuclear power plants. Most HOVs in the design are utilized for high safety significant functions. A testing methodology for HOVs is required to provide reasonable assurance that active HOVs will actuate under all design-basis conditions.

ALTERNATIVE Apply Mandatory Appendix IV for active HOVs that meet the criteria of Paragraph ISTA-1100. The application of this Mandatory Appendix to HOVs provides a level of safety equivalent to Subsection ISTC and as required by 10 CFR 50.55a(b)(3)(iii)(A). This alternative facilitates the licensee to periodically verify the capability of HOVs to perform their design-basis safety functions. This is accomplished through the comprehensive testing of these components pursuant to Mandatory Appendix IV to provide reasonable assurance that they will fulfill their safety functions under all design-basis conditions.

SCOPE An alternative to Subsection ISTC shall apply to active HOVs as follows:

Paragraph	Alternative
ISTC-3100(e): Preservice Testing for AOVs	Active AOVs and HOVs shall meet the Preservice Test requirements of Mandatory Appendix IV of this Division.
ISTC-3310: Effects of Valve Repair, Replacement, or Maintenance Values	Active AOVs and HOVs shall be tested as required by the replacement, repair, and maintenance requirements of Mandatory Appendix IV of this Division.
Table ISTC-3500-1, Note (3)	Note (3) applies to Active AOVs and HOVs.
ISTC-3560: Fail Safe Testing	AOV and HOV fail-safe test frequencies shall meet the requirements of Mandatory Appendix IV of this Division.

Paragraph	Alternative
ISTC-5130: Specific Testing, Pneumatically Operated Valves	Active AOVs and HOVs shall meet the requirements of Mandatory Appendix IV of this Division.
ISTC-5140: Specific Testing, Hydraulically Operated Valves	This section does not apply to NuScale active HOVs. Testing will be pursuant to ISTC-5130 and Mandatory Appendix IV.

The pneumatic/hydraulic (HOV) actuator design (Section 6.2.4) is similar to a pneumatic valve (AOV). No alteration of Mandatory Appendix IV criteria is required to apply the requirements to NuScale HOVs. Alternative test criteria are established pursuant to 10 CFR 50.55a(z) for comprehensive testing of HOVs to meet the requirements of 10 CFR 50.55a(b)(3)(iii)(A) for POVs.

HOVs are grouped plant-wide (multi-module) to optimize testing, examination, and maintenance activities. This test frequency, combined with the large population in each of the valve groups, ensures that there is sufficient data to confirm HOV performance. When six NPMs are installed and operating, three refueling outages occur annually. The frequencies established by Mandatory Appendix IV reasonably assures the satisfactory performance of HOVs through performance testing equivalent to the standards established in Subsection ISTC and 10 CFR 50.55a(b)(3)(iii)(A).

3.9.6.4.4 Emergency Core Cooling System Valve Alternate Authorization

Comprehensive valve testing is required for active PORVs to provide reasonable assurance that these components can perform their inservice functions during all design-basis conditions. The ECCS valves have additional design features beyond typical PORVs. ASME OM Code Mandatory Appendix IV provides a method for accomplishing comprehensive testing of the ECCS RRVs and RVVs; however, Subsection ISTC does not provide a means for utilizing this method. Thus, an alternate to the Code is required.

REQUIREMENT ECCS valves are PORVs with additional features that are subject to testing pursuant to the requirements of Subsection ISTC. ECCS valves are high safety-significant components that do require comprehensive testing to provide reasonable assurance that the valves will perform under design-basis conditions. Unlike AOVs or MOVs, no Mandatory Appendix exists to assess the operational readiness of certain active PORVs in water-cooled reactor nuclear power plants. A testing methodology for ECCS valves is required to provide reasonable assurance that ECCS valves will actuate under all design-basis conditions.

ALTERNATIVE Apply Mandatory Appendix IV for ECCS RRVs and RVVs that meet the criteria of Paragraph ISTA-1100. The application of this Mandatory Appendix to ECCS RRVs and RVVs provides a level of safety equivalent to

Subsection ISTC and as required by 10 CFR 50.55a(b)(3)(iii)(A). This alternative facilitates the licensee to periodically verify the capability of ECCS valves to perform their design-basis safety functions. This is accomplished through the comprehensive testing of these components pursuant to Mandatory Appendix IV to provide reasonable assurance that they will fulfill their safety functions under all design-basis conditions.

SCOPE

An alternative to Subsection ISTC shall apply to active ECCS RRVs and RVVs as follows:

Paragraph	Alternative
ISTC-3100(e): Preservice Testing for ECCS RRVs/RVVs	ECCS RRVs and RVVs shall meet the Preservice Test requirements of Mandatory Appendix IV of this Division with the following alternative applied to Mandatory Appendix IV, Paragraph IV-3300: a Performance assessment test performance, which include (1) testing of inadvertent block valve function (Section 3.9.6.4.4), (2) C_v flow capacity verification, and (3) low RCS pressure, energized opening test b A stroke test c A fail safe test, if applicable d Leak testing: Seat tightness of the RRVs and RVVs shall be verified by leak testing in accordance with the requirements of Mandatory Appendix I of this Division. [ISTC-5112
ISTC-3310: Effects of Valve Repair, Replacement, or Maintenance Values	ECCS RRVs and RVVs shall be tested as required by the replacement, repair, and maintenance requirements of Mandatory Appendix IV of this Division, Paragraph IV-3520.
ISTC-3560: Fail Safe Testing	ECCS RRV and RVV fail-safe test frequencies shall meet the requirements of Mandatory Appendix IV of this Division, Paragraph IV-3430.
ISTC-5110: Specific Testing, Power Operated Relief Valves	ECCS RRVs and RVVs shall meet the requirements of Mandatory Appendix IV of this Division, unless modified below.
ISTC-5111: Valve Testing Requirements	Applicable to RRVs and RVVs, as directed by Mandatory Appendix IV of this Division, Paragraph IV-3400.

Paragraph	Alternative
ISTC-5112: Leak Testing	Seat tightness of the RRVs and RVVs shall be verified by leak testing in accordance with the requirements of Mandatory Appendix I of this Division.
ISTC-5113: Valve Stroke Testing	Applicable to RRVs and RVVs, as directed by Mandatory Appendix IV of this Division, Paragraph IV-3420.
ISTC-5114: Stroke Test Acceptance Criteria	Applicable to RRVs and RVVs, as directed by Mandatory Appendix IV of this Division, Paragraph IV-7100.
ISTC-5115: Corrective Action	Applicable to RRVs and RVVs, as directed by Mandatory Appendix IV of this Division, Paragraph IV-6500 or IV-7200, as appropriate.

The ECCS RRV and RVV (Section 6.3.2.2) are PORVs that do not share design similarities with pneumatically operated valves. Therefore, some understanding of these differences is required when applying Mandatory Appendix IV criteria for comprehensive testing of RRVs and RVVs. Performance assessment tests of RRVs and RVVs are described above under the alternative applied to Paragraph IV-3300 for Preservice Testing. These tests are different from what would be performed for an AOV; however, the specified tests are applicable to the ECCS design and provide reasonable assurance that RRVs and RVVs will perform their intended safety function under design basis conditions. Alternative test criteria are established pursuant to 10 CFR 50.55a(z).

ECCS valves are grouped plant-wide (multi-module) to optimize testing, examination, and maintenance activities. This test frequency, combined with the population in each of the valve groups, ensures that there is sufficient data to confirm RRV and RVV performance. When six NPMs are installed and operating, three refueling outages occur annually. The frequencies established by the criteria of OM Code Mandatory Appendix IV provide reasonable assurance for the satisfactory performance of the RRVs and RVVs and an equivalent level of performance testing as initially established in 10 CFR 50.55a(b)(3)(iii)(A).

3.9.6.5 Augmented Valve Testing Program

Valves that are relied upon in the safety analyses, but do not meet the criteria of OM Code Paragraph ISTA-1100, are included in the augmented testing program. These valves provide a BDBE function, nonsafety-related backup of a safety-related function for containment isolation, and an ASME code class /non-code class break to meet Regulatory Guide 1.26 requirements. These components will be tested to the intent of the OM Code and applicable addenda, as endorsed by 10 CFR 50.55a(f), or where relief has been granted by the NRC in accordance with 10 CFR 50.55a(f) commensurate with their augmented requirements. The valve augmented test requirements are presented in Table 3.9-18.

3.9.7 References

3.9-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," New York, NY.

3.9-2 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," New York, NY.

3.9-3 American Society of Mechanical Engineers, *Operation and Maintenance of Nuclear Power Plants*, ASME OM-2017, New York, NY.

3.9-4 American Society of Mechanical Engineers, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," ASME QME-1-2017, New York, NY.

3.9-5 NuScale Power, LLC, "Comprehensive Vibration Assessment Program (CVAP) Analysis Technical Report," TR-121353, Revision 0.

3.9-6 NuScale Power, LLC, “NuScale Containment Leakage Integrity Assurance Technical Report,” TR-123952, Revision 0.

3.9-7 NuScale Power, LLC, “Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report,” TR-121354, Revision 0.

3.9-8 NuScale Power, LLC, "NuScale Power Module Short-Term Transient Analysis," TR-121517, Revision 0.

3.9-9 NuScale Power, LLC, “Methodology for the Determination of the Onset of Density Wave Oscillations (DWO),” TR-131981, Revision 0.

Table 3.9-1: Summary of Design Transients

Event Name	ASME Service Level	Events for 60 Year Design Life
Reactor heatup to hot shutdown	Level A	200
Reactor cooldown from hot shutdown	Level A	200
Power ascent from hot shutdown	Level A	700
Power descent to hot shutdown	Level A	300
Load following	Level A	19,750
Load regulation	Level A	767,100
Steady-state fluctuations	Level A	5,000,000
Load ramp increase	Level A	2000
Load ramp decrease	Level A	2000
Step load increase	Level A	3000
Step load decrease	Level A	3000
Large step load decrease	Level A	200
Refueling	Level A	60
Reactor coolant system makeup	Level A	175,200
Steam generator inventory control from hot shutdown	Level A	600
High point degasification	Level A	440
Containment evacuation	Level A	66,000
Containment flooding and drain	Level A	400
Secondary Leakage Tests	Level A	200
Initial Test Program	Level A	20
Decrease in feedwater temperature	Level B	180
Increase in secondary flow	Level B	30
Turbine trip without bypass	Level B	90
Turbine trip with bypass	Level B	180
Loss of normal AC power	Level B	60
Inadvertent main steam isolation valve (MSIV) closure	Level B	30
Inadvertent operation of the decay heat removal system	Level B	15
Reactor trip from full power	Level B	125
Control rod misoperation	Level B	60
Inadvertent pressurizer spray	Level B	15
Cold overpressure protection	Level B	30
CVCS malfunctions	Level B	30
Spurious emergency core cooling system valve actuation	Level C	5
Inadvertent opening of a reactor safety valve	Level C	5
CVCS pipe break	Level C	5
Steam generator tube failure	Level C	5
Steam piping failures	Level D	1
Feedwater piping failures	Level D	1
Control rod assembly ejection	Level D	1
Primary hydrostatic test	Test	10
Secondary hydrostatic test	Test	10
Containment hydrostatic test	Test	10

Note: The effects of earthquakes are not considered directly in the fluid systems design transient analyses. Where applicable, seismic loadings are considered in addition to the effects of transients in the fatigue analyses (Table 3.9-3 to Table 3.9-15 for component load combinations and Table 3.12-1 to Table 3.12-2 for piping load combinations). Section 3.7.3.2 describes the number of seismic cycles used in fatigue evaluations of appropriate components.

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

Load	Description
P	Operating pressure(1)
P_{des}	Design pressure(2)
PD	Operating pressure difference(3)
PD_{des}	Design pressure difference
DW	Deadweight
B	Buoyancy(8)
TH	Transient loads(4)
R(6)	Steam generator tube failure
REA(7)	Rod ejection accident
REF or RF	Refueling loads
EXT	Mechanical loads other than piping such as RPV and CNV support reactions, RVI and CNV interface loads, and dead weight of attached or interfacing components (such as fuel assemblies).
M	Piping, mechanical, and thermal loads on nozzles
MSPB	Main steam pipe break
FWPB	Feedwater pipe break
DBPB(5)	Design basis pipe break other than FWPB and MSPB
RSV	Reactor safety valve actuation
RVV/RRV	RVV/RRV actuation loads
ECCS	Emergency core cooling system actuation
SSE	Safe shutdown earthquake
OBE	Operating basis earthquake
L	Lifting and handling
LL	Live load
H	Pressure due to hydrostatic test
P_H	Hydrostatic test pressure
SCR	Mechanical loads due to rod drop resulting from a reactor trip.
ACT	Mechanical loads due to internal action of valve operation (other than RSV and ECCS), including any thrust loads.
PRE	Bolt preload
GS	Gasket seating load
TAM	Thermal anchor motion
DFL	Dynamic fluid loads; e.g., water hammer

Notes:

1. Operating pressure, "P," is the highest pressure during an applicable transient and may be internal or external.
2. As used for ASME stress analysis: internal design pressure shall use RPV design pressure with 0 psia in the CNV; external design pressure is 50 psia with 0 psia internal pressure.
3. Operating pressure difference, "PD," is the highest pressure difference during an applicable transient and may be internal or external.
4. Transient loads include transient pressure and thermal loads, such as rapid pressure fluctuations.
5. The DBPB includes CVCS pipe break and spurious valve actuation of the RVV, RRV, and RSV. The CVCS pipe break includes breaks for RPV high point degasification, PRZ spray, RCS discharge, and RCS injection piping inside of containment.
6. R results in negligible blowdown loads. R is limited to pressure and thermal transient.
7. REA does not result in a breach of the RCPB, therefore no blowdown or impact loads exist. R is limited to pressure and thermal transient.
8. Buoyancy loads are applicable to components for conditions when they are submerged.

Table 3.9-3: Load Combinations for Reactor Pressure Vessel Shell and Head

Plant Event	Service Level	Load Combination(1)(2)	Allowable Limit
Design	Design	P + DW + B	Design
RPV Hydrostatic Test(3)	Test	P + DW + B	Test
Normal Operating Transients	A	P + DW + B + TH	A
Refueling	N/A	DW + B + RF	B
Continued Operating Transients	B	P + DW + B + TH	B
		P + DW + B + TH + ECCS	B
		P + DW + B + TH ± (OBE or SSE)(4)	B
Design Basis Pipe Break(10)	C(5)	P + DW + B+ TH ± MAX(DBPB, ECCS, RSV)	C
SG Tube Rupture(6)		P + DW + B+ TH	C
Rod Ejection Accident(6)	D	P + DW + B+ TH	C(7)
MSPB and FWPB(8)		P + DW + B+ TH	D
Pipe Breaks + SSE		P + DW + B+ TH ± SRSS[SSE,MAX(DBPB,ECCS,RSV)](9)	D

Notes:

1. Applicable loads are defined in Table 3.9-2.
2. When the method of analysis does not retain the sign of cyclic dynamic loads (e.g., reflected pressure waves or seismic), they are combined with other loads in the most conservative combination.
3. Stress analysis of the hydrostatic test loads is not required if the actual test pressure throughout the system does not exceed the required minimum test pressure by more than 6 percent, in accordance with XIII- 3600(a).
4. This load combination is evaluated for primary plus secondary stress intensity range and fatigue, except that OBE loading is only applicable to fatigue analysis and is not required for the Level B stress intensity and stress intensity range evaluations. Nonetheless, when determining the applicability of the K_e factor for the fatigue evaluation, the OBE loading is considered in the calculation of the primary plus secondary stress intensity range, S_n, and in the simplified elastic-plastic analysis, if applicable.
5. If the total number of postulated occurrences for Service Level C conditions result in more than 25 stress cycles having an alternating stress intensity (S_{alt}) greater than the S_a value at 10^6 cycles determined from the applicable fatigue design curves given in ASME BPVC, Section III, Mandatory Appendix I, those cycles in excess of 25 are included in the fatigue analysis per NB-3113(b).
6. Dynamic load due to SG tube failure or rod ejection accident is negligible. Pressure and thermal transient response applies.
7. In accordance with NUREG-0800 SRP Section 15.4.8 Acceptance Criterion 2.
8. FWPB and MSPB are breaks outside of the CNV. Dynamic load due to FWPB and MSPB is negligible.
9. High-energy line breaks, ECCS actuation, and RSV actuation are combined with SSE by square root sum of the squares (SRSS) because the events are not concurrent. At each evaluated location, it is acceptable to combine the bounding line break or valve actuation with SSE for the stress analysis, rather than performing an analysis that considers every event at every location.
10. Transient pressure (P) and temperature (T) for DBPB also consider FWPB and MSPB.

Table 3.9-4: Load Combinations for Reactor Pressure Vessel Piping and Valve Nozzles

Plant Event	Service Level	Load Combination(1)(2)	Allowable Limit
Design	Design	P + DW + B + M ± DFL(3)	Design
RPV Hydrostatic Test(4)	Test	P + DW + B + M	Test
Normal Operating Transients	A	P + DW + B + TH + M ± DFL	A
Refueling	N/A	DW + B + RF	B
Continued Operating Transients(8)	B	P + DW + B + TH + M ± DFL	B
		P + DW + B + TH + M ± DFL + ECCS	B
		P + DW + B + TH + M ± DFL(5) ± (OBE or SSE)(6)	B
Design Basis Pipe Break(13)	C(7)	P + DW + B + TH + M ± DFL(8) ± MAX(DBPB, ECCS, RSV)	C
SG Tube Rupture(9)		P + DW + B + TH + M ± DFL(8)	C
Rod Ejection Accident(9)	D	P + DW + B + TH + M ± DFL(8)	C(10)
MSPB and FWP		P + DW + B + TH + M ± DFL(8)(11)	D
Pipe Breaks + SSE		P + DW + B + TH + M ± DFL(8) ± SRSS[SSE,MAX(DBPB,ECCS,RSV)](12)	D

Notes:

1. Applicable loads are defined in Table 3.9-2.
2. When the method of analysis does not retain the sign of cyclic dynamic loads (e.g., reflected pressure waves or seismic), they are combined with other loads in the most conservative combination.
3. The DFL for Service Level A are considered for the Design Condition.
4. Stress analysis of the hydrostatic test loads is not required if the actual test pressure throughout the system does not exceed the required minimum test pressure by more than 6 percent, in accordance with XIII- 3600(a).
5. Load combinations that include OBE may exclude the DFL load because the plant will continue to operate at least until the earthquake data processing is completed and the results can be compared to shutdown limits.
6. This load combination is evaluated for primary plus secondary stress intensity range and fatigue, except that OBE loading is only applicable to fatigue analysis and is not required for the Level B stress intensity and stress intensity range evaluations. Nonetheless, when determining the applicability of the Ke factor for the fatigue evaluation, the OBE loading is considered in the calculation of the primary plus secondary stress intensity range, Sn, and in the simplified elastic-plastic analysis, if applicable.
7. If the total number of postulated occurrences for Service Level C conditions result in more than 25 stress cycles having an alternating stress intensity (S_{alt}) greater than the S_a value at 10^6 cycles determined from the applicable fatigue design curves given in ASME BPVC, Section III, Mandatory Appendix I, those cycles in excess of 25 are included in the fatigue analysis per NB-3113(b).
8. The DFL to be combined with Level C and D events is the safety system actuation in response to the event (e.g., actuation of containment isolation valves or the DHRS, as applicable). The DFL may be omitted when pipe break load data includes the effect of a valve actuation in response to the event.
9. Dynamic load due to SG tube failure or rod ejection accident is negligible. Pressure transient response applies.
10. In accordance with NUREG-0800 SRP Section 15.4.8 Acceptance Criterion 2.
11. The FWPB and MSPB are breaks outside of the CNV. Dynamic load due to FWPB and MSPB is negligible.
12. High-energy line breaks, ECCS actuation, and RSV actuation are combined with SSE by SRSS because the events are not concurrent. At each evaluated location, it is acceptable to combine the bounding line break or valve actuation with SSE for the stress analysis, rather than performing an analysis that considers every event at every location.
13. Transient pressure (P) and temperature (T) for DBPB consider FWPB and MSPB.

Table 3.9-5: Load Combinations for Reactor Pressure Vessel Bolted Flange Connections

Plant Event	Service Level	Load Combination(1)(2)	Allowable Limit
Design	Design	P + DW + B + PRE + GS	Design
RPV Hydrostatic Test(3)	Test	P + DW + B + PRE + GS	Test
Normal Operating Transients	A	P + DW + B + TH + PRE + GS	A
Refueling	N/A	DW + B + PRE + GS + RF	B
Continued Operating Transients	B	P + DW + B + TH + PRE + GS	B
		P + DW + B + TH + PRE + GS + ECCS	B
		P + DW + B + TH + PRE + GS± (OBE or SSE)(4)	B
Design Basis Pipe Break(10)	C(5)	P + DW + B + TH + PRE + GS ± MAX(DBPB, ECCS, RSV)	C
SG Tube Rupture(6)		P + DW + B + TH + PRE + GS	C
Rod Ejection Accident(6)	D	P + DW + B + TH + PRE + GS	C(7)
MSPB and FWPB		P + DW + B + TH + PRE + GS(8)	D
Pipe Breaks + SSE		P + DW + B + TH + PRE + GS(9)	D

Notes:

1. Applicable loads are defined in Table 3.9-2.
2. When the method of analysis does not retain the sign of cyclic dynamic loads (e.g., reflected pressure waves or seismic), they are combined with other loads in the most conservative combination.
3. Stress analysis of the hydrostatic test loads is not required if the actual test pressure throughout the system does not exceed the required minimum test pressure by more than 6 percent, in accordance with XIII- 3600(a).
4. This load combination is evaluated for primary plus secondary stress intensity range and fatigue, except that OBE loading is only applicable to fatigue analysis and is not required for the Level B stress intensity and stress intensity range evaluations. Nonetheless, when determining the applicability of the Ke factor for the fatigue evaluation, the OBE loading is considered in the calculation of the primary plus secondary stress intensity range, S_n, and in the simplified elastic-plastic analysis, if applicable.
5. If the total number of postulated occurrences for Service Level C conditions result in more than 25 stress cycles having an alternating stress intensity (S_{alt}) greater than the S_a value at 10^6 cycles determined from the applicable fatigue design curves given in ASME BPVC, Section III, Mandatory Appendix I, those cycles in excess of 25 are included in the fatigue analysis per NB-3113(b).
6. Dynamic load due to SG tube failure or rod ejection accident is negligible. Pressure and thermal transient response applies.
7. In accordance with NUREG-0800 SRP Section 15.4.8 Acceptance Criterion 2.
8. The FWPB and MSPB are breaks outside of the CNV. Dynamic load due to FWPB and MSPB is negligible.
9. High-energy line breaks, ECCS actuation, and RSV actuation are combined with SSE by SRSS because the events are not concurrent. At each evaluated location, it is acceptable to combine the bounding line break or valve actuation with SSE for the stress analysis, rather than performing an analysis that considers every event at every location.
10. Transient pressure (P) and temperature (T) for DBPB consider FWPB and MSPB.

Table 3.9-6: Load Combinations for Class 1 Supports

Plant Event	Service Level	Load Combination[1][2]	Allowable Limit[3]
Design	Design	DW + PRE + M ± DFL + TAM[4]	Design
Normal Operating Transients	A	DW + PRE + M ± DFL[4] + TAM	A
Refueling	N/A	DW + PRE + M+ LL +RF + TAM	B
Continued Operating Transients	B	DW + PRE + M ± DFL + TAM	B
		DW + PRE + M ± DFL + TAM + ECCS	B
		DW + PRE + M ± DFL + TAM ± (OBE or SSE)	B
Design Basis Pipe Break	C	DW + PRE + M ± DFL[5] + TAM ± MAX(DBPB,ECCS, RSV)	C
SG Tube Rupture[6]		DW + PRE + M ± DFL[5] + TAM	C
Rod Ejection Accident[6]	D	DW + PRE + M ± DFL[5] + TAM	C[7]
MSPB and FWPB		DW + PRE + M ± DFL[5] + TAM[8]	D
Pipe Breaks + SSE		DW + PRE + M ± DFL[5] +TAM ± SRSS[SSE,MAX(DBPB/ECCS/RSV)][9]	D

Notes:

1. Applicable loads are defined in Table 3.9-2.
2. When the method of analysis does not retain the sign of cyclic dynamic loads (e.g., reflected pressure waves or seismic), they are combined with other loads in the most conservative combination.
3. Support service limits meet the regulatory positions of RG 1.124 and RG 1.130, as applicable.
4. The DFL for Service Level A are considered for the Design Condition.
5. The DFL to be combined with Level C and D events is the safety system actuation in response to the event (e.g., actuation of containment isolation valves or the DHRS, as applicable). The DFL may be omitted when pipe break load data includes the effect of a valve actuation in response to the event.
6. Dynamic load due to SG tube failure or rod ejection accident is negligible. Pressure and temperature transient response applies.
7. In accordance with NUREG-0800 SRP Section 15.4.8 Acceptance Criterion 2.
8. The FWPB and MSPB are breaks outside of the CNV. Dynamic load due to FWPB and MSPB is negligible.
9. High-energy line breaks, ECCS actuation, and RSV actuation are combined with SSE by SRSS because the events are not concurrent. At each evaluated location, it is acceptable to combine the bounding line break or valve actuation with SSE for the stress analysis, rather than performing an analysis that considers every event at every location.

Table 3.9-7: Load Combinations for Reactor Vessel Internals American Society of Mechanical Engineers Stress Analysis

Plant Event	Service Level	Load Combination(1)	Allowable Limit(2)
Design(3)	Design	PD_{des} + DW + B + EXT	Design
Normal Operating Transients	A	PD + DW + B + EXT + SCR (7) + TH	A(8)
Refueling		DW + B + RF	
Continued Operating Transients	B	PD + DW + B + EXT + TH	B
		PD + DW + B + EXT + SCR + TH	
		PD + DW + B + EXT + SCR + TH + ECCS	
		PD + DW + B + EXT + SCR + TH ± (OBE or SSE)(5)	
Design Basis Pipe Break	C	PD + DW + B + EXT + SCR + TH + DBPB	C
SG Tube Failure(9)		PD + DW + B + EXT + SCR + TH	
Rod Ejection Accident(9)	D	PD + DW + B + EXT + SCR + TH	C(6)
Main Steam and Feedwater Pipe Breaks		PD + DW + B + EXT + SCR + MSPB	D
		PD + DW + B + EXT + SCR + FWPB	
Pipe Breaks + SSE		PD + DW + B + EXT + SCR ± SRSS[SSE, MAX(DBPB, MSPB, FWPB)](4)	

Notes:

1. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
2. Stress limits are as defined in the applicable subsection of ASME BPVC Section III, Subsection NG for the specified service level.
3. The load combination for the Test Condition, when the RVI is installed during RPV hydrostatic testing, does not need to be analyzed because the pressure difference is negligible; therefore, the Design Condition is bounding.
4. Dynamic loads are combined using square-root-sum-of-the-squares (SRSS) in accordance with RG 1.92 and NUREG-0484.
5. Fatigue analysis are evaluated in accordance with NG-3200. Seismic loading is only applicable to the fatigue analysis.
6. In accordance with NUREG-0800 SRP Section 15.4.8 Acceptance Criterion 2.
7. The SCRAM load is only applicable for Power Descent to Hot Shutdown transient.
8. The Refueling event is not required to be considered in the primary plus secondary stress intensity range and fatigue evaluations.
9. Dynamic load due to SG tube failure or rod ejection accident is negligible. Pressure and thermal transient response applies.

Table 3.9-8: Load Combinations for Control Rod Drive Mechanism Pressure Housing American Society of Mechanical Engineers Stress Analysis

Plant Event	Service Level	Load Combination(1)	Allowable Limit(2)
Design	Design	P_{des} + DW + EXT	Design
Hydrostatic Test (3)	Test	P_H + DW	Test
Normal Operating Transients (4)	A	P + DW + EXT + SCR + TH + REF	Level A
Continued Operating Transients	B	P + DW + EXT + SCR + TH	Level B
		P + DW + EXT + SCR + TH + ECCS	Level B
		P + DW + EXT + SCR + TH ± (OBE or SSE)(5)	Fatigue Only
Design Basis Pipe Break	C	P + DW + EXT + SCR + TH ± MAX(DBPB,ECCS)	Level C
SG Tube Failure		P + DW + EXT + SCR + R	Level C
Rod Ejection Accident	D	P + DW + EXT + SCR + REA	Level C(6)
MS and FW Pipe Breaks		P + DW + EXT + SCR ± MAX(MSPB,FWPB)	Level D
Pipe Breaks + SSE		P + DW + EXT + SCR ± SRSS(SSE + DBPB/MSPB/FWPB) (7)	Level D

Notes:
1. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
2. Stress limits are as defined in the applicable subsection of ASME BPVC Section III, Subsection NB for the specified service level.
3. Stress analysis of hydrostatic test loads is not required if the actual test pressure throughout the system does not exceed the required minimum test pressure by more than 6 percent, in accordance with Article XIII-3600(a) -Mandatory and Nonmandatory Appendices.
4. Service Level A includes refueling/servicing operations.
5. Fatigue analysis are evaluated in accordance with RG 1.207 and NUREG/CR-6909. Seismic loading is only applicable to the fatigue analysis.
6. In accordance with NUREG-0800 Section 15.4.8 Acceptance Criterion 2.
7. Dynamic loads are combined using square-root-sum-of-the-squares (SRSS) in accordance with RG 1.92 and NUREG-0484.

Table 3.9-9: Load Combinations for Decay Heat Removal System Condenser

Plant Event	Service Level	Load Combination(1)(2)(6)	Allowable Limit
Design	-	P_{des} + DW + B + EXT	Design
Hydrostatic Testing(3)	Test	P + DW + B + EXT	Test
Refueling	A	DW + B + L + REF	Level A
Normal Operating Transients(4)	A	P + DW + B + EXT + TH	Level A
Continued Operating Transients(4)	B	P + DW + B + EXT + TH	Level A
		P + DW + B + EXT + ECCS	Level B
		P + DW + B + EXT + TH ± OBE(5)	Level B
SG Tube Rupture	C	P + DW + B + EXT + R	Level B
Design Basis Pipe Break		P + DW + B + EXT ± MAX(DBPB, RSV, ECCS)	Level B
Rod Ejection Accident	D	P + DW + B + EXT + REA	Level B
Main Steam and Feedwater Pipe Breaks (MSPB/FWPB)		P + DW + B + EXT + MSPB/FWPB	Level B
SSE + DBPB/ MSPB/FWPB		P + DW + B + EXT ± SRSS(SSE, DBPB/MSPB/FWPB)	Level B

Notes:
1. Applicable loads are defined in Table 3.9-2.
2. When the method of analysis does not retain the sign of cyclic dynamic loads (e.g., response spectrum seismic analysis) they are combined with other loads in the most conservative combination.
3. Stress analysis of the hydrostatic test loads is not required if the actual test pressure throughout the system does not exceed the required minimum test pressure by more than 6 percent, in accordance with Appendix XIII-3600(a).
4. Transient refers to any plant condition that results in actuation of the DHRS.
5. The OBE loading is only applicable to the fatigue analysis, if required.
6. Where dynamics effects of the load are negligible, transient pressure and thermal effects are considered.

Table 3.9-10: Load Combinations for Decay Heat Removal System Condenser Supports

Plant Event	Service Level	Load Combination(1)(2)(3)	Allowable Limit
Design	-	DW + PRE + EXT	Design
Normal Operating Transients	A	DW + PRE + EXT	A
Refueling	A	DW + PRE + EXT + L + REF	B
Continued Operating Transients	B	DW + PRE + EXT	B
		DW + PRE + EXT + ECCS	B
Design Basis Pipe Break	C	DW + PRE + EXT ± MAX(DBPB, RSV, ECCS)	B
Main Steam and Feedwater Pipe Breaks (MSPB/FWPB)	D	DW + PRE + EXT + MSPB/FWPB	B
SSE + MSPB/FWPB	D	DW + PRE + EXT + SRSS(SSE, MSPB/FWPB)	B
SSE + DBPB/MSPB/FWPB	D	DW + PRE + EXT ± SRSS(SSE, DBPB/MSPB/FWPB)	C

Notes:

1. Applicable loads are defined in Table 3.9-2.
2. When the method of analysis does not retain the sign of cyclic dynamic loads (e.g., response spectrum seismic analysis) they are combined with other loads in the most conservative combination.
3. Buoyancy effects are conservatively neglected for condenser support loading combinations.

Table 3.9-11: Loading Combinations for Class 2 Containment Isolation Valves and Decay Heat Removal System Actuation Valves

Plant Event	Service Level	Load Combination(1)	Service Limit(3)
Normal Operating Transients	A	P + DW + TH	Level A
Continued Operating Transients	B	P + DW + TH	Level B
OBE(2)	B	P + DW + TH ± OBE	Level B
DBPB	C	P + DW + TH + ACT + DBPB	Level B
SG Tube Rupture	C	P + DW + TH + R	Level B
REA	D	P + DW + TH + REA	Level B
MSPB and FWPB	D	P + DW + TH + ACT + MSPB/FWPB	Level B
Pipe Breaks + SSE	D	P + DW ± SRSS(MSPB/FWPB/DBPB + SSE+ ACT(4))	Level B

Notes:
(1) Applicable loads are defined in Table 3.9-2.
(2) OBE acceleration factors (if applicable) are considered to be 1/3 SSE acceleration.
(3) For Level C and Level D plant events, the valves meet service level B limits.
(4) High-energy line breaks, valve actuation are combined with SSE by SRSS because the events are not concurrent.

Table 3.9-12: Load Combinations for Reactor Safety Valves

Plant Event	Service Level	Load Combination(1)	Service Limit(3)
Normal Operating Transients	A	P + DW + TH	Level A
Continued Operating Transients	B	P + DW + TH	Level B
OBE(2)	B	P + DW + TH ± OBE	Level B
DBPB	C	P + DW + TH + ACT + DBPB	Level B
SG Tube Rupture	C	P + DW + TH + R	Level B
REA	D	P + DW + TH + REA	Level B
MSPB and FWPB	D	P + DW + TH + ACT + MSPB/FWPB	Level B
Pipe Breaks + SSE	D	P + DW ± SRSS(MSPB/FWPB/DBPB + SSE + ACT(4))	Level B

Notes:
1. Applicable loads are defined in Table 3.9-2.
2. The OBE acceleration factors (if applicable) are considered to be 1/3 SSE acceleration factors.
3. For Level C and Level D plant events, the valves meet service level B limits.
4. High-energy line breaks, valve actuation are combined with SSE by SRSS because the events are not concurrent.

Table 3.9-13: Load Combinations for Emergency Core Cooling System Valves

Plant Event	Service Level	Load Combination(1)	Allowable Limit
Design	Design	P_{des} + DW + EXT	Design
Testing	Testing	H + DW + EXT	Testing
Normal Operating Transients	A	P + DW + EXT + TH	Level A
Continued Operating Transients	B	P + DW + EXT + TH + RVV/RRV	Level B
OBE(3)	B	P + DW + EXT + TH ± OBE + RVV/RRV	Level B
DBPB	C	P + DW + EXT + DBPB + RVV/RRV	Level B(5)
SG Tube Rupture	C	P + DW + EXT + R	Level B(5)
REA	D	P + DW + EXT + REA	Level B(5)
MSPB and FWPB	D	P + DW + EXT + MSPB/FWPB	Level B(5)
Pipe Breaks + SSE	D	P + DW + EXT ± SRSS(SSE + MSPB/ FWPB/DBPB)(4)	Level B(5)

Notes:

1. Applicable loads are defined in Section 3.9.3.1.1 and Table 3.9-2.
2. For the Design conditions, external mechanical loads (EXT), are based on bounding EXT loads for Service Level A.
3. The OBE is considered in the fatigue analyses.
4. Dynamic loads are combined by square root of the sum of the squares (SRSS) in accordance with RG 1.92 and NUREG-0484. Alternatively, if time history responses are used, straight addition of the event is allowed where more than one load is acting concurrently.
5. For plant events that result in ECCS actuation, the ECCS valves meet service level B limits because the ECCS is required or may be used in these events as accident mitigation. The RVV/RRV actuation loads are considered in the fatigue evaluation.

Table 3.9-14: Required American Society of Mechanical Engineers Code Load Combinations for Primary System Containment Isolation Valves

Plant Event	Service Level	Load Combination(1)	Service Limit(3)
Normal Operating Transients	A	P + DW + TH	Level A
Continued Operating Transients	B	P + DW + TH	Level B
OBE(2)	B	P + DW + TH ± OBE	Level B
DBPB	C	P + DW + TH + ACT + DBPB	Level B
SG Tube Rupture	C	P + DW + TH + R	Level B
REA	D	P + DW + TH + REA	Level B
MSPB and FWPB	D	P + DW + TH + ACT + MSPB/FWPB	Level B
Pipe Breaks + SSE	D	P + DW ± SRSS(MSPB/FWPB/DBPB + SSE + ACT(4))	Level B

Notes:
1. Applicable loads are defined in Table 3.9-2.
2. The OBE acceleration factors (if applicable) are considered to be 1/3 SSE acceleration factors.
3. For Level C and Level D plant events, the valves meet service level B limits.
4. High-energy line breaks, valve actuation are combined with SSE by SRSS because the events are not concurrent.

Table 3.9-15: Load Combinations for Thermal Relief Valves

Plant Event	Service Level	Load Combination(1)	Service Limit(3)
Normal Operating Transients	A	P + DW + TH	Level A
Continued Operating Transients	B	P + DW + TH	Level B
OBE(2)	B	P + DW + TH ± OBE	Level B
DBPB	C	P + DW + TH + ACT + DBPB	Level B
SG Tube Rupture	C	P + DW + TH + R	Level B
REA	D	P + DW + TH + REA	Level B
MSPB and FWPB	D	P + DW + TH + ACT + MSPB/FWPB	Level B
Pipe Breaks + SSE	D	P + DW ± SRSS(MSPB/FWPB/DBPB + SSE + ACT(4))	Level B

Notes:
1. Applicable loads are defined in Table 3.9-2.
2. The OBE acceleration factors (if applicable) are considered to be 1/3 SSE acceleration factors.
3. For Level C and Level D plant events, the valves meet service level B limits.
4. High-energy line breaks, valve actuation are combined with SSE by SRSS because the events are not concurrent.

Table 3.9-16: Active Valve List

Valve No.	Description	ASME Class	Function[1]
ASME Class 1, 2, and 3			
Chemical and Volume Control System			
CVC-AOV-0336	CVCS Discharge Isolation Valve	3	5
CVC-SV-0404	RPV High Point Degasification Isolation Valve	3	5
CVC-AOV-0089	Demineralized Water Supply to CVC Makeup Upstream Isolation Valve	3	3
CVC-AOV-0090	Demineralized Water Supply to CVC Makeup Downstream Isolation Valve	3	3
Containment System			
CVC-CKV-0323	Pressurizer Spray Check Valve	3	6
CVC-CKV-0329	CVCS Injection Check Valve	3	6
CVC-HOV-0324	Pressurizer Spray Outboard Containment Isolation Valve	1	1, 2, 3, 4
CVC-HOV-0325	Pressurizer Spray Inboard Containment Isolation Valve	1	1, 2, 3, 4
CVC-HOV-0330	CVCS Injection Outboard Containment Isolation Valve	1	1, 2, 3, 4
CVC-HOV-0331	CVCS Injection Inboard Containment Isolation Valve	1	1, 2, 3, 4
CVC-HOV-0334	CVCS Discharge Inboard Containment Isolation Valve	1	1, 2, 3, 4
CVC-HOV-0335	CVCS Discharge Outboard Containment Isolation Valve	1	1, 2, 3, 4
CVC-HOV-0401	RPV High Point Degasification Inboard Containment Isolation Valve	1	1, 2, 3, 4
CVC-HOV-0402	RPV High Point Degasification Outboard Containment Isolation Valve	1	1, 2, 3, 4
CE-HOV-0001	Containment Evacuation Inboard Containment Isolation Valve	2	2, 3, 4
CE-HOV-0002	Containment Evacuation Outboard Containment Isolation Valve	2	2, 3, 4
CFD-HOV-0021	Containment Flooding & Drain Outboard Containment Isolation Valve	2	2, 3, 4
CFD-HOV-0022	Containment Flooding & Drain Inboard Containment Isolation Valve	2	2, 3, 4
RCCW-HOV-0184	Reactor Component Cooling Water Inlet Outboard Containment Isolation Valve	2	2, 3, 4
RCCW-HOV-0185	Reactor Component Cooling Water Inlet Inboard Containment Isolation Valve	2	2, 3, 4
RCCW-HOV-0190	Reactor Component Cooling Water Outlet Inboard Containment Isolation Valve	2	2, 3, 4
RCCW-HOV-0191	Reactor Component Cooling Water Outlet Outboard Containment Isolation Valve	2	2, 3, 4
FW-HOV-0137	Feedwater Isolation Valve	2	2, 3, 4
FW-HOV-0237	Feedwater Isolation Valve	2	2, 3, 4
FW-CKV-0136	Feedwater Isolation Check Valve	2	3
FW-CKV-0236	Feedwater Isolation Check Valve	2	3
MS-HOV-0101	Main Steam Isolation Valve	2	2, 3, 4
MS-HOV-0201	Main Steam Isolation Valve	2	2, 3, 4
MS-HOV-0103	Main Steam Isolation Bypass Valve	2	2, 3, 4
MS-HOV-0203	Main Steam Isolation Bypass Valve	2	2, 3, 4
Decay Heat Removal System			
DHR-HOV-0111	Decay Heat Removal System Actuation Valve	2	3, 4
DHR-HOV-0121	Decay Heat Removal System Actuation Valve	2	3, 4
DHR-HOV-0211	Decay Heat Removal System Actuation Valve	2	3, 4
DHR-HOV-0221	Decay Heat Removal System Actuation Valve	2	3, 4

Table 3.9-16: Active Valve List (Continued)

Valve No.	Description	ASME Class	Function[1]
Emergency Core Cooling System[2]			
ECC-POV-0001A	Reactor Vent Valve A	1	1, 3, 4
ECC-POV-0001B	Reactor Vent Valve B	1	1, 3, 4
ECC-POV-0002A	Reactor Recirculation Valve A	1	1, 3, 4
ECC-POV-0002B	Reactor Recirculation Valve B	1	1, 3, 4
Safety and Relief Valves			
RCS-PSV-0003A	Reactor Safety Valve A	1	1, 3
RCS-PSV-0003B	Reactor Safety Valve B	1	1, 3
SGS-RV-0102	Steam Generator System Thermal Relief Valve	2	2
SGS-RV-0202	Steam Generator System Thermal Relief Valve	2	2
Non-Code Class Valves			
Condensate and Feedwater System			
FW-AOV-0134	Feedwater Regulating Valve	NC	7
FW-AOV-0234	Feedwater Regulating Valve	NC	7
FW-CKV-0135	Backup Feedwater Check Valve	NC	7
FW-CKV-0235	Backup Feedwater Check Valve	NC	7
Main Steam System			
MS-AOV-0102	Backup Main Steam Isolation Valve	NC	7
MS-AOV-0202	Backup Main Steam Isolation Valve	NC	7
MS-AOV-0104	Backup Main Steam Isolation Bypass Valve	NC	7
MS-AOV-0204	Backup Main Steam Isolation Bypass Valve	NC	7

1 - Function 1 - Reactor coolant pressure boundary
2 - Containment isolation
3 - Accident mitigation
4 - Safe shutdown
5 - Nonsafety-related, but provide an augmented quality function (NRC Quality Group C/D boundary)
6 - Nonsafety-related, but provide an augmented quality function (NRC Quality Group C/D boundary, backup containment isolation)
7 - Nonsafety backup to a safety-related function (Section 15.0.0.6.6)

2 - Trip and reset valves are included with each RVV and RRV.

Table 3.9-17: Valve Inservice Test Requirements per ASME OM Code

Valve No.	Description	Valve / Actuator[1]	Function Position	Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
Chemical and Volume Control System								
CVC-AOV-0089	Demineralized Water Supply to CVCS Makeup Upstream Isolation Valve	BALL Remote AO	Closed	Active Boron Dilution Prevention	NC Category B	Position Verification Test/2 Years Exercise Full Stroke/Quarterly Failsafe Test/Quarterly Performance Assessment Test	POV 1	5, 16
CVC-AOV-0090	Demineralized Water Supply to CVCS Makeup Downstream Isolation Valve	BALL Remote AO	Closed	Active Boron Dilution Prevention	NC Category B	Position Verification Test/2 Years Exercise Full Stroke/Quarterly Failsafe Test/Quarterly Performance Assessment Test	POV 1	5, 16
Condensate and Feedwater System								
FW-AOV-0134	Feedwater Regulating Valve A	FCV Remote AO	Closed	Active Backup Feedwater Isolation Backup Containment Isolation Backup DHRS Boundary	NC Category A	Position Verification Test Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 5	15, 17
FW-AOV-0234	Feedwater Regulating Valve B	FCV Remote AO	Closed	Active Backup Feedwater Isolation Backup Containment Isolation Backup DHRS Boundary	NC Category A	Position Verification Test Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 5	15, 17
FW-CKV-0135	Backup Feedwater Check Valve A	Nozzle Check	Closed	Active Decay Heat Removal Boundary	NC Category C	Check Exercise/ Cold Shutdown	CKV 1	18
FW-CKV-0235	Backup Feedwater Check Valve B	Nozzle Check	Closed	Active Decay Heat Removal Boundary	NC Category C	Check Exercise/ Cold Shutdown	CKV 1	18
Containment System								
CVC-HOV-0324	Pressurizer Spray Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 16

Table 3.9-17: Valve Inservice Test Requirements per ASME OM Code (Continued)

Valve No.	Description	Valve / Actuator[1]	Function Position	Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
CVC-HOV-0325	Pressurizer Spray Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 16
CVC-HOV-0330	Chemical and Volume Control System Injection Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 16
CVC-HOV-0331	Chemical and Volume Control System Injection Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 16
CVC-HOV-0334	Chemical and Volume Control System Discharge Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 16
CVC-HOV-0335	Chemical and Volume Control System Discharge Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 16
CVC-HOV-0401	RPV High Point Degasification Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 16
CVC-HOV-0402	RPV High Point Degasification Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 16

Table 3.9-17: Valve Inservice Test Requirements per ASME OM Code (Continued)

Valve No.	Description	Valve / Actuator[1]	Function Position	Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
CE-HOV-0001	Containment Evacuation System Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	POV 3	6, 16
CE-HOV-0002	Containment Evacuation System Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	POV 3	6, 16
CFD-HOV-0021	Containment Flooding & Drain System Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 16
CFD-HOV-0022	Containment Flooding & Drain System Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 16
RCCW-HOV-0184	Reactor Component Cooling Water System Inlet Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 7, 16
RCCW-HOV-0185	Reactor Component Cooling Water System Inlet Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 7, 16

Table 3.9-17: Valve Inservice Test Requirements per ASME OM Code (Continued)

Valve No.	Description	Valve / Actuator[1]	Function Position	Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
RCCW-HOV-0190	Reactor Component Cooling Water System Outlet Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 7, 16
RCCW-HOV-0191	Reactor Component Cooling Water System Outlet Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	POV 2	6, 7, 16
FW-HOV-0137	Feedwater Isolation Valve A	BALL Remote HO	Closed	Active Feedwater Isolation Containment Isolation Decay Heat Removal Boundary	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 3	8, 15, 16
FW-HOV-0237	Feedwater Isolation Valve B	BALL Remote HO	Closed	Active Feedwater Isolation Containment Isolation Decay Heat Removal Boundary	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 3	8, 15, 16
FW-CKV-0136	Feedwater Check Valve A	NOZZLE CHECK	Closed	Active Feedwater Isolation Decay Heat Removal Boundary	Class 2 Category C	Check Exercise/ Refueling	CKV 1	9
FW-CKV-0236	Feedwater Check Valve B	NOZZLE CHECK	Closed	Active Feedwater Isolation Decay Heat Removal Boundary	Class 2 Category C	Check Exercise/ Refueling	CKV 1	9

Table 3.9-17: Valve Inservice Test Requirements per ASME OM Code (Continued)

Valve No.	Description	Valve / Actuator[1]	Function Position	Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
MS-HOV-0101	Main Steam Isolation Valve A	BALL Remote HO	Closed	Active Steam Line Isolation Containment Isolation Decay Heat Removal Boundary	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 4	10, 15, 16
MS-HOV-0201	Main Steam Isolation Valve B	BALL Remote HO	Closed	Active Steam Line Isolation Containment Isolation Decay Heat Removal System Boundary	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 4	10, 15, 16
MS-HOV-0103	Main Steam Isolation Bypass Valve A	BALL Remote HO	Closed	Active Steam Line Isolation Containment Isolation Decay Heat Removal Boundary	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 2	10, 15, 16
MS-HOV-0203	Main Steam Isolation Bypass Valve B	BALL Remote HO	Closed	Active Steam Line Isolation Containment Isolation Decay Heat Removal Boundary	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 2	10, 15, 16
Decay Heat Removal System								
DHR-HOV-0111	Decay Heat Removal System Actuation Valve	BALL Remote HO	Open	Active Decay Heat Removal	Class 2 Category B	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Performance Assessment Test	POV 3	11, 16
DHR-HOV-0121	Decay Heat Removal System Actuation Valve	BALL Remote HO	Open	Active Decay Heat Removal	Class 2 Category B	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Performance Assessment Test	POV 3	11, 16

Table 3.9-17: Valve Inservice Test Requirements per ASME OM Code (Continued)

Valve No.	Description	Valve / Actuator[1]	Function Position	Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
DHR-HOV-0211	Decay Heat Removal System Actuation Valve	BALL Remote HO	Open	Active Decay Heat Removal	Class 2 Category B	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Performance Assessment Test	POV 3	11, 16
DHR-HOV-0221	Decay Heat Removal System Actuation Valve	BALL Remote HO	Open	Active Decay Heat Removal	Class 2 Category B	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Performance Assessment Test	POV 3	11, 16
Emergency Core Cooling System								
ECC-POV-0001A	Reactor Vent Valve A	GLOBE Remote HO	Open/ Closed	Active Core Cooling Recirculation Path Reactor Coolant Pressure Boundary LTOP	Class 1 Category B/C	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Performance Assessment Test	POV 6	12, 16
ECC-POV-0001B	Reactor Vent Valve B	GLOBE Remote HO	Open/ Closed	Active Core Cooling Recirculation Path Reactor Coolant Pressure Boundary LTOP	Class 1 Category B/C	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Performance Assessment Test	POV 6	12, 16
ECC-POV-0002A	Reactor Recirculation Valve A	GLOBE Remote HO	Open/ Closed	Active Core Cooling Recirculation Path Reactor Coolant Pressure Boundary	Class 1 Category B/C	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Performance Assessment Test	POV 6	12, 16
ECC-POV-0002B	Reactor Recirculation Valve B	GLOBE Remote HO	Open/ Closed	Active Core Cooling Recirculation Path Reactor Coolant Pressure Boundary	Class 1 Category B/C	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Performance Assessment Test	POV 6	12, 16

Table 3.9-17: Valve Inservice Test Requirements per ASME OM Code (Continued)

Valve No.	Description	Valve / Actuator[1]	Function Position	Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
Main Steam System								
MS-AOV-0102	Backup Main Steam Isolation Valve	GATE Remote AO	Closed	Active Steam Line Isolation Containment Isolation Decay Heat Removal Boundary	NC Category A	Position Verification Test Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 7	15, 19
MS-AOV-0202	Backup Main Steam Isolation Valve	GATE Remote AO	Closed	Active Steam Line Isolation Containment Isolation Decay Heat Removal Boundary	NC Category A	Position Verification Test Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 7	15, 19
MS-AOV-0104	Backup Main Steam Isolation Bypass Valve	GATE Remote AO	Closed	Active Steam Line Isolation Containment Isolation Decay Heat Removal Boundary	NC Category A	Position Verification Test Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 8	15, 19
MS-AOV-0204	Backup Main Steam Isolation Bypass Valve	GATE Remote AO	Closed	Active Steam Line Isolation Containment Isolation Decay Heat Removal Boundary	NC Category A	Position Verification Test Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	POV 8	15, 19
Pressure Relief Devices								
RCS-PSV-0003A	Reactor Safety Valve A	SAFETY Self Actuating	Open/ Closed	Overpressure Protection Reactor Coolant Pressure Boundary	Class 1 Category B/C	Position Verification Test, 2 Years (alternated) Class 1 Safety Valve Test/ 5 Years and 20% in 2 Years	PRD 1	13
RCS-PSV-0003B	Reactor Safety Valve B	SAFETY Self Actuating	Open/ Closed	Overpressure Protection Reactor Coolant Pressure Boundary	Class 1 Category B/C	Position Verification Test, 2 Years (alternated) Class 1 Safety Valve Test/ 5 Years and 20% in 2 Years	PRD 1	13

Table 3.9-17: Valve Inservice Test Requirements per ASME OM Code (Continued)

Valve No.	Description	Valve / Actuator[1]	Function Position	Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
SGS-RV-0102	Steam Generator System Thermal Relief Valve A	THERMAL RELIEF Self Actuating	Open/ Closed	Thermal Overpressure Protection Steam Generator System Pressure Boundary	Class 2 Category A/C	10 Years	PRD 2	14
SGS-RV-0202	Steam Generator System Thermal Relief Valve B	THERMAL RELIEF Self Actuating	Open/ Closed	Thermal Overpressure Protection Steam Generator System Pressure Boundary	Class 2 Category A/C	10 Years	PRD 2	14
00-RBCM-RPD-0001A	Pool Area Rupture Disk A	RUPTURE DISK	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 3	21
00-RBCM-RPD-0001B	Pool Area Rupture Disk B	RUPTURE DISK	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 3	21
RBCM-RPD-0002A	Steam Gallery Blowout Panel A	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 4	21
RBCM-RPD-0002B	Steam Gallery Blowout Panel B	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 4	21
RBCM-RPD-0002C	Steam Gallery Blowout Panel C	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 4	21
RBCM-RPD-0002D	Steam Gallery Blowout Panel D	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 4	21
RBCM-RPD-0002E	Steam Gallery Blowout Panel E	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 4	21
RBCM-RPD-0002F	Steam Gallery Blowout Panel F	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 4	21

Table 3.9-17: Valve Inservice Test Requirements per ASME OM Code (Continued)

Valve No.	Description	Valve / Actuator[1]	Function Position	Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
01-RBCM-RPD-0003	CVCS Pipe Chase Blowout Panel 01	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 5	21
02-RBCM-RPD-0003	CVCS Pipe Chase Blowout Panel 02	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 5	21
03-RBCM-RPD-0003	CVCS Pipe Chase Blowout Panel 03	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 5	21
04-RBCM-RPD-0003	CVCS Pipe Chase Blowout Panel 04	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 5	21
05-RBCM-RPD-0003	CVCS Pipe Chase Blowout Panel 05	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 5	21
06-RBCM-RPD-0003	CVCS Pipe Chase Blowout Panel 06	RUPTURE PANEL	Open	RXB Overpressure Protection	NC Category D	Nonreclosing Pressure Relief Device Inspect and replace 5 years	PRD 5	21

Notes:

1. AO air operated; CITF Containment Isolation Test Fixture; CKV check valves; CIV containment isolation valve; CNV containment vessel; CVCS chemical and volume control system; DHR decay heat removal; DWS demineralized water system; ECCS emergency core cooling system; FCV flow control valve; FWIV feedwater isolation valve; FWRV feedwater regulating valve; HO hydraulic operated; LTOP low temperature overpressure protection; MCR main control room; MPS module protection system; MSIV main steam isolation valve; MSIBV main steam isolation bypass valve; NPM NuScale Power Module; PORV power operated relief valve; RCCW reactor component cooling water; RPV reactor pressure vessel; RRV reactor recirculation valve; RVV reactor vent valve; SG steam generator; SO solenoid operated
2. The design does not use safety-related electric power to mitigate accidents or for the safe shutdown of the NPM; therefore, valves listed as having an active safety function have an active-to-failed function to transfer to its safe position on loss of motive power. Valves with an active function are tested by observing the operation of the actuator upon loss of valve actuating power.

3. Cold Shutdown Outage as defined in ISTA-2000 is Mode 3, safe shutdown, with reactor coolant temperatures < 200 °F. The term “cold shutdown” is used throughout Section 3.9.6 for clarity with the OM Code requirements (Section 3.9.6.4.1).

4. Valve Groups: Valves are grouped as required by OM Mandatory Appendices I, II and IV. Pressure Relief Devices (Mandatory Appendix I) are grouped by valve type, valve function, and Code class. Check valves (Mandatory Appendix II) are grouped by valve type and valve size. The POVs (Mandatory Appendix IV) are grouped by actuator type, obturator type, and valve size.

Mandatory Appendix	Group No.	Valve	No. of Components
IV	POV 1	Pneumatic, ball	2
IV	POV 2	Pneumatic/hydraulic, small actuator-type, ball	16
IV	POV 3	Pneumatic/hydraulic, small actuator-type with dump valve, ball	8
IV	POV 4	Pneumatic/hydraulic, large actuator-type, ball	2
IV	POV 5	Pneumatic, flow control	2
IV	POV 6	PORV with remote pilot valves	4
IV	POV 7	Pneumatic, gate, 12-inch	2
IV	POV 8	Pneumatic, gate, 4-inch	2
II	CKV 1	Nozzle check, ≤ 4 inches	4
I	PRD 1	Class 1 safety	2
I	PRD 2	Class 2 thermal relief	2
I	PRD 3	Non-code class, nonreclosing device (rupture disk)	2
I	PRD 4	Non-code class, nonreclosing device (blowout panel, size 1)	6
I	PRD 5	Non-code class, nonreclosing device (blowout panel, size 2)	6

5. The CVCS Makeup and Module Isolation Valves (Section 9.3.4): The common DWS and LRWS supply line to the CVCS makeup pumps includes two safety-related CVCS demineralized water supply isolation valves in series. The valves close on an MPS signal to mitigate an inadvertent boron dilution event.

6. Primary System Containment Isolation Valves (Section 6.2.4, Figure 6.2-4): The PSCIVs are HO to open, nitrogen gas to close. These valves are located on nozzle penetrations on the CNV head and are intended to satisfy the requirements of GDC 55 and 56. The PSCIVs are designed with two valve actuators installed in a single valve body that is welded directly to a CITF that is welded directly to the CNV nozzle safe-end. The valves close automatically on an MPS signal or loss of power to isolate containment and preserve RCS inventory. When the valve is deenergized, parallel hydraulic vent paths open, allowing fluid to vent from the valve actuator. The hydraulic vent paths allow the nitrogen gas cylinder to overcome hydraulic pressure to close the valve. The nitrogen cylinder is sealed and its pressure monitored by plant instrumentation, with alarms and indication available in the MCR. The exercise test and performance assessment test (Note 16) shall determine the state of the nitrogen cylinder (pressure, temperature), the state of the obturator (stroke time, diagnostics), and the state of each hydraulic vent path (by testing each vent path individually). These valves have a CITF that allows 10 CFR 50 Appendix J, Type C testing locally in the direction of containment accident pressure.

7. The RCCW CIVs (Section 6.2.4) cannot be full stroked during normal operation because this will interrupt cooling flow the CRDMs. The CRDMs operate in the containment vacuum and depend on RCCW cooling for heat removal. Interrupting cooling flow to the CRDMs can cause overheating and lead to a possible rod drop.

8. Feedwater Isolation Valves (Section 6.2.4, Figure 6.2-5b): The FWIVs are HO to open, nitrogen gas to close. These valves are located on two nozzle penetrations on the CNV head and are intended to satisfy the requirements of GDC 57. The FWIV is designed with the actuator installed inboard and a feedwater check valve installed outboard in the same valve body. The valve is welded directly to a CITF that is welded directly to the CNV nozzle safe-end. The

valves close automatically on an MPS signal or loss of power to isolate the feedwater line and preserve DHRS inventory. When the valve is deenergized, parallel hydraulic vent paths open, allowing fluid to vent from the valve actuator. This allows the nitrogen gas cylinder to overcome hydraulic pressure and close the valve. The nitrogen cylinder is sealed and its pressure monitored by plant instrumentation, with alarms and indication available in the MCR. The exercise test and performance assessment test (Note 19) shall determine the state of the nitrogen cylinder (pressure, temperature), the state of the obturator (stroke time, diagnostics), and the state of each hydraulic vent path (by testing each vent path individually). These valves have a CITF that allows Technical Specification leakage testing (Note 15) locally in the direction of DHRS pressure. These valves cannot be full-stroke or part stroke exercised during plant operation because closing the valves interrupts feedwater flow resulting in possible SG level transients and may initiate a turbine or NPM trip.

9. Feedwater Check Valves (Section 6.2.4, Figure 6.2-5b): The feedwater check valves are credited for rapidly acting to the safety function position (closed) to preserve DHRS inventory on a loss of feedwater. The FCVs are normally closed nozzle check valves. The FWIV is credited with providing the primary DHRS/feedwater boundary and has specific leakage criteria. The FCV maintains the DHRS boundary until the FWIV is fully closed and therefore, has no specific leakage criteria. The FCV is located in the same valve body as the FWIV and is located outboard of the two (FWIV located nearest the CNV). These valves cannot be full-stroke or part stroke exercised closed during plant operation because closing the valves interrupts feedwater flow resulting in possible SG level transients and may initiate a turbine or NPM trip. The FWIV/FCV body is equipped with a CITF that allows leakage testing following system shutdown. The closed exercise-test (safety-function position) will be performed at cold shutdown with a leak test to verify the valve is fully closed. Normal feedwater operation satisfies the open exercise (nonsafety-function position) for these valves pursuant to ISTC-3550, Valves in Regular Use, at a frequency that satisfies requirements of the OM Code by periodically measuring FW flow to confirm the valve obturator is fully open.

10. Main Steam Isolation Valves and Bypass (Section 6.2.4, Figure 6.2-5a): The MSIVs and MSIBVs are HO to open, nitrogen gas to close. These valves located on the two MS nozzle penetrations on the CNV head and are intended to satisfy the requirements of GDC 57. One actuator is located in a single valve body that is welded to a ASME Class 2 pipe. The valves close automatically on an MPS signal or loss of power to isolate the main stream line and preserve DHRS inventory (the MSIBV is normally closed). When the valve is deenergized, parallel hydraulic vent paths open allowing fluid to vent from the valve actuator. This allows the nitrogen gas cylinder to overcome hydraulic pressure and close the valve. The nitrogen cylinder is sealed and its pressure monitored by plant instrumentation, with alarms and indication available in the MCR. The exercise test and performance assessment test (Note 16) shall determine the state of the nitrogen cylinder (pressure, temperature), the state of the obturator (stroke time, diagnostics), and the state of each hydraulic vent path (by testing each vent path individually). These valves have a CITF that allows leakage testing (Note 15) locally in the direction of steam flow and DHRS isolation. These valves cannot be full-stroke or part stroke exercised during plant operation because closing the valves interrupts steam flow resulting in possible SG pressure and level transients and may initiate a turbine or NPM trip.

11. Decay Heat Removal System Actuation Valves (Section 5.4.3): DHRS actuation valves are HO to close, nitrogen gas to open. These valves are located on two closed loops outside the CNV and are intended to satisfy the requirements of GDC 57. There are two valves in parallel in each loop, four valves total. Either valve is designed to fulfill the system safety function requirement. The valves open automatically on an MPS signal or loss of power to initiate decay heat removal circulation through the DHRS condenser and corresponding SG. When the valve is deenergized, parallel hydraulic vent paths open allowing fluid to vent from the valve actuator. The hydraulic vent paths allow the nitrogen gas cylinder to overcome hydraulic pressure to open the valve. The nitrogen cylinder is sealed and its pressure monitored by plant instrumentation, with alarms and indication available in the MCR. The exercise test and performance assessment test (Note 16) shall determine the state of the nitrogen cylinder (pressure, temperature), the state of the obturator (stroke time, diagnostics), and the state of each hydraulic vent path (by testing each vent path individually). These valves cannot be full-stroke or part stroke exercised during plant operation because opening the valves would unnecessarily subject the SG nozzles to thermal transients from the decay heat condenser condensate flow.

12. Emergency Core Cooling Valves (Section 6.3.2, Figure 6.3-3): The emergency core cooling system, RRVs, and RVVs are pilot-operated relief valves that each consist of a main valve, two trip pilots, a reset pilot and an IAB. These components are considered as one valve assembly and are exercised tested as a unit during cold shutdown. However, performance assessment testing per Note 16 may be performed separately to provide additional diagnostic information to assess valve performance. These valves cannot be full or partial stroke exercised during plant operation because cycling a valve opens an RCS vent path resulting in a potential loss of core cooling. The active safety function of the valves is to open and remain open when actuated. The closed safety function to

support the RCPB is passive. The reset pilot is a nonsafety function and is not tested as part of the ASME OM Code IST program. The trip valve is tested during failsafe and exercise testing.

The RRVs and RVVs do not have specific leakage criteria. Seat tightness is in accord with the requirements of the OM Code Mandatory Appendix I. The ECCS valve seat leakage will be RCS unidentified leakage and must meet Technical Specification surveillance criteria. The seat tightness criteria is accordance with the methods prescribed in OM Code Mandatory Appendix I, Table I-8220-1. The associated pilot valve bodies form part of the reactor coolant and containment boundaries and are subject to 10 CFR 50 Appendix J Type B testing. The IAB is a subcomponent of the ECCS valve and is subject to performance assessment testing.

ISTC-5110 Power Operated Relief Valves - RRVs and RVVs have attributes of both power operated valves (ISTC-5100) and relief valves (ISTC-5240). Performance assessment testing per Note 16 includes a functional test of the IAB at normal RCS pressure to confirm that the ECCS valve does not open. Testing also includes an operational test to demonstrate that the valves not exercise tested will open on low RCS pressure even though the trip valves remain energized (closed).

13. Reactor Safety Valves (Section 5.1.3): These valves are not exercised for IST; their position indication components are tested by local inspection without valve exercise. The RSVs do not have specific leakage criteria. Seat tightness will be in accord with the requirements of the OM Code Mandatory Appendix I. The RSV seat leakage will be RCS unidentified leakage and must meet Technical Specification surveillance criteria. As-left seat tightness criteria shall be no observed leakage utilizing the methods prescribed in OMCode Mandatory Appendix I, Table I-8220-1.

14. Steam Generator System Thermal Relief Valves (Section 5.4.1): These thermal relief valves are located inside containment on each SG system feedwater header.

15. Secondary systems containment isolation valves close to complete the DHRS boundary. These valves have specific leakage criteria and are tested per Technical Specification surveillance test (Technical Specification SR 3.7.1.2 and SR 3.7.2.2).

16. These valves are subject to performance assessment testing per the requirements of 10 CFR 50.55a. The test frequencies are to be established in accordance with the intent of OM Code Mandatory Appendix IV. The approach detailed in Mandatory Appendix IV shall be applied to AOVs, HOVs, and ECCS valves.

    Mandatory Appendix IV and this Plan provides a level of safety equivalent to Subsection ISTC and as required by 10 CFR 50.55a(b)(3)(iii)(A). Section 3.9.6.3.2 (3) contains the factors to be considered in the evaluation of performance assessment testing. Section 3.9.6.4.2 shall be used to determine emergency core cooling system IAB test method and frequency.

17. Feedwater Regulating Valves (Section 10.4.7): The FWRVs are nonsafety-related, not risk-significant backup isolation valves to the safety-related FWIVs and are credited in safety analysis. These valves have the same design pressure and temperature as the RCS. These valves cannot be full-stroke or part-stroke exercised during plant operation because closing the valves interrupts feedwater flow, resulting in possible SG level transients, and may initiate a turbine or NPM trip.

18. Backup Feedwater Check Valves (Section 10.4.7): The backup feedwater check valves are nonsafety-related, not risk-significant backup check valves to the safety-related FCV and are credited in safety analysis. These valves are credited for rapid acting to the safety function position (closed) to preserve DHRS inventory on a loss of feedwater. The backup FCVs are normally closed, nozzle check valves. The FWRV is credited with providing the backup DHRS/ feedwater boundary and has specific leakage criteria. The backup FCV maintains the DHRS boundary until the FWRV is fully closed and, therefore, has no specific leakage criteria. These valves cannot be full-stroke or part-stroke exercised closed during plant operation because closing the valves interrupts feedwater flow, resulting in possible SG level transients, and may initiate a turbine or NPM trip. The nozzle check design is a spring-to-close design. Nonintrusive testing can be used to verify valve closure (safety-function position) during cold shutdown. Normal feedwater operation satisfies the open exercise (nonsafety function position) for these valves pursuant to ISTC-3550, Valves in Regular Use, at a frequency that satisfies requirements of the OM Code by periodically measuring FW flow and pressure to confirm the valves are fully open.

19. Backup Main Steam Isolation Valves and Bypass (Section 10.3.2): The backup MSIVs and MSIBVs are nonsafety-related, not risk significant backup isolation valves to the safety-related MSIVs and MSIBVs and are credited in safety analysis. These valves have the same design pressure and temperature as the RCS. These valves cannot be full-stroke or part stroke exercised during plant operation because closing the valves would interrupt steam flow resulting in possible SG pressure and level transients and may initiate a turbine or NPM trip.
20. SGS-RV-0102/0202 provide thermal overpressure protection to the SGS and DHRS when the MS and FW systems are water solid during startup and shutdown. The closed safety function is to provide SGS closed loop integrity. The open safety function is thermal overpressure protection. The valves do not meet the criteria for Appendix J leak rate testing because these valves relieve to containment. In addition, the valve setpoint is approximately 1,000 psi higher than peak operating pressure, and the leakage criteria for replacement valves should be specified so that reliable operation can be reasonably assured for the ten year period.
21. These nonreclosing devices open to relieve pressure inside the RXB to maintain structural integrity during design basis events. These non-Code Class components meet the criteria of ISTA-1100.

Table 3.9-18: Valve Augmented Requirements

Valve No.	Description	Valve / Actuator[1]	Function Position	Augmented Function(s)[2]	ASME Class / IST Category	IST Type[3]	Notes
Chemical Volume and Control System							
CVC-AOV-0336	CVCS Discharge Isolation Valve	BALL Remote AO	Closed	Active BDBE Containment Isolation	Class 3 Category A	Position Verification Test Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	5
CVC-SV-0404	NDS Supply to Reactor Module Isolation Valve	GLOBE Remote SO	Closed	Active BDBE Containment Isolation	Class 3 Category A	Position Verification Test Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test	5
CVC-CKV-0329	CVCS Injection Check Valve	Nozzle Check	Closed	Active BDBE Containment Isolation	Class 3 Category A/C	Check Exercise/Cold Shutdown Leak Test	4, 5
CVC-CKV-0323	Pressurizer Spray Check Valves	Nozzle Check	Closed	Active BDBE Containment Isolation	Class 3 Category A/C	Check Exercise/Cold Shutdown Leak Test	4, 5

Notes:

1. AO air operated; CVCS chemical volume and control; NPM NuScale Power Module; RPV reactor pressure vessel
2. Valves with augmented test requirements are relied on in the safety analyses, and these components either provide a nonsafety backup to a safety-related function or are nonsafety-related that provide an augmented quality function. The design does not use safety-related electric power to mitigate accidents or for the safe shutdown of the NPM; therefore, all valves listed have an active-to-failed function to transfer to its backup position on loss of motive power. Valves with an active function are tested by observing the operation of the actuator upon loss of valve actuating power.
3. Cold Shutdown Outage as defined in ASME OM Code, Paragraph ISTA-2000 is Mode 3, safe shutdown, with all reactor coolant temperatures < 200 degrees F. The term "cold shutdown" is used throughout Section 3.9.6 for clarity with the OM Code requirements (Section 3.9.6.4.1).
4. Backup CVCS Check Valves: The backup CVCS check valves are normally closed, nozzle check valves. These valves cannot be full- stroke or part-stroke exercised closed during plant operation because system flow must be reversed to demonstrate valve closure. The nozzle check design is a spring-to-close design. Nonintrusive testing can be used to verify valve closure (safety-function position) at cold shutdown. Normal CVCS operation satisfies the open (nonsafety-function position) exercise for these valves pursuant to ISTC-3550, Valves in Regular Use, at a frequency that satisfies the requirements for augmented testing by periodically measuring line flow and pressure to confirm the valves are fully open.
5. Backup Containment Isolation Valves: Third isolation valves that provide a nonsafety backup function as defined by Regulatory Guide 1.26, C.2(c) footnote 6 as having “high leaktight integrity.” These valves define the NRC Quality Group C/D and Seismic I/III classification break. The power operated valves receive a nonsafety containment isolation signal.

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

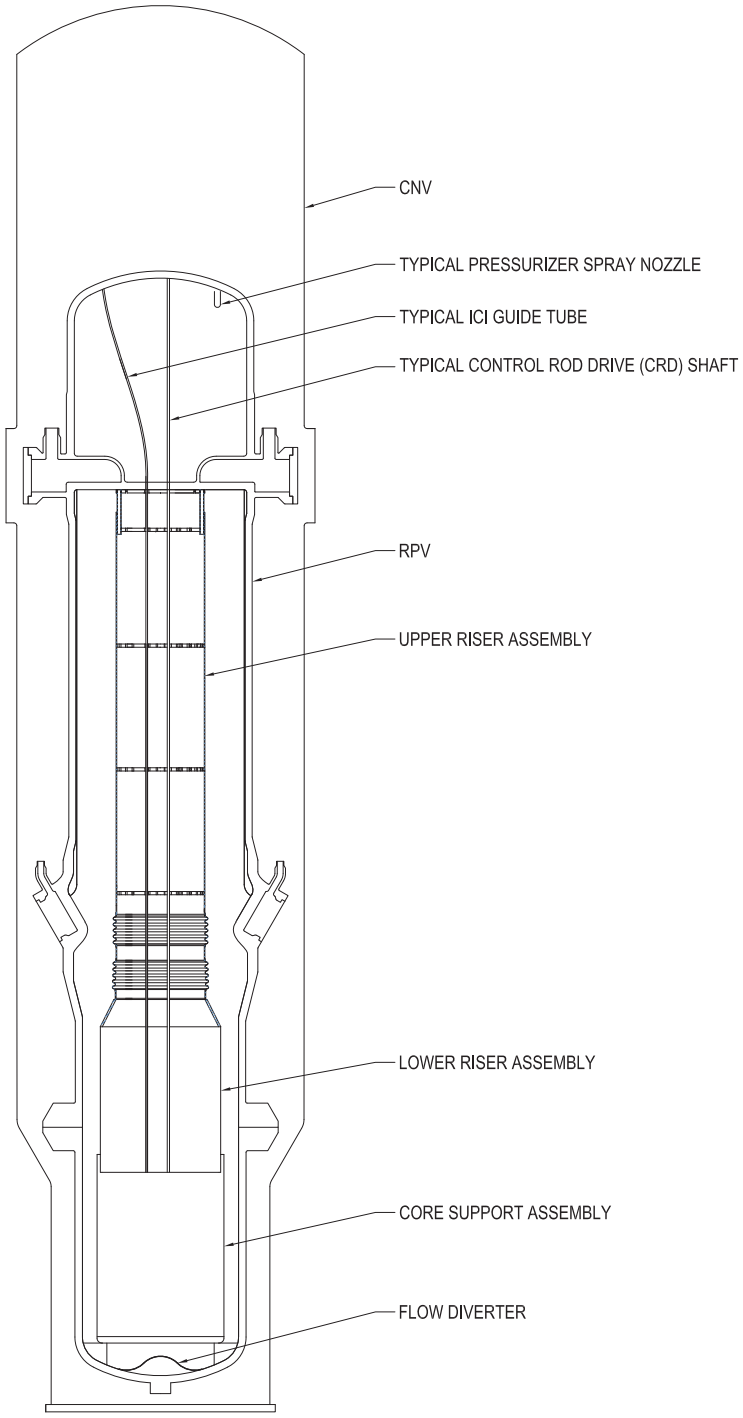


Figure 3.9-2: Upper Riser Assembly

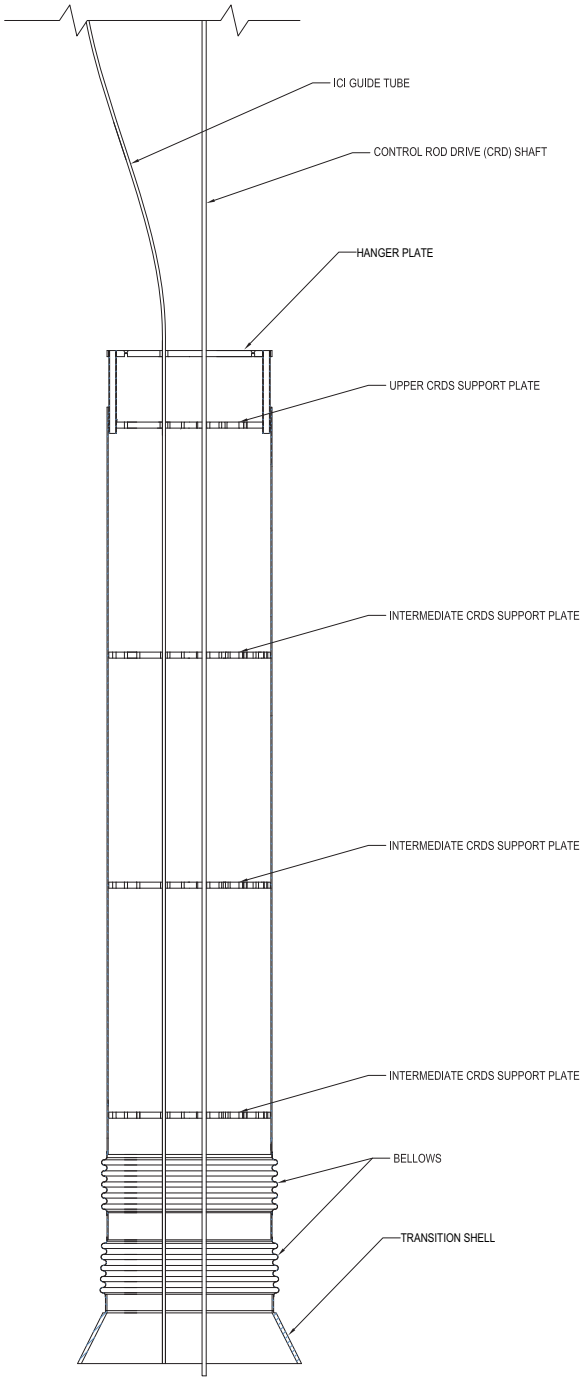


Figure 3.9-3: Lower Riser Assembly

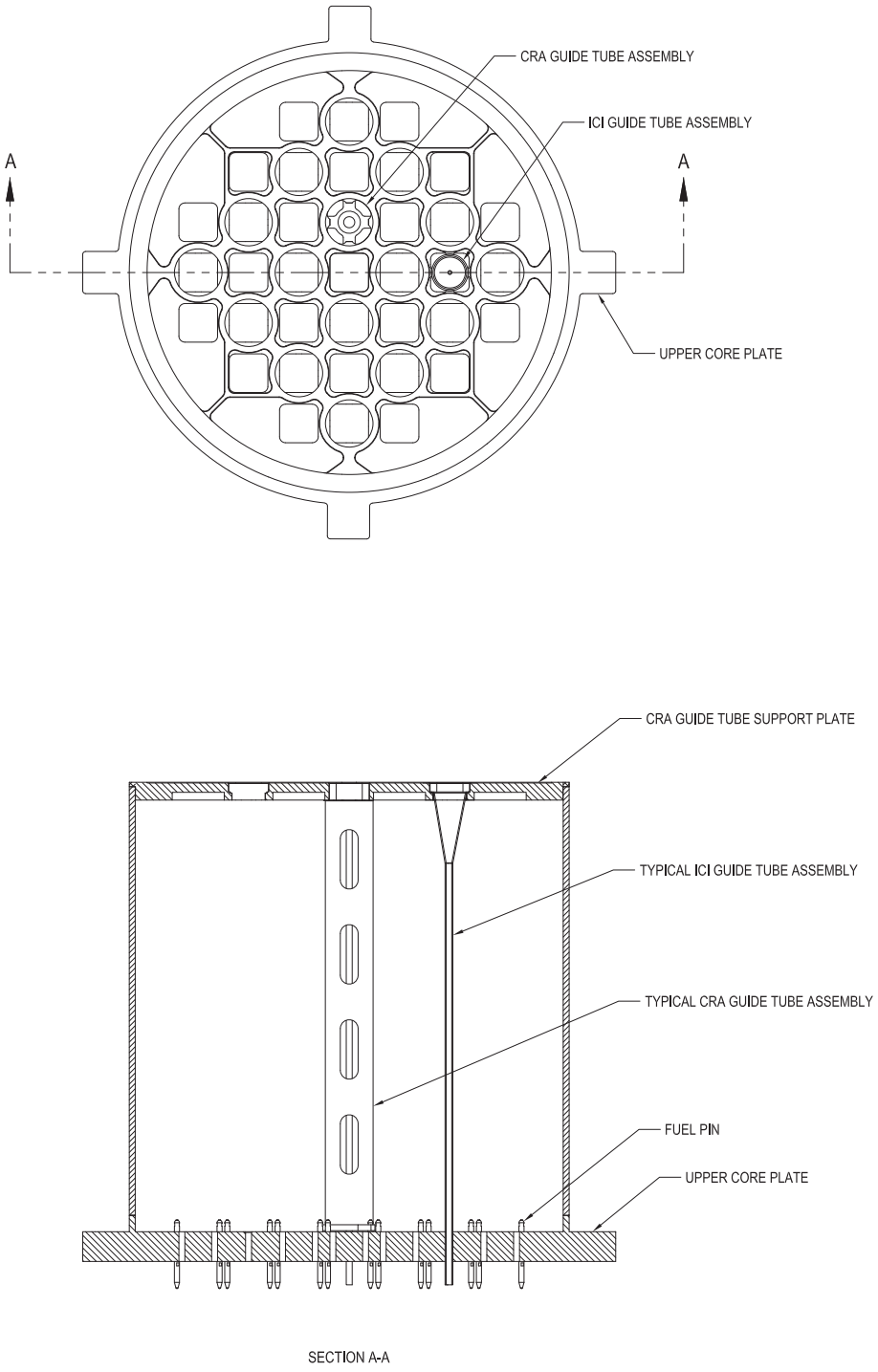
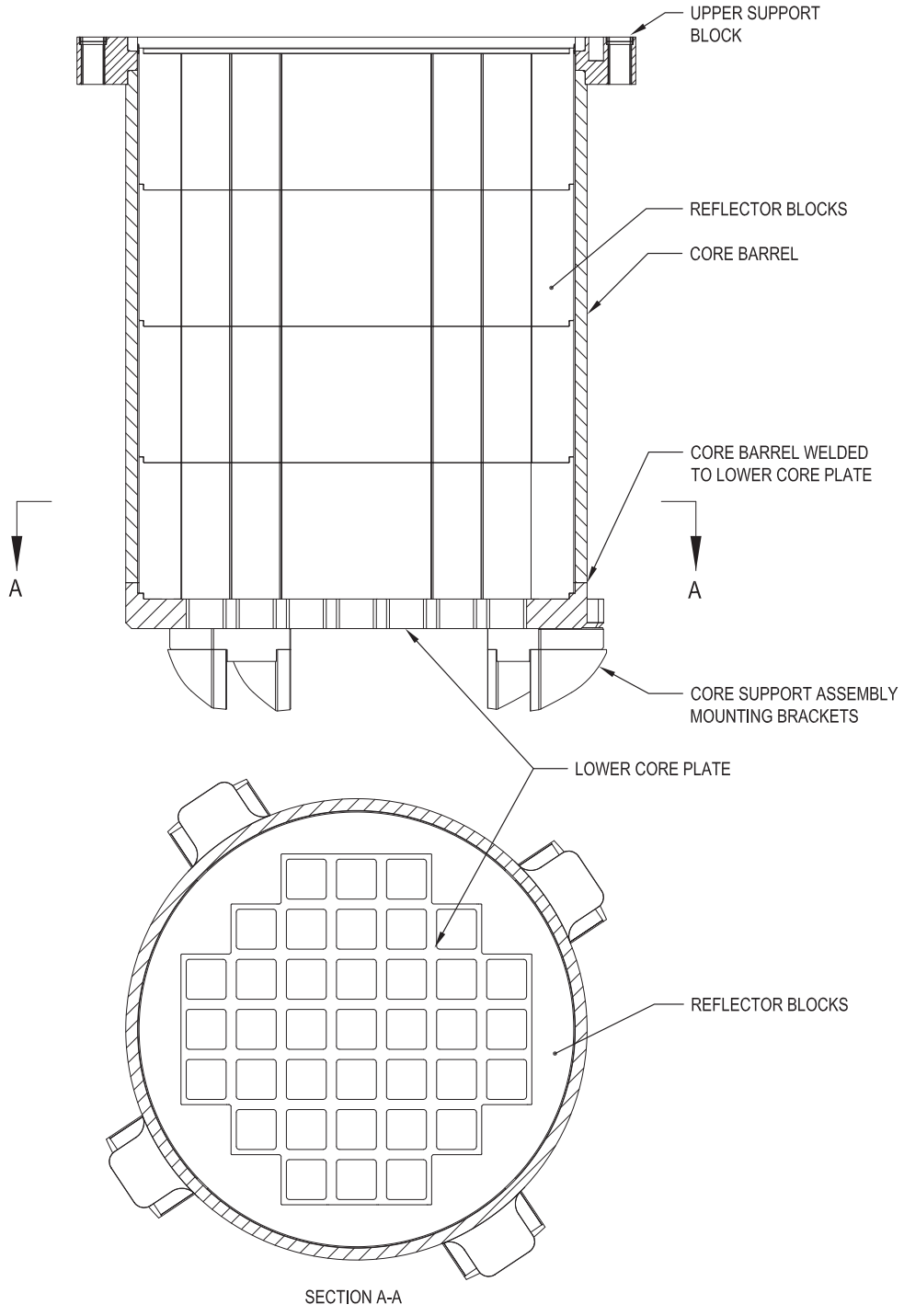


Figure 3.9-4: Core Support Assembly



3.10 Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment

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

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

The design of Seismic Category II structures, systems, and components prevents unacceptable structural failure of, or interaction with, Seismic Category I items during a safe shutdown earthquake (SSE).

The equipment to be qualified includes equipment necessary for safe shutdown, emergency core cooling, containment heat removal, containment isolation, and for mitigating the consequences of accidents or preventing a significant release of radioactive material to the environment. Also included is equipment in the reactor protection system, the engineered safety features and other support systems that directly or indirectly support the performance of one or more of the above safety functions.

Seismic qualification of the containment vessel, reactor pressure vessel, upper reactor vessel internals, lower reactor vessel internals and reactor core, and control rod drive mechanisms is addressed in Appendix 3A. Seismic design and analysis of Seismic Category I structures are addressed in Section 3.7 and Section 3.8. Seismic qualification of the Reactor Building crane and the bioshield are addressed in Section 3.7.3.

3.10.1 Seismic Qualification Criteria

3.10.1.1 Qualification Standards

Section 3.10.2 describes the methodologies for seismic and dynamic qualification of mechanical and electrical equipment. These methods comply with the requirements of General Design Criterion (GDC) 1, GDC 2, GDC 4, GDC 14, GDC 30, and 10 CFR 50 Appendix S. Chapter 17 describes the methods used to implement the requirements of 10 CFR 50, Appendix B.

The design requirements of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (Reference 3.10-2) ensure the structural integrity of Seismic Category I pressure boundary components. Other Seismic Category I equipment is qualified in accordance with Institute of Electrical and Electronics Engineers (IEEE) 344-2013 standard (Reference 3.10-1) endorsed by Regulatory Guide (RG) 1.100.

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

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

The seismic qualification of active mechanical equipment also uses the methods and requirements of American Society of Mechanical Engineers (ASME) QME-1-2017 (Reference 3.10-3) as described in RG 1.100.

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

The operating basis earthquake (OBE) is defined as one third of the SSE. As such, the OBE is eliminated from explicit analysis or design per Appendix S of 10 CFR 50. The consideration of low-level seismic effects (fatigue), required by IEEE 344-2013 (Reference 3.10-1) to qualify electrical and mechanical equipment, utilizes two SSE events, with 10 maximum stress-cycles each, for a total of 20 full cycles. This approach is considered equivalent to the cyclic load basis of one SSE and five OBEs, as outlined in Section 3.7.3.

The environmental qualification program is described in Section 3.11. The methodology for seismic analysis of systems is provided in Section 3.7.3. In accordance with the guidance of RG 1.100, Table 3.9-16 provides a list of safety-related active valves.

3.10.1.2 Performance Requirements for Seismic Qualification

Logical groupings of equipment classified as Seismic Category I have an equipment qualification record file (EQRF) developed. Section 3.11 and Appendix 3C provide the environmental conditions of the mechanical and electrical equipment, including the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents. The equipment specification(s) define the performance requirements for the electrical equipment and instrumentation. The EQRF identifies and evaluates the test response spectrum (TRS) and required response spectrum (RRS) for the seismic qualification. The RRS is bounded by the TRS to demonstrate the conservative qualification of equipment.

The corresponding design specification or equipment qualification specification define the performance requirements for Seismic Category I active mechanical equipment. Non-active Seismic Category I mechanical equipment have a single performance requirement - to maintain their structural integrity.

3.10.1.3 Performance Criteria

The qualification of Seismic Category I mechanical and electrical equipment demonstrates that the equipment is capable of performing its safety-related function under applicable plant loading conditions, including the SSE as defined in Section 3.7, in concert with other concurrent loadings.

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

The guidance and requirements of RG 1.100 and IEEE 344-2013 (Reference 3.10-1) are the source of the methods and procedures used for seismic and dynamic qualification of mechanical and electrical equipment. ASME QME-1-2017 (Reference 3.10-3) is used with the exceptions noted in RG 1.100 for the qualification of active mechanical equipment.

The Seismic Category I equipment is qualified to withstand the SSE in combination with other relevant static and dynamic loads without adverse impacts to the safety functions. Section 3.9.3 defines the acceptable load combinations for mechanical equipment.

Seismic Category I instrumentation and electrical equipment is qualified by methods consistent with IEEE 344-2013. The choice of qualification method is a function of factors such as expense, viability, equipment complexity, and previous seismic qualification data. The EQRF identifies the qualification method for a particular instrument or piece of electrical equipment.

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

3.10.2.1 Qualification by Testing

Seismic qualification of mechanical and electrical equipment by testing is performed in accordance with the requirements of IEEE 344-2013 (Reference 3.10-1). For equipment qualified by testing, the test simulates normal loadings, such as thermal and flow-induced loads, concurrently with the seismic and other dynamic loadings. The loads include forces imposed by piping onto the equipment. The survival and operability of the equipment is verified during and after the testing.

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

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

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

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

For the seismic and dynamic portion of the loads, the test input motion should be applied to one vertical axis and one principal horizontal axis (or two orthogonal horizontal axes) simultaneously, unless it can be demonstrated that the equipment response in the vertical direction is not sensitive to the vibratory motion in the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. An acceptable alternative is to test with vertical and horizontal inputs in-phase, and then repeat the test with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.

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

3.10.2.2 Qualification by Analysis

Qualification by analysis is limited to equipment that must only maintain its structural integrity to perform its safety function. A methodology for calculating the fatigue associated with aging and OBEs is described in IEEE 344-2013 (Reference 3.10-1). The methods of qualification by analysis are dynamic analysis and static coefficient analysis. The analysis accounts for the complexity of the equipment and accurately represents the response of the equipment to seismic

excitation. The two methods of analysis are described below. The analysis shows that the fatigue-inducing effects of the OBEs in combination with other normal, fatigue-inducing operational loads followed by an SSE do not cause the failure of the analyzed equipment to perform its safety function.

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

Dynamic Analysis

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

Static Coefficient Analysis

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

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

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

The following are other analyses that can be used for qualification to

- determine the input response of sub-assemblies or sub-components of equipment subject to testing.
- determine whether the natural frequency of the pump shaft or rotor is within the frequency range of the vibratory excitations.

- determine the differential pressure acting on a valve disc that considers system arrangement and valve closing dynamics, including the differential pressure and impact energy effects of a loss-of-coolant accident.
- verify the resultant maximum calculated stress in the valve body is within the limits defined in ASME Section III.

3.10.2.3 Qualification by Testing and Analysis

When testing or analysis alone are not practical to sufficiently qualify equipment, combined testing and analysis methods are used. The requirements of IEEE 344-2013 (Reference 3.10-1) are used to perform equipment qualification by combined testing and analysis. Operability and structural integrity of components are demonstrated by calculating component deflections and stresses under various loads. These results are then compared to the allowable levels, per the applicable codes.

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

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

Installed equipment (or equivalent equipment with the same inertial mass effects and dynamic coupling to the equipment mounting) is used for qualification of supports for electrical equipment and instrumentation, which includes electrical cabinets, control consoles, electrical panels, and instrument racks. The stresses and deflections are compared to the applicable codes and regulations. When testing is not practical, equipment may be analyzed to confirm its structural integrity. The analysis accounts for the complexity of the supports and accurately represent the response to seismic excitation and vibratory motions.

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

The mounting location determines the input motion the equipment is subjected to for the qualification test. Equipment supports are tested using the same methodology employed to qualify equipment. If the equipment is installed in a non-operational mode for the support test, the support's response during the test at the location of the equipment's mounting is monitored and characterized by an RRS used for separate functional qualification of the equipment. In such a case, equipment should be tested separately for functionality, and the TRS for the equipment must be more

conservative in amplitude and frequency content than the monitored response from the support test.

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

3.10.4 Test and Analysis Results and Equipment Qualification Record Files

The results of seismic qualification testing and analysis, per the criteria in Section 3.10.1, Section 3.10.2 and Section 3.10.3 are included in the corresponding EQRFs. The EQRF files are created and maintained during the equipment selection and procurement phase for the equipment requiring qualification. The EQRF provides a detailed description of the equipment and their support structures, qualification methodology, test and analysis results. The EQRFs are updated and maintained current and auditable. The EQRF data are maintained for the life of the plant. Information to be included in the EQRFs include the following:

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

3.10.5 References

3.10-1 Institute of Electrical and Electronics Engineers,"IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," IEEE Standard 344-2013, Piscataway, NJ.

3.10-2 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section III, “Rules for Construction of Nuclear Facility Components,” New York, NY.

3.10-3 American Society of Mechanical Engineers, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," ASME QME-1-2017, New York, NY.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

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

This section addresses equipment capable of performing credited functions under normal environmental conditions, anticipated operational occurrences (AOOs), accident, and post-accident environmental conditions.

A review of mechanical, electrical, and I&C equipment associated with systems essential for reactivity control, decay heat removal, PAM, containment isolation, maintenance of reactor coolant system pressure boundary integrity, control room habitability, event severity mitigation or system support functions determines whether the equipment requires environmental qualification to meet their credited function.

Included in this equipment scope is

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

Section 3.10 addresses seismic and dynamic qualification of safety-related electrical and mechanical equipment.

This section discusses compliance with the regulatory requirements cited above as they apply to the Environmental Qualification Program.

- General Design Criterion 1: Components in the scope of this section subject to environmental design and qualification require auditable records to document environmental design and qualification requirements have been met.
- General Design Criterion 2: Components in the scope of this section are designed with consideration of the environmental conditions or effects resulting from natural phenomena as part of the environmental conditions evaluated, including their location within safety designed structures.
- General Design Criterion 4: Components in the scope of this section are designed and qualified to accommodate the effects of, and be compatible with, the

environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

- General Design Criterion 23: Components in the scope of this section subject to environmental design and qualification requirements are designed with consideration of the failure mode of the equipment.
- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criteria III, "Design Control." Component design in the scope of this section complies with Criterion III.
- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criteria XI, "Test Control." The test procedures for the environmental program comply with Criterion XI.
- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," Criteria XVII, "QA Records." The qualification record requirements for the environmental program comply with Criterion XVII.
- 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants." This regulation establishes the specific requirements for the environmental qualification of certain electric equipment located in a "harsh" environment. The scope of 10 CFR 50.49 does not include environmental qualification of electric equipment located in a "mild" environment. A "mild" environment is one that is at no time significantly more severe than the environment during normal plant operation, including AOOs. This section assures conformance to 10 CFR 50.49 for the environmental qualification of electrical equipment performing a credited function that is located in a harsh environment. Appendix 3C contains more details.

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Equipment Identification

Equipment identification includes electrical and mechanical equipment that perform a credited function for a design basis event (DBE) that results in a significant change in environmental conditions within the plant that has the potential to result in environmentally induced common cause failures. Design basis events include infrequent events (IEs), AOOs, and design basis accidents (DBAs) as analyzed within the scope of Section 3.6 and Chapter 15. The identification of equipment that requires environmental qualification includes

- equipment relied upon to detect and mitigate a DBE that produces a harsh environment.
- equipment with credited function relied upon for its ability to achieve or maintain a safe shutdown condition for a DBE that produces a harsh environment.
- certain PAM equipment subject to environmental qualification, as required by 10 CFR 50.49 (b)(3).

The equipment subject to environmental qualification consists of mechanical, electrical, and I&C equipment located in either harsh or mild environments.

Equipment required to be environmentally qualified has one or more of the following credited functions: reactor trip, engineered safeguards actuation, PAM, or containment isolation.

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

Table 3.11-1 provides the list of equipment in harsh environments and requiring environmental qualification. Table 3C-1 shows equipment location zones indicated in Table 3.11-1.

3.11.1.2 Definition of Environmental Conditions

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

Environmental service conditions fall into the following categories.

- A harsh environment
  - is a significant change from normal (including DBE and post-accident conditions) that has the potential to result in environmental or radiation induced common-cause failure mechanisms.
  - is an environment that is the result of events as cited above that significantly alters the environmental parameters of temperature, humidity, submergence, or radiation such as
    - temperature
      - >120-degree F and >18-degree F increase above normal operating conditions.
    - humidity:
      - steam exposure
        - >99 percent relative humidity condensing conditions.
        - ≥85 percent relative humidity with temperatures ≥120 degree F for equipment with solid state circuitry.
    - submergence
      - areas where equipment is subject to submergence that is not subjected to submergence under normal operating conditions.

- Radiation
  - total integrated dose of > 1.0E04 rad (60 year normal plus 30 day accident dose) for electrical and mechanical equipment.
  - total integrated dose of > 1.0E03 rad (60 year normal plus 30 day accident dose) for equipment with solid state circuitry.

- A mild environment
  - areas where the environment at no time is significantly more severe than the environment during normal plant operation, including AOOs.

3.11.1.3 Equipment Post-Accident Operating Time

The post-accident operating time is the period of time, beginning with design basis event initiation, during which the equipment must continue to perform its credited function. The post-accident operating time, or operating time, duration can vary, and is based on the credited function of the equipment. Both operating and “not failing” in a manner detrimental to plant safety are considered in the assignment of post accident operating times.

Table 3.11-1 specifies post accident operating times for the equipment listed. The required post-accident operating time for equipment varies from 1 hour to 720 hours. The conservative bases for the operating times are established for PAM equipment and equipment required for long-term core cooling.

Table 3C-4 defines and documents the four distinct post-accident operating time frames for equipment located in harsh environments listed in Table 3.11-1.

3.11.2 Governing Regulatory and Industry Codes

3.11.2.1 Environmental Qualification of Electrical Equipment

Harsh Environments

For electrical equipment required to function during or following exposure to a harsh environment, compliance with the environmental provisions of GDC 4 are achieved by demonstrating compliance with 10 CFR 50.49 in accordance with Design Specific Review Standard 3.11, regulatory guidance, and industry standards. Table 3.11-2 lists applicable regulatory guides and industry standards for environmental qualification.

Mild Environments

For electrical equipment located in mild environments, compliance with the environmental design provisions of GDC 4 are achieved and demonstrated by proper incorporation of relevant environmental conditions into the design process, including the equipment specification. Regulatory Guide 1.209 endorses IEEE Standard 323-2003 (Reference 3.11-12) for qualification of computer-based I&C systems in mild environments.

3.11.2.2 Qualification of Mechanical Equipment

Mechanical equipment environmental qualification and documentation is in accordance with GDC 1, GDC 2, GDC 4, and GDC 23 as demonstrated by the approach presented in this section. Table 3.11-2 lists applicable regulatory guides and industry standards for environmental qualification.

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

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

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

Equipment that only has a credited function of maintaining its structural integrity, for support or to protect the integrity of a pressure boundary, is qualified in accordance with the guidance of American Society of Mechanical Engineers (ASME) QME-1-2017 (Reference 3.11-13).

For mechanical equipment located in a mild environment, acceptable environmental design is demonstrated by conformance with the design and purchase specifications for the equipment. The specifications contain a description of the equipment functional requirements for a specific environmental zone during normal environmental conditions and AOOs.

For mechanical equipment that must function during or following exposure to a harsh environment, demonstrating the non-metallic parts and components of the equipment are suitable for the postulated design basis environmental conditions ensures compliance with the environmental design provisions of GDC 4. Qualification of safety-related mechanical equipment that performs an active function during or following exposure to harsh environmental conditions is in accordance with Appendix QR-B of Reference 3.11-13, as endorsed by RG 1.100, Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants, with the following exceptions.

- exception to QR-B5200, Identification and Specification of Qualification Requirements, (g) material activation energy
- exception to QR-B5300, Selection of Qualification Methods for determination and recording of shelf life of nonmetallics
- exception to QR-B5500 Documentation, (h) shelf life preservation requirements

Section 3C.3 provides the details of the exceptions to Reference 3.11-13.

The design of mechanical equipment located in harsh environments ensures equipment performance under appropriate environmental conditions. The primary focus with mechanical equipment is on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). Table 3.11-1 lists the mechanical components that contain non-metallic or consumable parts located in harsh environments that require environmental qualification.

3.11.2.3 Justification for Using Latest Institute of Electrical and Electronics Engineers Standards Not Endorsed by a Regulatory Guide

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

The IEEE periodically updates the standards to incorporate evolutionary thinking and approaches of the nuclear industry with regard to environmental qualification.

Table 3.11-2 provides a summary comparison of the current IEEE standards for environmental qualification and the associated RGs that endorse them. Recent IEEE standards, not currently endorsed by the Nuclear Regulatory Commission, are discussed and justified below.

3.11.2.3.1 Institute of Electrical and Electronics Engineers Standard 741-1997

Regulatory Position C of RG 1.63 endorses Section 5.4 of IEEE Standard 741-1986 (Reference 3.11-1) for external circuit protection of electric penetration assemblies. The design incorporates IEEE Standard. 741-1997 regarding protection of Class IE equipment (Reference 3.11-4) as the design philosophy does not deviate from the existing RG.

3.11.3 Qualification Test Results

The equipment qualification record file documents the summaries and results of qualification tests for electrical and mechanical equipment located in harsh environments.

Qualification of equipment in mild environments is based on certification of performance in accordance with applicable regulatory guidance. Section 3.11.2 and Table 3.11-2 identify guidance that governs documentation of the summaries and results of seismic qualification tests for electrical and mechanical equipment and components in the harsh environment areas. Appendix 3C provides additional information.

COL Item 3.11-1: An applicant that references the NuScale Power Plant US460 standard design will submit a full description of the Environmental Qualification Program and milestones and completion dates for program implementation.

3.11.4 Estimated Chemical and Radiation Environment

3.11.4.1 Chemical Environments

Appendix 3C defines applicable chemical environments for normal and AOO conditions. The chemical environments from the most limiting design basis event are also considered in the qualification of the equipment and are presented in Appendix 3C.

Chemicals used for water chemistry and pH control are considered as well as the borated water environment that is present inside containment and outside containment. Section 5.2.3 discusses primary side water chemistry, Section 6.1.1 discusses reactor pool and spent fuel pool chemistry, and Section 10.3.5 discusses secondary side water chemistry.

3.11.4.2 Radiation Environments

Appendix 3C defines radiation environments for normal and accident conditions.

Normal operation radiation dose calculations use the source terms and analysis. The radiation doses are monitored during plant life and compared to the calculated doses. If the measured doses are higher than the calculated doses, the equipment listed in Table 3.11-1 is evaluated to ensure it remains qualified. Section 12.3 discusses normal operational dose rates.

The normal operations dose rates for environmental qualification are derived from direct gamma radiation emitted by radioactive fluids. Beta radiation and Bremsstrahlung radiation during normal operations are considered negligible contributors to doses in comparison to the gamma radiation and therefore are omitted. Normal doses within the CNV and other areas also account for neutron fluence, when applicable, by equating the neutron fluence to an equivalent dose in rads.

Accident dose rates include a submersion dose and a direct dose contribution. The submersion dose is derived from both the gamma and beta radiation. Beta radiation is attenuated by low-density equipment enclosures. Alpha radiation is neglected from both the normal and accident environmental qualification dose rates because the alpha particle is easily attenuated by air.

Reference 3.11-10 and Section 12.2.1 present the methodology that forms the basis for the accident dose rate calculations. Section 15.0.3 discusses the assumptions associated with the accident dose rates. Appendix 3C provides additional information on normal and accident dose rates used for environmental qualification.

COL Item 3.11-2: An applicant that references the NuScale Power Plant US460 standard design will ensure the Environmental Qualification Program cited in COL Item 3.11-1 includes a description of how equipment located in harsh conditions will be monitored and managed throughout plant life. This description will include methodology to ensure equipment located in harsh environments will remain qualified if the measured dose is higher than the calculated dose.

3.11.5 Environmental Qualification Operational Program

An Environmental Qualification Operational Program ensures continued capability of qualified mechanical and electrical equipment to perform its design function throughout its qualified life. The Environmental Qualification Operational Program contains the following aspects specific to the environmental qualification of mechanical and electrical equipment:

- evaluation of environmental qualification results to establish activities to support continued environmental qualification for the entire time an item is installed in the plant,
- determination of surveillance and preventive maintenance activities based on environmental qualification results,
- consideration of environmental qualification maintenance recommendations from equipment vendors,
- evaluation of operating experience in developing surveillance and preventive maintenance activities for specific equipment,
- development of plant procedures that specify individual equipment identification, appropriate references, installation requirements, surveillance and maintenance requirements, post-maintenance testing requirements, condition monitoring requirements, replacement part identification, and applicable design changes and modifications,
- development of plant procedures for reviewing equipment performance and environmental qualification operational activities, and for trending the results to incorporate lessons learned through appropriate modifications to the Environmental Qualification Operational Program,
- development of plant procedures for the control and maintenance of environmental qualification records, and
- update to Table 3.11-1 to include commodities necessary to support equipment listed in Table 3.11-1. Examples of commodity items that are subject to environmental qualification include, but are not limited to, equipment items such as cables, connectors, electrical splices, conduit seals, thread sealants, terminal blocks, or lubricants.

COL Item 3.11-3: An applicant that references the NuScale Power Plant US460 standard design will implement an Environmental Qualification Operational Program that incorporates the aspects in Section 3.11.5 specific to the environmental qualification of mechanical and electrical equipment. This program will include an update to Table 3.11-1 to include commodities that support equipment listed in Table 3.11-1.

3.11.6 References

3.11-1 Institute of Electrical and Electronics Engineers, “IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations,” Std. 741-1986, Piscataway, NJ.

3.11-2 Institute of Electrical and Electronics Engineers, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," IEEE Standard 323-1974, Piscataway, NJ.

3.11-3 Institute of Electrical and Electronics Engineers, "IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generation Stations," IEEE Standard 317-1983, Piscataway, NJ.

3.11-4 Institute of Electrical and Electronics Engineers, "IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations," IEEE Standard 741-1997, Piscataway, NJ.

3.11-5 Institute of Electrical and Electronics Engineers, "IEEE Standard for Qualification of Safety-Related Actuators for Nuclear Power Generating Stations," IEEE Standard 382-2006, Piscataway, NJ.

3.11-6 Institute of Electrical and Electronics Engineers, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," IEEE Standard 7-4.3.2-2003, Piscataway, NJ.

3.11-7 Institute of Electrical and Electronics Engineers, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," IEEE Standard 603-1991, Piscataway, NJ.

3.11-8 Institute of Electrical and Electronics Engineers, "IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations," IEEE Standard 572-2006, Piscataway, NJ.

3.11-9 Institute of Electrical and Electronics Engineers, “IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations,” IEEE Standard 535-2013, New York, NY.

3.11-10 NuScale Power, LLC, "Accident Source Term Methodology," TR-0915-17565-P-A, Revision 4.

3.11-11 Institute of Electrical and Electronics Engineers, "IEEE Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations," IEEE Standard 383-2003, Piscataway, NJ.

3.11-12 Institute of Electrical and Electronics Engineers, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," IEEE Standard 323-2003, Piscataway, NJ.

3.11-13 American Society of Mechanical Engineers, Qualification of Active Mechanical Equipment Used In Nuclear Facilities, ASME QME-1-2017, New York, NY.

3.11-14 Institute of Electrical and Electronics Engineers, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE Standard 497-2016, Piscataway, NJ.

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
Containment System (A013)						
I&C Division I Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	B,C,D	720
I&C Division II Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	B,C,D	720
PZR Heater Power Division I Nozzle Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	N/A	720
PZR Heater Power Division II Nozzle Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	N/A	720
I&C Channel A Instrument Seal Assembly (ISA)	CNV-6, RXBP-1	Harsh	Electrical Mechanical	A	C	720
I&C Channel C Instrument Seal Assembly (ISA)	CNV-6, RXBP-1	Harsh	Electrical Mechanical	A	C	720
I&C Channel B Instrument Seal Assembly (ISA)	CNV-6, RXBP-1	Harsh	Electrical Mechanical	A	C	720
I&C Channel D Instrument Seal Assembly (ISA)	CNV-6, RXBP-2	Harsh	Electrical Mechanical	A	C	720
CRDM Power 1 Nozzle Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	N/A	720
RPI Group #1 Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	N/A	720
RPI Group #2 Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	N/A	720

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
I&C Separation Group A Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	B,C,D	720
I&C Separation Group B Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	B,C,D	720
I&C Separation Group C Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	B,C,D	720
I&C Separation Group D Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	B,C,D	720
CRDM Power 2 Nozzle Electrical Penetration Assembly (EPA)	CNV-5, RXBP-1	Harsh	Electrical Mechanical	A	N/A	720
Main Steam Isolation Valve (MSIV) #1 and #2	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
MS Isolation Bypass Valve (MSIBV) #1 and #2	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
Feedwater Isolation Valve (FWIV) #1 and #2	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
Feedwater Isolation Check Valve #1 and #2	RXBP-1	Harsh	Mechanical	A B	N/A	1 720
RCCW Supply CIV, Inboard and Outboard	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
RCCW Return CIV, Inboard and Outboard	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
RPV High Point Degas CIV, Inboard and Outboard	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
RPV High Point Degas Solenoid Valve	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
PZR Spray Flow Check Valve	RXBP-1	Harsh	Mechanical	A B	N/A	1 720

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
PZR Spray CIV, Inboard and Outboard	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
CVC Injection Flow Check Valve	RXBP-	Harsh	Mechanical	A B	N/A	1 720
CVC Injection CIV, Inboard and Outboard	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
CVC Discharge CIV, Inboard and Outboard	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
CVC Discharge Air Operated Valve	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
Containment Flood and Drain CIV, Inboard and Outboard	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
Containment Evacuation CIV, Inboard and Outboard	RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
Central Hydraulic Power Unit Skid A and Skid B	RXBG-8	Harsh	Electrical Mechanical	A B	N/A	1 720
Passive Autocatalytic Recombiner (PAR)	CNV-4 or CNV-5	Harsh	Mechanical	B	N/A	720
Containment Narrow Range Pressure Element A/B/C/D	CNV-6	Harsh	Electrical	A	B	720
Containment Wide Range Pressure Element A/B	CNV-	Harsh	Electrical	A	B,C,D	720
Containment Level Indication A/B/C/D	RXBP-1, CNV-1 - CNV-6	Harsh	Electrical	A	N/A	720
SG #1 and SG #2 Main Steam Temperature Indication A/B/C/D	RXBP-1	Harsh	Electrical	A	N/A	720
FWIV #1 Position Indication A / B	RXBP-1	Harsh	Electrical	A	B,C,D	720
FWIV #2 Position Indication A / B	RXBP-1	Harsh	Electrical	A	B,C,D	720

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
FWIV #1 and FWIV #2 Nitrogen Accumulator Pressure Transmitter	RXBP-1	Harsh	Mechanical	B	N/A	720
MSIV #1 Position Indication A / B	RXBP-1	Harsh	Electrical	A	B,C,D	720
MSIV #2 Position Indication A / B	RXBP-1	Harsh	Electrical	A	B,C,D	720
MSIV #1 and MSIV #2 Nitrogen Accumulator Pressure Transmitter	RXBP-1	Harsh	Mechanical	B	N/A	720
MSIBV #1 Position Indication A / B	RXBP-1	Harsh	Electrical	A	B,C,D	720
MSIBV #2 Position Indication A / B	RXBP-1	Harsh	Electrical	A	B,C,D	720
MSIBV #1 and MSIBV #2 Nitrogen Accumulator Pressure Transmitter	RXBP-1	Harsh	Mechanical	B	N/A	720
RCCW Return CIV Position Indicator, Inboard and Outboard	RXBP-1	Harsh	Electrical	A	B,C,D	720
RCCW Return CIV Nitrogen Accumulator Pressure Transmitter, Inboard and Outboard	RXBP-1	Harsh	Mechanical	B	N/A	720
RCCW Supply CIV Position Indicator, Inboard and Outboard	RXBP-1	Harsh	Electrical	A	B,C,D	720
RCCW Supply CIV Nitrogen Accumulator Pressure Transmitter, Inboard and Outboard	RXBP-1	Harsh	Mechanical	B	N/A	720
CVC High Point Degas CIV Position Indicator, Inboard and Outboard	RXBP-1	Harsh	Electrical	A	B,C,D	720

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
CVC High Point Degas CIV Nitrogen Accumulator Pressure Transmitter, Inboard and Outboard	RXBP-1	Harsh	Mechanical	B	N/A	720
RPV High Point Degas Solenoid Valve Position Indication	RXBP-1	Harsh	Electrical	A	N/A	720
PZR Spray CIV Position Indicator, Inboard and Outboard	RXBP-1	Harsh	Electrical	A	B,C,D	720
PZR Spray CIV Nitrogen Accumulator Pressure Transmitter, Inboard and Outboard	RXBP-1	Harsh	Mechanical	B	N/A	720
CVC Injection CIV Position Indicator, Inboard and Outboard	RXBP-1	Harsh	Electrical	A	B,C,D	720
CVC Injection CIV Nitrogen Accumulator Pressure Transmitter, Inboard and Outboard	RXBP-1	Harsh	Mechanical	B	N/A	720
CVC Discharge CIV Position Indicator, Inboard and Outboard	RXBP-1	Harsh	Electrical	A	B,C,D	720
CVC Discharge CIV Nitrogen Accumulator Pressure Transmitter, Inboard and Outboard	RXBP-1	Harsh	Mechanical	B	N/A	720
CVC Discharge Air Operated Valve Position Indication	RXBP-1	Harsh	Electrical	A	B,C,D	720
CES CIV Position Indicator, Inboard and Outboard	RXBP-1	Harsh	Electrical	A	B,C,D	720

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
CES CIV Nitrogen Accumulator Pressure Transmitter, Inboard and Outboard	RXBP-1	Harsh	Mechanical	B	N/A	720
CFD CIV Position Indicator, Inboard and Outboard	RXBP-1	Harsh	Electrical	A	B,C,D	720
CFD CIV Nitrogen Accumulator Pressure Transmitter, Inboard and Outboard	RXBP-1	Harsh	Mechanical	B	N/A	720
STEAM GENERATOR SYSTEM (A014)						
Thermal Relief Valves	CNV-3	Harsh	Mechanical	B	N/A	720
CONTROL ROD DRIVE SYSTEM (A022)						
CRDM Pressure Boundary	CNV-5, CNV-6	Harsh	Mechanical	B	N/A	720
CRDM Magnetic Jack Assembly	CNV-5, CNV-6	Harsh	Electrical Mechanical	A B	N/A	1 720
Control Rod Drive Shaft Assembly	CNV-5, CNV-6	Harsh	Mechanical	A	N/A	720
Rod Hold Out (RHO) Ball Grip Assembly	CNV-5, CNV-6	Harsh	Mechanical	B	N/A	720
CRDS Cooling Water Pressure Relief Valves	CNV-6	Harsh	Mechanical	B	N/A	720
CRDS Cooling Water Secondary Supply Header High Point Vent Valves	CNV-6	Harsh	Mechanical	B	N/A	720
REACTOR COOLANT SYSTEM (A030)						
Wide Range RCS Pressure Element A/B/C/D	CNV-4	Harsh	Mechanical	A	B,C,D	720
RCS Cold Leg Temperature Element A/B/C/D	CNV-2	Harsh	Electrical	B	N/A	720
RCS Hot Leg Temperature Element A/B/C/D	CNV-3	Harsh	Electrical	A	B	720
RCS Flow Elements	CNV-2	Harsh	Electrical	A	B	720

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
Reactor Safety Valves	CNV-5	Harsh	Electrical Mechanical	A	N/A	720
Reactor Safety Valve Position Indicator	CNV-5	Harsh	Electrical	A	NUREG-0737	720
Pressurizer Heaters	CNV-4	Harsh	Electrical Mechanical	A A	N/A	1 720
Pressurizer Level	CNV-5, CNV-6 RXBP-1	Harsh	Electrical Mechanical	A B	N/A	1 720
RPV Riser Level A/B/C/D	CNV-4 CNV-5 CNV-6 RXBP-1	Harsh	Electrical Mechanical	A	B,C,D	720
Narrow Range Pressurizer Pressure Elements A/B/C/D	CNV-4	Harsh	Mechanical	A	N/A	720
CHEMICAL AND VOLUME CONTROL SYSTEM (B010)						
DW Supply to CVC Makeup Upstream Isolation Valve, Upstream and Downstream	RXBG-2	Harsh	Mechanical	A B	N/A	1 720
DW Supply to CVC Makeup Isolation Solenoid Valve, Upstream and Downstream	RXBG-2	Harsh	Electrical	A	N/A	1
DW Supply to CVC Makeup Isolation Valve Position Indicator	RXBG-2	Harsh	Electrical	A	D	720
EMERGENCY CORE COOLING SYSTEM (B020)						
Reactor Vent Valves	CNV-5	Harsh	Mechanical	A B	N/A	36 720
Reactor Recirculation Valves	CNV-2	Harsh	Mechanical	A B	N/A	36 720
Reactor Vent Valve Trip Valves	RXBP-3	Harsh	Electrical Mechanical	A B	N/A	36 720
Reactor Recirculation Valve Trip Valves	RXBP-3	Harsh	Electrical Mechanical	A B	N/A	36 720
Reset Valves	RXBP-3	Harsh	Electrical Mechanical	A B	N/A	36 720

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
RVV Position Indicator	CNV-5	Harsh	Electrical	A	D	720
RRV Position Indicator	CNV-2	Harsh	Electrical	A	D	720
RVV and RRV Trip Valve Position Indicator	RXBP-3	Harsh	Electrical	A	D	720
DECAY HEAT REMOVAL SYSTEM (B030)						
DHRS Actuation Valves	RXPB-1	Harsh	Electrical Mechanical	A B	N/A	1 720
SG Main Steam Pressure Transmitters A/B/C/D	RXPB-1	Harsh	Electrical Mechanical	A B	N/A	1 720
DHRS Actuation Valve Position Indicators	RXPB-1	Harsh	Electrical	A	D	720
DHRS Nitrogen Accumulator Pressure Transmitters	RXPB-1	Harsh	Mechanical	B	N/A	720
DHRS Condenser Outlet Pressure Transmitters A/B/C	RXPB-3	Harsh	Electrical	A	D	720
REACTOR BUILDING VENTILATION (B090)						
RBV Exhaust Air HELB Dampers and Position Indicators	RXBG-9	Harsh	Electrical Mechanical	A B	N/A	1 720
RBV Supply Air HELB Dampers and Position Indicators	RXBG-6	Harsh	Electrical Mechanical	A B	N/A	1 720
RBV Room 010-418 Radiation Element A/B	RXBG-6	Harsh	Electrical	A	N/A	720
RBV Room 010-420 Radiation Element A/B	RXBG-6	Harsh	Electrical	A	N/A	720
RBV Room 010-436 Radiation Element A/B	RXBG-6	Harsh	Electrical	A	N/A	720
RBV Room 010-318 Radiation Element A/B	RXBG-4	Harsh	Electrical	A	N/A	720

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
RBV Room 010-431 Radiation Element A/B	RXBG-6	Harsh	Electrical	A	N/A	720
RBV Room 010-331 Radiation Element A/B	RXBG-4	Harsh	Electrical	A	N/A	720
RBV Room 010-334 Radiation Element A/B	RXBG-4	Harsh	Electrical	A	N/A	720
RBV Room 010-418 Radiation Indicating Transmitter A/B	RXBG-6	Harsh	Electrical	A	N/A	720
RBV Room 010-420 Radiation Indicating Transmitter A/B	RXBG-6	Harsh	Electrical	A	N/A	720
RBV Room 010-436 Radiation Indicating Transmitter A/B	RXBG-6	Harsh	Electrical	A	N/A	720
RBV Room 010-318 Radiation Indicating Transmitter A/B	RXBG-4	Harsh	Electrical	A	N/A	720
RBV Room 010-431 Radiation Indicating Transmitter A/B	RXBG-6	Harsh	Electrical	A	N/A	720
RBV Room 010-331 Radiation Indicating Transmitter A/B	RXBG-4	Harsh	Electrical	A	N/A	720
RBV Room 010-334 Radiation Indicating Transmitter A/B	RXBG-4	Harsh	Electrical	A	N/A	720
RBV Room 010-418 HELB Temperature Switch A/B	RXBG-6	Harsh	Electrical	A	N/A	1
RBV Room 010-420 HELB Temperature Switch A/B	RXBG-6	Harsh	Electrical	A	N/A	1
RBV Room 010-436 HELB Temperature Switch A/B	RXBG-6	Harsh	Electrical	A	N/A	1

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
RBV Room 010-316 HELB Temperature Switch A/B	RXBG-4	Harsh	Electrical	A	N/A	1
RBV Room 010-431 HELB Temperature Switch A/B	RXBG-4	Harsh	Electrical	A	N/A	1
RBV Room 010-331 HELB Temperature Switch A/B	RXBG-4	Harsh	Electrical	A	N/A	1
RBV Room 010-304 HELB Temperature Switch A/B	RXBG-4	Harsh	Electrical	A	N/A	1
FUEL HANDLING EQUIPMENT (B140)						
Fuel Handling Machine Power Cabinet	RXBP-2	Harsh	Electrical	B	N/A	720
Fuel Handling Machine Control Cabinet	RXBP-2	Harsh	Electrical	B	N/A	720
New Fuel Elevator Power Cabinet	RXBP-2	Harsh	Electrical	B	N/A	720
New Fuel Elevator Control Cabinet	RXBP-2	Harsh	Electrical	B	N/A	720
New Fuel Elevator	RXBP-2	Harsh	Electrical Mechanical	B	N/A	720
Fuel Handling Machine	RXBP-2	Harsh	Electrical Mechanical	B	N/A	720
ULTIMATE HEAT SINK (B175)						
Spent Fuel Pool Level Element A/B	RXPB-2	Harsh	Electrical	A	D	720
Spent Fuel Pool Level Indicating Transmitter A/B	RXPB-2	Harsh	Electrical	A	D	720
MAIN STEAM (C010)						
SMSIV Bypass Line Isolation Valve SG1 and SG2	RXBG-8	Harsh	Mechanical	A B	N/A	1 720
SMSIV Bypass Line Isolation Valve Solenoid SG1 and SG2	RXBG-8	Harsh	Electrical	A	N/A	1

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
SMSIV Bypass Line Isolation Valve Position Indicators SG1 and SG2	RXBG-8	Harsh	Electrical	A	D	720
SG1 and SG2 SMSIV	RXBG-8	Harsh	Mechanical	A B	N/A	1 720
SG1 and SG2 SMSIV Position Indicators	RXBG-8	Harsh	Electrical	A	D	720
SG1 and SG2 SMSIV Continuous Position Transmitters	RXBG-8	Harsh	Electrical	B	N/A	720
SG1 and SG2 SMSIV Bypass Valves	RXBG-8	Harsh	Mechanical	A B	N/A	1 720
SG1 and SG2 SMSIV Bypass Valve Position Indicators	RXBG-8	Harsh	Electrical	A	D	720
SG1 and SG2 SMSIV Bypass Valve Continuous Position Transmitters	RXBG-8	Harsh	Electrical	B	N/A	720
CONDENSATE AND FEEDWATER SYSTEM (C020)						
SG1 and SG2 FWRVs	RXBG-8	Harsh	Mechanical	A B	N/A	1 720
SG1 and SG2 FWRV Position Indicators A/B	RXBG-8	Harsh	Electrical	A	D	720
SG1 and SG2 FWRV Solenoid Valves A/B	RXBG-8	Harsh	Electrical	A	N/A	1
SG1 and SG2 FW Check Valves	RXBG-8	Harsh	Mechanical	A	N/A	720
MODULE PROTECTION SYSTEM (E011)						
Division I and Division II MCR Isolation Switches	RXBG-4	Harsh	Electrical	B	N/A	720
Under-the-Bioshield Temperature Sensor B/C	RXBP-1	Harsh	Electrical	A	D	720

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
Under-the-Bioshield Temperature Sensor A/D	RXBP-1	Harsh	Electrical	A	N/A	36
NEUTRON MONITORING SYSTEM (E013)						
NMS Deployment Mechanism HPU and Control Skid	RXBG-8	Harsh	Electrical Mechanical	A B	B, D	720
Operating Bay NMS Positioning Support Mechanism, Separation Groups A/B/C/D	RXBP-3	Harsh	Electrical Mechanical	A B	N/A	720
Operating Bay NMS Positioning Support Mechanism NMS-Flood Separation Groups B/C	RXPB-3	Harsh	Electrical Mechanical	A B	D	720
NMS-Excore Neutron Detector and Moderator Assembly Separation Groups A/B/C/D	RXPB-3	Harsh	Electrical	A	B, D (Groups B and C only)	720
NMS-Flood Neutron Detector and Moderator Assembly Separation Groups B/C	RXPB-3	Harsh	Electrical	A	B, D	720
INCORE INSTRUMENTATION SYSTEM (E034)						
ICI Stringer Assembly 01 through 12	RXBP-1, CNV-1 thru CNV-6	Harsh	Electrical Mechanical	A A	B,C,D	720 720
Core Exit Thermocouple - ICI Stringer Assembly-01 through 12	CNV-1 thru CNV-6	Harsh	Electrical	A	B,C,D	720
ICI Cable Junction Box, Separation Groups B and C	RXBP-1	Harsh	Electrical	A	B,C,D	720
RADIATION MONITORING SYSTEM (E120)						
Under Bioshield Area Radiation Monitor	RXBP-1	Harsh	Electrical	A	B,C,F	720

Table 3.11-1: List of Environmentally Qualified Equipment Located in Harsh Environments (Continued)

Description[4][5]	Environmental Qualification Zone[1]	Environmental Qualification Environment	Qualification Program	Environmental Qualification Category[3]	PAM Type[2]	Operating Time (Hrs)
REACTOR BUILDING CRANE (F011)						
RXB Crane Power Cabinet	RXBP-2	Harsh	Electrical	B	N/A	720
RXB Crane Control Cabinet	RXBP-2	Harsh	Electrical	B	N/A	720
RXB Crane	RXBP-2	Harsh	Electrical Mechanical	B	N/A	720
REACTOR BUILDING COMPONENTS (F012)						
RBCM Steam Gallery Blow Off Panel	RXBG-8	Harsh	Mechanical	A	N/A	72
RBCM CVC HELB Chase Blow Off Panel	RXBG-8	Harsh	Mechanical	A	N/A	72
RXB Over-Pressurization Vents	RXBG-2	Harsh	Mechanical	A	N/A	72

Notes:

1. Environmental Zone Locations are delineated in Table 3C-1.
2. PAM Type Variables:
   - Type B: those variables that provide information to indicate whether plant credited functions are being accomplished.
   - Type C: those variables to be monitored to provide information to indicate whether the primary reactor containment, the fuel cladding, or the reactor coolant pressure boundary remain intact and do not have a potential to be breached.
   - Type D: those variables that provide information to indicate the operation of individual safety systems and other systems that perform credited functions. These variables are to help the operator make appropriate decisions in using the individual systems performing design functions in mitigating the consequences of an accident.
3. EQ Categories:
   - A Equipment that experience the environmental conditions of design basis accidents for which it must function to mitigate said accidents, and that is qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.
   - B Equipment that experience the environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that is qualified to demonstrate the capability to withstand the accident environment for the time during which it must not fail with safety margin to failure.
4. This listing is based on a single module evaluation and does not consider multi-module interactions because the secondary module(s) effects that are created by the primary module affected are enveloped by their own qualifications.
5. Commodities necessary to support equipment listed in Table 3.11-1 are environmentally qualified for the environmental conditions they are subjected to. Examples of commodity items that are subject to environmental qualification includes, but are not limited to, equipment items such as cables, connectors, electrical splices, conduit seals, thread sealants, terminal blocks, or lubricants.

Table 3.11-2: NRC Guidance and Industry Standards for Environmental Qualification

RG	Endorsed Industry Standard	Comments
1.40		Regulatory Guide 1.40 is not applicable because the NuScale Power Plant US460 standard design does not use environmentally-qualified continuous duty motors.
1.63	IEEE Standard 317- 1983 (Reference 3.11-3)	The portion of the RG 1.63 guidance that endorses IEEE-317-1983 is applicable. Requirements of IEEE 741-1997 (Reference 3.11-4) are used for external circuit protection of electrical penetration assemblies instead of IEEE 741-1986 (Reference 3.11-1) as endorsed by RG 1.63. The 1997 version, including the additional design enhancements, is consistent with RG 1.63.
1.73	IEEE Standard 382-2006 (Reference 3.11-5)	This guidance is applicable except for portions directed towards high-temperature gas-cooled reactor designs.
1.89	IEEE Standard 323-1974 (Reference 3.11-2) and implementing criteria of 10 CFR 50.49	This RG is applicable except for aspects specific to boiling water reactors or related to structures, systems and components not relevant to the NuScale design, and the reference to RG 1.4 for source term, because the source term provisions of RG 1.4 are superseded by RG 1.183 for new reactors. NUREG-0588 Category I guidance is used to enhance the guidance provided by RG 1.89.
1.97 (as supplemented by RG 1.89)	IEEE Standard 497-2016 (Reference 3.11-14) with clarifying positions specified in Section C of RG 1.97.	Section 7.2.13 describes compliance with RG 1.97 and the method used to identify PAM equipment. The NuScale Power Plant US460 standard design does not include Type A PAM variables by design. The NuScale Power Plant US460 standard design satisfies power supply requirements in Section 6.6 of IEEE Standard 497-2016
1.100	ASME QME-1-2017 (Reference 3.11-13)	The NuScale Power Plant US460 standard design complies with Appendix QR-B of ASME QME-1-2017 with the following exceptions: QR-B5200, Identification and Specification of Qualification Requirements, (g) material activation energy. QR-B5300 Selection of Qualification Methods for determination and recording of shelf life of nonmetallics. QR-B5500 Documentation, (h) shelf life preservation requirements. Appendix 3C describes the exceptions cited.
1.152	IEEE Standard 7-4.3.2-2003 (Reference 3.11-6)	No exceptions
1.153	IEEE Standard 603-1991 (Reference 3.11-7)	No exceptions
1.156	IEEE Standard 572-2006 (Reference 3.11-8)	No exceptions
1.158	IEEE Standard 535-2013 (Reference 3.11-9)	Regulatory Guide 1.158 is not applicable because the augmented DC power system batteries are non-Class 1E. However, guidance in IEEE Standard 535-2013 is used as supplemental guidance to IEEE Standard 323-2003 (Reference 3.11-12) to address aging of valve regulated lead acid batteries.
1.180		Section 7.2.2 contains additional details of electromagnetic interference and radio frequency interference qualification.

Table 3.11-2: NRC Guidance and Industry Standards for Environmental Qualification (Continued)

RG	Endorsed Industry Standard	Comments
1.183		NuScale Topical Report TR-0915-17565-P-A (Reference 3.11-10) and Section 12.2.1 describes an alternate methodology for source terms for design basis events.
1.209	Endorses, in part, IEEE Standard 323-2003 (Reference 3.11-12)	No exceptions
1.211	Endorses, in part, IEEE Standard 383-2003 (Reference 3.11-11)	No exceptions

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

3.12.1 Introduction

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

NuScale Power, LLC, utilizes a graded level of detail approach in piping design. This approach is discussed in the Nuclear Regulatory Commission (NRC) white paper - Piping Level of Detail for Design Certification (Reference 3.12-9).

Piping system designs, supported by preliminary analysis, have been completed (e.g., layout, pipe size) for the Class 1, 2, & 3 piping systems within the NuScale Power Module (NPM) and the requirements for the design, analysis, materials, fabrication, inspection, examination, testing, certification, packaging, shipping, and installation of these systems are documented in an ASME design specification. Preliminary stress analyses are performed for the Class 1, 2, & 3 high-energy piping larger than nominal pipe size (NPS) 1 both inside and outside containment in order to confirm the adequacy of the piping layout and support locations, and to support high energy line break evaluations (Section 3.6). Preliminary analyses evaluate applicable loads listed in Section 3.12.5.3 according to the applicable code rules, including detailed fatigue evaluations where applicable.

Piping evaluations are not completed for the containment flooding and drain line, the containment evacuation line, and the reactor component cooling water lines because they are not high-energy, and for the emergency core cooling system (ECCS) lines and other small instrument lines because they are smaller than NPS 1. The lines excluded from preliminary stress evaluations are NPS 2 or smaller, except for the containment evacuation line, which is NPS 4. Because the support and nozzle loads for these lines are small, they can be routed using good engineering judgment at this phase in the design. Detailed evaluations are also not yet performed for the welds between the containment isolation valves and the containment isolation test fixtures for the discharge, injection, pressurizer spray, degasification and feedwater lines outside containment. However, as stated above, the piping systems attached to outboard side of these containment isolation valves have been analyzed.

3.12.2 Codes and Standards

As discussed in the following sections, codes and standards used in the design of piping systems and piping supports support compliance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 1, GDC 2, GDC 4, GDC 14, GDC 15, and 10 CFR 50 Appendix S. The design codes for ASME Class 1, 2, and 3 piping systems are described below.

3.12.2.1 American Society of Mechanical Engineers Boiler and Pressure Vessel Code

The design code specified for ASME Code Section III Class 1, 2, and 3 piping in the design is in Reference 3.12-1. The conditions of use for ASME Boiler and Pressure Vessel Code (BPVC) Section III are applied in accordance with 10 CFR 50.55a (b)(1) as applicable to the 2017 Edition.

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

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

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

Quality Group D (Regulatory Guide [RG] 1.26) piping is designed and analyzed to the 2018 edition of ASME B31.1 (Reference 3.12-2).

3.12.2.2 American Society of Mechanical Engineers Code Cases

American Society of Mechanical Engineers Code Cases may be used if they are either conditionally or unconditionally approved in RG 1.84.

3.12.2.3 Design Specification

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

3.12.3 Piping Analysis Methods

3.12.3.1 Experimental Stress Analyses Methods

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

3.12.3.2 Modal Response Spectrum Method

The effects of the ground motion during a safe shutdown earthquake (SSE) event are transmitted through structures and components to the piping systems at support and anchorage locations. Seismic Category I piping systems are required to be designed to withstand the effects of the SSE and maintain the capability of performing their safety functions. The response spectrum method is a dynamic method of analysis used for piping systems. This analysis method applies in-structure response spectra (which are amplified from the fundamental seismic ground motion spectra) to the piping system in all three directions.

The in-structure response spectra are applied to the locations where the piping system is attached to or supported by a structure, such as Reactor Building (RXB) piping supports or NPM vessel nozzles. The response spectrum analysis is performed using either the uniform support motion (USM) method or the independent support motion (ISM) method.

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

3.12.3.2.1 Development of In-structure Response Spectra

To perform the response spectrum analysis, an in-structure response spectra must be developed for the structures that support the piping system.

In-structure response spectra of the NPM are determined using dynamic analysis of a three-dimensional, finite element model of the NPM structural system as described in US460 NuScale Power Module Seismic Analysis Technical Report TR-121515 (Reference 3.12-10). For piping that is attached to the building, the in-structure response spectra of the RXB is used, which is described in Section 3.7.

The in-structure response spectra include accelerations for three orthogonal directions (two horizontal and one vertical) from the time history motions of the supporting structure. Uncertainties in the structural frequencies, which represent uncertainty or approximations of material and structural properties, are accounted for by peak broadening ±15 percent.

3.12.3.2.2 Uniform Support Motion

For piping systems that may be supported at multiple points within a structure the seismic motions of each support location may vary. An acceptable approach for analyzing these piping systems is to define a uniform response spectrum (URS) that envelops the individual response spectra at the various

support locations. This method is referred to as the uniform support motion method. The methods and guidance of RG 1.92 is used for combining modal and spatial responses for USM method of analysis. Either Revision 3 or Revision 1 of RG 1.92 may be used for the design. If the software used for analysis does not have the capability to comply entirely with Revision 3 of RG 1.92, conformance to Revision 1 is permitted. Revision 3 of RG 1.92 states:

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

3.12.3.2.3 Modal Combination

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

When performing response spectra analyses that comply with Revision 1 of RG 1.92, modal responses of the piping system are only considered below a defined cutoff frequency at which spectral accelerations approximately return to the zero period acceleration (ZPA). Above the ZPA frequency the system is considered to be rigid because the components are not significantly excited by the seismic ground or in-structure motion. However, structures, systems, and components may have important natural vibration modes at frequencies higher than the ZPA frequency because of more rigidly restrained components or significant lumped masses near rigid restraints, which are not considered in the low frequency modal analysis. Therefore, the contribution of mass associated with modes higher than the ZPA are accounted for as described in Section 3.12.3.2.6.

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

3.12.3.2.4 Periodic Modal Responses

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

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

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

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

For a system that has closely spaced modes, the double sum methods are used to combine the periodic modal responses. Analyses based on RG 1.92, Revision 1 use the Absolute Double Sum method for closely spaced modes (Equation (8) of RG 1.92 Revision 1). Analyses based on RG 1.92, Revision 3 use the Signed Double Sum method (Equation (1) of RG 1.92). These double sum equations include modal correlation coefficients that are uniquely defined, depending on the method chosen for evaluating the correlation coefficient.

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

3.12.3.2.5 Modes with Both Periodic and Rigid Response Components

In the transition region where modal responses consist of both periodic and rigid components, the response components can be separated by the methods described in C.1.3.1 or C.1.3.2 of RG 1.92, Revision 3.

3.12.3.2.6 Residual Rigid Response

The contribution of the "missing mass" of piping systems above the ZPA is accounted for using the method provided in Section C.1.4 of RG 1.92, Revision 3. This method assumes that modes above the cutoff frequency (equivalent to the ZPA frequency for seismic loads) respond as rigid modes excited at the ZPA.

For the missing mass method, the modal responses are determined for those modes with natural frequencies less than the ZPA. For each

degree-of-freedom included in the dynamic analysis, the fractions of degree-of-freedom mass included and not included in the summation of the modes are determined. Modes higher than the ZPA are assumed to respond in phase with the ZPA and with each other; therefore, they are combined algebraically and applied to the degree-of-freedom masses not included in the low frequency modal analysis (below the ZPA).

An alternative approach to including the contribution of high-frequency modes is to use the Static ZPA method provided in RG 1.92, Revision 3. If Static ZPA is used, then periodic and rigid components are combined with Combination Method B from Regulatory Position C.1.5.2 of RG 1.92, Revision 3.

3.12.3.2.7 Complete Inertial Response

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

3.12.3.2.8 Directional Combination

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

3.12.3.2.9 Seismic Anchor Motion

In addition to dynamic inertia loads imparted to the piping system, the effects of piping seismic anchor motion (SAM) displacements are also considered as a static load. The maximum relative support displacements are obtained from the structural response analysis of the NPM and RXB.

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

Analyses of piping systems due to SAMs are performed statically. The system response due to inertial effects and due to anchor motions are combined by the absolute sum method for USM analysis, when combining of the results is necessary.

3.12.3.2.10 Independent Support Motion Method

The USM method can result in considerable overestimation of seismic responses. Therefore, an alternate method, which is discussed in Section 3.7.3, is the ISM method. The ISM method is generally used for piping systems supported by more than one structure, but may be used for piping systems with multiple supports located in a single structure. This method of analysis is performed by grouping piping supports (such as supports attached to the same portion of a structure) and applying a single response spectrum to each group. One group of supports is moved at a time using the input response spectrum specified for those supports, with the other groups being stationary.

As discussed in Section 3.7.3, when performing the ISM method of analysis, calculated responses are combined using the methods and guidance of Section 2.4 of NUREG-1061, Volume 4. For each mode and direction, responses from the individual grouped analyses are combined by absolute summation. Then, spatial (directional) and modal component responses of the piping system are combined as described in Section 3.12.3.2. Per NUREG-1061, Volume 4, consideration of closely spaced modes need not be taken into account; therefore, spatial and modal results are combined using the SRSS method described in Section 3.12.3.2.4. Because closely-spaced modes are not considered, the sequence is not important and modes may be combined first or spatial components may be combined first.

Responses for modes below the ZPA are treated as periodic responses, while the residual rigid response is calculated using the missing mass method as described in Section 3.12.3.2.6. The low frequency response (periodic response) and the high frequency response (residual rigid response) are combined using the SRSS method.

If the responses due to anchor motions are combined with inertial effects, the SRSS method is used; however, the response due to anchor motions may be evaluated separately from inertial effects per ASME Section III (for example, NB-3656(a)(3) and (b)(4)). Damping values from RG 1.61 Revision 1 are used when performing analysis using the ISM method. Section 3.12.3.4 contains discussion of appropriate damping.

3.12.3.3 Time-History Method

Seismic analysis of piping systems may also be performed using the time history method (as opposed to the response spectrum method). The time history method can provide more realistic results for multiply-supported systems but it requires increased analytical effort. Therefore, the time history method of analysis for seismic input is generally reserved for major components. Time history analysis can be performed by direct integration of the coupled equations of motion or by modal superposition.

When direct integration time history analysis is used to analyze seismic loads on piping systems the analysis is executed according to the methodology described in Reference 3.12-10.

The modal superposition method is performed by decoupling the multiple degree-of-freedom equations of motion by changing the equations of motion from normal (displacement) coordinates to modal coordinates. The equations are solved linearly as single degree-of-freedom equations and then the results for all modes are superimposed (i.e., algebraically summed) at each time step for each location.

When modal superposition time history analysis is used to analyze seismic loads on piping systems, the methods of RG 1.92 Revision 3 are used to calculate the "missing mass" and combine it with the periodic response, and when combining spatial results. Damping values used for these analyses are per Section 3.12.3.2.2. Additionally, time step sensitivity evaluations are performed to show that the selected time step provides acceptable convergence.

Analysis of piping system response or component response due to other transient loads such as water hammer, steam hammer, and impingement may also be performed using the time history method (Section 3.12.5.3).

3.12.3.4 Damping Values

When performing analysis of piping systems by the USM or ISM methods, damping values are applied as permitted by RG 1.61. For analysis of piping systems, a constant damping value of 4 percent is applied for analysis of SSE loads (for all frequencies) as permitted by RG 1.61. Frequency dependent damping is not used, though it is conditionally permitted by RG 1.61. If the analysis of a piping model includes other nonpiping components (such as supports or structural elements) that have different damping values per RG 1.61, then composite modal damping values are determined using Equation (1) or Equation (2) from SRP 3.7.2, Acceptance Criterion 13. Note that when composite modal damping is determined using these methods, the damping does not exceed 20 percent of critical. Equations for determining system composite modal damping are provided in American Society of Civil Engineers (ASCE) 4-2016 (Reference 3.12-11).

3.12.3.5 Inelastic Analysis Method

Inelastic analysis methods are not used to qualify piping for the NuScale plant design.

3.12.3.6 Equivalent Static Load Method

The equivalent static load method of seismic analysis is not used to qualify piping for the NuScale plant design.

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

Non-seismic piping that is located in proximity to the seismic Category I piping, and whose failure could impair the safety function of the Seismic Category I piping, is classified as seismic Category II and is analyzed and qualified to the same seismic criteria as the seismic Category I piping, thereby precluding adverse interaction during the SSE.

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

3.12.3.8 Seismic Category I Buried Piping

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

3.12.4 Piping Modeling Technique

3.12.4.1 Computer Codes

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

ANSYS

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

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

AutoPIPE

The computer program AutoPIPE is used for the design and analysis of NuScale piping systems. AutoPIPE is used for analysis of piping due to static loads and for dynamic loads. AutoPIPE also performs design checks for ASME Code Class 1, 2, and 3 and ASME B31.1 piping. The dynamic analyses required for seismic evaluations such as response spectrum analysis and time history analysis are performed using AutoPIPE.

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

3.12.4.2 Dynamic Piping Model

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

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

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

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

The mass of strut supports attached to the piping are considered as follows, regardless of orientation. The full weight of the pipe clamp plus half the estimated weight of the strut is applied at the pipe attachment location. The other half of the strut weight is considered as supported by the anchoring structure.

The mass of attached supports (e.g., lugs) is included if the weight exceeds 10 percent of the weight of the adjacent pipe spans. The subject span is defined as the piping components up to the next support location in both directions from the lug considered for inclusion in the model.

3.12.4.3 Piping Benchmark Program

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

COL Item 3.12-1: An applicant that references the NuScale Power Plant US460 standard design may use a piping analysis program other than the programs listed in Section 3.12.4; however, the applicant will implement a benchmark program using the models for the NuScale Power Plant US460 standard design.

3.12.4.4 Decoupling Criteria

Decoupling Criteria

Piping models are generally terminated at structural anchors, which effectively isolate the system from additional static and dynamic effects beyond the anchor. These structural anchors are typically vessel nozzles, but may also be pipe supports that restrain all six degrees of freedom. Branch lines for which the routing is unknown can be decoupled from the analysis of the main ASME Class 1, 2, or 3 piping run using the following criteria.

Decoupling methodology may only be applied at locations where the structural interaction between adjacent segments of piping is limited and can be sufficiently accounted for using standard methods. Therefore, this approach may only be used at locations where there is a significant change in pipe size, such as branch lines of larger piping. Branch lines (such as instrument lines) smaller than the main run of the analyzed piping may be excluded from the analysis if the moment of inertia of the branch line is less than or equal to one twenty-fifth that of the run pipe moment of inertia (Reference 3.12-14). Decoupling of branch lines is not used in the following situations:

- An anchor or restraint on the branch pipe is located near the run pipe and significantly restrains the movement of the run pipe.
- The branch is located near a sensitive connection (e.g., equipment nozzle) and more precise magnitudes of reactions are required to determine the acceptability of the loads at the connection (Reference 3.12-14).
- There is a large mass (e.g., large valve or fitting) on the branch line in the span between the connection to the run pipe and the nearest support.

These criteria ensure that the effects of the smaller decoupled branch line on the larger run piping can generally be considered negligible. However, stress intensification factors and stress indices associated with the connection of the smaller line are considered in the analysis of the larger piping. The additional mass of the smaller line is considered for inclusion in the model of the larger piping to account for the decoupled line. When included, the added mass is at least half of the mass of the portion of the decoupled line up to the nearest support.

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

Overlap Region Methodology

It is preferred to model an entire piping system with relevant connections and supports included in the same analysis. If it is not feasible to analyze a piping system as a single model then the structural overlap methodology provided in NUREG/CR-1980 may be used.

When the structural overlap methodology is applied, the conditions and criteria in Section 2 of NUREG/CR-1980 are satisfied. It is required that there are at least four rigid restraints in each of three mutually perpendicular directions in the overlap region (including the ends) when the method is applied. For axial restraints only, this requirement may be relaxed to a single restraint in any straight segment. Additionally, piping system analyses that include the overlap region are required to show acceptable results for the piping components and supports in the overlap region.

3.12.5 Piping Stress Analysis Criteria

3.12.5.1 Seismic Input Envelope Versus Site-Specific Spectra

The standard plant piping is evaluated using the certified seismic design response spectra and the certified seismic design response spectra-high frequency described in Section 3.7.1.

Section 3.7.2 describes the floor response spectra in detail.

COL Item 3.12-2: An applicant that references the NuScale Power Plant US460 standard design will confirm that the site-specific seismic response is within the parameters specified in Section 3.7. An applicant may perform a site-specific piping stress analysis in accordance with the methodologies described in this section, as appropriate.

3.12.5.2 Design Transients

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

3.12.5.3 Loadings and Load Combinations

Pressure

The design differential pressure between the inside and outside of the piping pressure boundary components (P_{des}) is used for the analysis of ASME Code Class 1, 2, and 3 piping, and the ASME B31.1 piping. The minimum required piping wall thickness for ASME Code Class 1, 2, and 3 piping is calculated using NB-3640, NC-3640, and ND-3640, as applicable, at the design pressure using material properties from ASME Section II (Reference 3.12-13) Part D at the applicable design temperature. The design pressure of piping systems includes allowances for pressure addition sources (such as pumps), pressure surges, control system error, and system configuration effects such as static pressure heads. Design pressures (P_{des}) and service pressures (P & P_{max}) are used in load combinations as noted in Table 3.12-1 and Table 3.12-2 for calculating stresses considering the condition and service level.

Deadweight

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

Thermal Expansion

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

The anchors of piping systems may also be subject to thermal expansion, such as thermal anchor motions of equipment nozzles (such as those of the reactor pressure vessel and containment vessel), support/restraints, and run piping for decoupled branch lines. Thermal anchor motions less than or equal to one-sixteenth inch are excluded from consideration. For decoupled branch lines, thermal anchor motions are obtained from the applicable analysis of the run pipe.

The reference temperature for thermal analysis of piping systems is taken as 70 degrees Fahrenheit. At this reference temperature, loads due to thermal expansion of piping are zero. For ASME Code Class 2 and 3 piping systems with an operating temperature of 150 degrees Fahrenheit or less, thermal analysis is not required except when required due to interface with other components (Section 3.12.5.11).

Buoyancy

Buoyancy loads (B) are applicable to submerged piping. Buoyancy is calculated based on the weight of the water displaced.

Seismic

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

The operating basis earthquake (OBE) is defined as one-third of the SSE. Due to the selection of the OBE as one-third of the SSE, the OBE effects are not included as design loads (as allowed by 10 CFR 50 Appendix S), but the OBE cyclic effects are included in fatigue evaluations of ASME Code Class 1 piping.

Relief Valve Thrust

Reaction loads are imparted onto piping system components when relief valves are actuated open. These loads are considered for actuation of reactor safety valves (RSVs) and of ECCS valves.

Guidance for the design and analysis of safety valve installations is provided in ASME Section III (Reference 3.12-1) Nonmandatory Appendix O. The analysis of these loads is discussed further in Section 3.12.5.8.

Water and Steam Hammer

Pressure waves are created when the flow of fluid in a piping system is abruptly altered. This alteration can be initiated by mechanisms such as rapid valve actuation, pumps starting, or the collapsing of steam voids. If water or steam hammer loads are credible and significant for a piping system or portion of piping, they are included in the analysis.

Wind, Hurricane, Tornado Loads

The ASME Code Class 1, 2, and 3 piping in the design is not routed in areas exposed to wind, hurricane, or tornado loads.

Design Basis Pipe Break Loads

The loads due to design basis pipe breaks (DBPBs) are included in the analysis of ASME Class 1, 2, and 3 piping for the appropriate service conditions. Loads are imparted onto piping system components in the form of pipe whip, jet impingement, elevated temperatures, and hydraulic dynamic effects. External dynamic effects of pipe breaks are discussed further in Section 3.6.

Thermal and Pressure Transient Loads

Thermal and pressure transient loads are included for the analysis of ASME Code Class 1 piping. For ASME Code Class 1 piping, these transient loads are included

as Service Level A and B loads and their effects are determined by calculating the primary plus secondary stress intensity ranges, as the piping system goes from one load set (such as pressure, temperature, moment, and force loading) to another load set that follows it in time. Section 3.12.5.7 addresses the operating experience from NRC Bulletin 88-08 for Class 1 piping.

For ASME Code Class 2 and 3 piping, transient loads are also considered in the analyses by using the bounding pressure and temperature ranges in individual load combination cases. Loads created by thermal stratification, cycling, or striping that may occur in unisolable piping are accounted for in the design and analysis of the ASME Class 2 and 3 piping per the operating experience from NRC Bulletin 88-08.

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

Hydrotests

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

Load Combinations

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

3.12.5.4 Fatigue Evaluation of American Society of Mechanical Engineers Code Class 1 Piping

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

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

During the life of the plant, at least one SSE and five OBEs, with 10 maximum stress cycles per event, are assumed. The fatigue analysis for Class 1 piping may utilize either of two approaches.

Two SSE events, with 10 maximum stress cycles each for a total of 20 full cycles, are evaluated. This approach is considered equivalent to the cyclic basis of one SSE and five OBEs.

Alternatively, the number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Annex D of Institute of Electrical and Electronics Engineers Standard 344-1987 (Reference 3.12-4). When this method is used, and if the amplitude of the vibration is taken as one-third of the amplitude of the SSE, then 312 fractional amplitude SSE cycles are considered.

3.12.5.5 Fatigue Evaluation of American Society of Mechanical Engineers Code Class 2 and 3 Piping

Design and analysis of ASME Code Class 2 and 3 piping systems and piping components addresses fatigue effects by applying stress range reduction factors as provided in NC/ND-3611.2(e) to the allowable stress range for thermal expansion stresses.

3.12.5.6 Thermal Oscillations in Piping Connected to the Reactor Coolant System

The piping sections that can not be isolated and are connected to the reactor coolant system (RCS) can experience temperature stratification and oscillation due to mixing with stagnant lower temperature fluid with the higher temperature fluid at the connection interface (due to turbulent penetrating flow or leakage past an isolation component). These thermal conditions add fatigue loads to piping components because of constrained thermal deflections that must be either accounted for by analysis or precluded by design. Thermal oscillations in RCS connected piping are determined to be the cause of pressure boundary component failures at multiple operating nuclear plants as described in NRC Bulletin 88-08 including supplements. Therefore, unisolable sections of piping connected to the RCS of the NuScale Power Plant US460 standard design are evaluated for susceptibility to temperature oscillations that may affect the integrity of the components.

The methodology of Electric Power and Research Institute (EPRI) technical report TR-103581 (Reference 3.12-6) is used to assess unisolable piping connected to the RCS for thermal oscillations and stratification in the NuScale Power Plant US460 standard design. Since the issuance of EPRI TR-103581, EPRI has continued to update its guidance for the assessment of these phenomena (MRP-146 and subsequent revisions). These updates have led to changes in the thermal oscillation and stratification screening criteria from what was documented in EPRI TR-103581. Although this recent guidance is proprietary to EPRI, publicly available information (Reference 3.12-3, Reference 3.12-5, and Reference 3.12-15) are used to determine screening criteria for the NuScale Power Plant US460 standard design, to ensure that the assessment of whether or not a line is susceptible to thermal stratification or cycling is consistent with current industry practice.

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

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

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

As noted previously, screening criteria are used to determine if piping systems are susceptible to these phenomena. Lines that meet one or more of the below criteria are not susceptible to thermal stratification or cycling:

- The branch line is not stagnant during normal plant operation (Reference 3.12-3).
- Pipe size is NPS 1 or smaller (Reference 3.12-5, and Reference 3.12-15).
- The branch line is an up-horizontal or horizontal pipe with no potential for inleakage (Reference 3.12-3).

The following piping systems connected to the RCS in the NuScale Power Plant US460 standard design are evaluated for thermal stratification and cycling:

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

The screening evaluation is as follows. The RCS discharge line, RCS injection line, and pressurizer spray lines are not stagnant during power operations, therefore these lines are not susceptible. The reactor pressure vessel high point degasification line is a vapor-filled, up-horizontal line with no potential for inleakage, therefore this line is also not susceptible. The ECCS lines are normally stagnant and have horizontal portions but they are smaller than NPS 1 and therefore are not susceptible. The evaluated lines satisfy the screening criteria, and therefore do not require further evaluation.

3.12.5.7 Thermal Stratification

Thermal Stratification is discussed in Section 3.12.5.7.1 through Section 3.12.5.7.3.

3.12.5.7.1 Pressurizer Surge Line Stratification

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

3.12.5.7.2 Spray Line Stratification

The portions of the spray lines that are Class 1 are primarily in a vertical orientation, which reduces the susceptibility to thermal stratification. Additionally, a small, constant flow of spray bypass normally precludes stagnant fluid in these lines.

3.12.5.7.3 Feedwater Line Stratification

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

The NuScale Power Plant US460 standard design feedwater lines are designed to minimize adverse loading due to thermal stratification. The SG feedwater nozzles (located on the feedwater inlet plenums) and the adjacent feedwater lines are either vertical or angled downward from the horizontal to minimize thermal stratification load.

3.12.5.8 Safety Relief Valve Design, Installation, and Testing

The design of safety valves and relief valves for the overpressure protection of ASME Class 1, 2, and 3 components considers the recommendations of the ASME Code (Reference 3.12-1) Nonmandatory Appendix O. Appendix O of the ASME Code includes valve arrangement considerations as well as guidance for determining loads required to be included in the analysis as a result of valve

actuation. Appendix O categorizes pressure relief device installations in two configurations; closed discharge systems and open discharge systems. Closed discharge systems are relief devices that discharge into a distant location through a pipe connected directly to the relief valve, and open discharge systems are relief devices that discharge to atmospheric conditions.

ASME Section III relief valves utilized in the NuScale US460 standard design discharge into containment without piping and therefore are considered to be an open discharge system configuration. Open discharge systems are analyzed with applicable reaction forces including the effects of the suddenly applied load. This analysis is achieved by static methods using a dynamic load factor or by modeling the system and performing a dynamic analysis.

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

- Load combinations include the most severe combination of the applicable loads because of internal fluid weight, momentum and pressure, dead weight of valves and piping, thermal load under heatup, steady state and transient valve operation, reaction forces when valves are discharging, and seismic forces.
- The contribution from reaction forces and moments are included by use of static analysis with a dynamic load factor or by using the maximum instantaneous values of forces and moments for each location as determined by dynamic system analysis. A dynamic load factor of 2.0 is used or the guidance provided in ASME B31.1 (Reference 3.12-2) Nonmandatory Appendix II is used to calculate an appropriate dynamic load factor.
- Where more than one relief valve or safety valve is installed to protect the same pressure boundary, the sequence of valve openings that induce the maximum instantaneous value of stress at each location is used for loading at that location.
- Stresses are evaluated and applicable stress limits satisfied for connecting systems for which safety/relief valves are installed.

Load combinations and stress criteria are provided in Table 3.12-1 for ASME Code Class 1 and Table 3.12-2 for Class 2, and 3 piping. Loads from ASME Code Class 1, 2, and 3 pressure relief devices (such as the RSVs and the ECCS vent valves) are considered, although they are not mounted on piping systems, because the discharge fluid interacts with other piping inside containment.

3.12.5.9 Functional Capability

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

maintained. NUREG-1367 concludes that piping system functional capability is maintained for Service Level C and D loading conditions provided that

- dynamic loads are reversing. Includes loads due to earthquakes, building-filtered loads such as those due to vibration of buildings caused by relief-valve actuation in boiling-water reactors, and pressure wave loads (not slug-flow fluid hammer).
- dynamic moments are calculated using an elastic response spectrum analysis with +/-15 percent peak broadening and with not more than 5 percent damping.
- steady-state (e.g., weight) stresses do not exceed 0.25 *Sy*.
- D_o/t does not exceed 50.
- external pressure does not exceed internal pressure.

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

These requirements are invoked for Service Level C and D plant events for ASME Class 1, 2, and 3 piping that is required to transfer fluid during those events. Additional clarifications to the above criteria are that criterion 2 only applies to seismic loads, and criterion 5 need not be satisfied as long as the external differential pressure during Service Level C and D events are qualified to ASME Section III stress limits.

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

3.12.5.10 Welded Attachments

Welded attachments for ASME Class 1, 2, and 3 piping are permitted provided the effects of the attachment on the piping are considered in accordance with ASME Code, Section III Nonmandatory Appendix Y.

3.12.5.11 Minimum Temperature for Thermal Analyses

No thermal analysis is required for ASME Code Class 2 and 3 piping systems with an operating temperature equal to or less than 150 degrees Fahrenheit, unless the Class 2 or 3 piping system is connected to a Class 1 component. A thermal analysis is performed for Class 2 or 3 piping systems connected to a Class 1 component so that the effects of piping expansion can be included in the analysis of the Class 1 component. However, qualification of the stresses in the Class 2 or 3 piping is not required by Section III.

3.12.5.12 Intersystem Loss-of-Coolant Accident

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

3.12.5.13 Effects of Environment on Fatigue Design

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

3.12.6 Piping Support Design Criteria

3.12.6.1 Applicable Codes

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

Subsection NF of the ASME Code categorizes piping supports into three types, and specific requirements are provided for each type of support. The three types of supports are described as plate and shell type, linear type, and standard supports. Plate and shell type supports are fabricated from plate and shell elements (such as a skirt or saddle) and are normally subject to a biaxial stress field (NF-1212). A linear type support is defined as acting under essentially a single component of direct stress, but may also be subject to shear stresses. Examples of linear type supports are tension/compression struts, beams subject to bending, trusses, frames, rings, arches, and cables (NF-1213). Standard supports are typified by the supports described in MSS SP-58 (Reference 3.12-12), which consist of standard catalog parts (Figure NF-1214-1). Standard support capacities may be determined by load rating procedures (e.g., NF-3280), plate and shell analysis, or by linear analysis.

3.12.6.2 Jurisdictional Boundaries

There are two jurisdictional boundaries of piping supports, these are the boundary between the pipe support and the supported or restrained piping, and the boundary between the support and the anchor structure or component. In accordance with NF-1131 of the ASME Code, the jurisdictional boundaries between ASME Class 1, 2, and 3 supports and other components, including piping systems, meet the requirements of NB/NC/ND/NE-1132 as applicable to the class of the component.

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

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

The boundary for piping supports that are attached to building steel is at the interface with the building steel and the weld conforms to the requirements of Subsection NF of the ASME Code.

3.12.6.3 Loads and Load Combinations

Table 3.12-3 lists the required load combinations for ASME Code Class 1, 2, and 3 supports.

3.12.6.4 Pipe Support Base Plate and Anchor Bolt Design

Class 1, 2, and 3 pipe supports are supported by the NPM or attached directly to building steel; therefore, base plates and anchor bolts are not used.

3.12.6.5 Use of Energy Absorbers and Limit Stops

Energy absorbers or limit stops are not used for ASME Code Class 1, 2, or 3 piping.

3.12.6.6 Use of Snubbers

Snubbers are not used for ASME Code Class 1, 2, or 3 piping.

3.12.6.7 Pipe Support Stiffness

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

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

For ASME Code Class 1, 2, or 3 supports modeled as rigid in the piping system analysis, two checks for deflection are performed. One check compares the deflection in the restrained direction(s) to a maximum of one-sixteenth inch for SSE loads or the minimum support design loads. Another check compares the deflection in the restrained direction(s) to a maximum of one-eighth inch for the

worst case deflection for the specified load combinations. When evaluating pipe support deflections, dynamic flexible elements of the attaching components or component supports are also considered.

3.12.6.8 Seismic Self-Weight Excitation

Seismic self-weight excitation is considered in the design of pipe supports. Unless determined to be insignificant, the inertial response of the support mass is evaluated and included as a seismic load in the support analysis. Dynamic analysis methods (such as the response spectrum method) similar to that used for the pipe system analysis, are used for complex support structures. When using the URS method, the seismic response of piping supports due to excitation of the pipe support mass, the seismic piping inertial response, and the loads from SAMs are combined by absolute sum. Alternatively, an equivalent-static analysis procedure may be used to determine pipe support responses due to self-weight excitation if the following criteria are met:

- The support can be realistically represented by a simple model. A simple uniform cantilever beam or a uniform cantilever beam with a concentrated mass qualify as simple models.
- The method produces conservative results in responses.
- The design and analysis account for the relative motion between all points of support.
- If the fundamental frequency of the pipe support occurs at a frequency in the ZPA range, a factor of 1.5 is applied to the ZPA of the applicable in-structure response spectra. A factor of less than 1.5 may be used with adequate justification.
- If the fundamental frequency of the pipe support occurs at a frequency outside of the rigid range (i.e., ZPA) a factor of 1.5 is applied to the peak acceleration of the applicable in-structure response spectra.

RG 1.61 provides damping values for welded steel and bolted steel connections.

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

3.12.6.9 Design of Supplementary Steel

As discussed in Section 3.12.6.1, ASME Class 1, 2, and 3 pipe supports in the design are designed to Subsection NF of the ASME Code. Pipe support members used in connecting ASME Class 1, 2, and 3 piping to the supporting vessels, component supports, or RXB structures are within the jurisdictional boundaries of

Subsection NF of the ASME Code (Section 3.12.6.2). Supplemental structural steel is not used in pipe support members.

3.12.6.10 Consideration of Friction Forces

Frictional forces are considered for the design of pipe supports for applicable loading conditions. Consideration of frictional forces is limited to loading from deadweight/buoyancy loads, thermal expansion loads, anchor or support movement (due to temperature or pressure), and other applicable signed loads, such as those from relief/safety valve discharge to an open system. Frictional forces are not considered for dynamic cyclic loads, such as those from an earthquake or reflected waves due to flow transients.

Frictional forces are not calculated in the piping analysis; rather they are manually determined when performing stress analysis of the pipe supports. The magnitude of the frictional force is the applied pipe force normal to the support surface multiplied by the appropriate coefficient of friction. Alternatively, the friction force may be calculated as the product KX, where K is the stiffness of the support in the direction of pipe movement X. Frictional forces act in the direction of pipe movement (i.e., the unrestrained direction). If pipe movement due to operating and service conditions reverse, the frictional force is considered in both the positive and negative directions.

A minimum coefficient of friction value of 0.30 is used for steel to steel interfaces. A lower coefficient of friction may be justified if low friction slide/bearing plates are used, with the minimum value being 0.10.

3.12.6.11 Pipe Support Gaps and Clearances

For rigid guide pipe supports, a maximum nominal cold condition gap of one-sixteenth inch is included radially. Deadweight supports are specified to be in contact with the piping in the direction of gravity, with a maximum gap of one-eighth inch above the pipe when providing vertical restraint. These gaps allow unrestrained radial thermal expansion of piping and unrestrained rotation. Pipe support gaps in the unrestrained direction(s) are specified large enough to accommodate the maximum deflection of the piping systems at the support.

The specified cold condition gap is checked against the maximum combined radial growth of the pipe due to internal pressure and thermal expansion to ensure adequate clearance exists. Eq. 3.12-1 is used to calculate the pipe radius after expansion. Because the geometrical changes due to internal pressure and thermal expansion are small, it is acceptable to superimpose the calculated radial strains and apply them to the pipe radius in the cold condition. The radial strain terms in Eq. 3.12-1 are equivalent to circumferential strains.

$$r_2 = r_1[1 + (\varepsilon_{r,p} + \varepsilon_{r,th})] \qquad \text{Eq. 3.12-1}$$

where,

r_2 = Pipe radius after expansion (in.),

r_1 = Pipe radius in the cold condition (in.),

$\varepsilon_{r,\,p}$ = Radial strain due to internal pressure (in./in.), and

$\varepsilon_{r,\,th}$ = Radial strain due to thermal expansion (in./in.).

The design does not use specialized stiff pipe clamps in ASME Class 1, 2, or 3 piping NPS 1 or larger that would induce high local stresses on the pipe, as discussed in NRC Information Notice 83-80. Where rigid clamp supports with no gap or with an interference fit are used in ASME Class 1, 2, or 3 piping smaller than NPS 1 (or tubing smaller than 1 inch), the local effect on piping (or tubing) stresses is considered.

3.12.6.12 Instrumentation Line Support Criteria

Supports of ASME Code Class 1, 2, and 3 instrument tubing are classified to the same Class 1, 2, or 3 classification as the tubing they support. These supports are designed, manufactured, tested, and installed to the requirements of ASME Code, Section III, Subsection NF.

3.12.6.13 Pipe Deflection Limits

Where standard piping supports or standard piping support parts are used, the manufacturer's recommended deflection limits are followed. In the NuScale Power design, spring supports are not used for ASME Code Class 1, 2, and 3 piping.

Where rods or strut supports are used in the design, a minus tolerance of 1 degree is applied to the manufacturer given swing angle limit, reducing the allowable swing angle. Correspondingly, the installation tolerances of these types of supports is 1 degree.

3.12.7 References

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3.12-14 Welding Research Council, Inc., “Technical Position on Damping and on Industry Practice,” WRC Bulletin 300, December 1984, Shaker Heights, OH.

3.12-15 Electric Power Research Institute, “MRP Thermal Fatigue Program Update,” presentation at the Materials R&D Tech Exchange Meeting, May 23, 2017, ADAMS Accession No. ML17142A005.

Table 3.12-1: Required Load Combinations for Class 1 Piping

Plant Event	Service Level	Load Combination[1][12]	Allowable Limit
Design	-	P_{des} + DW + B + RF + DFL[2]	Design
Normal Operating Transients	A	P + DW + B + TH + DFL	Level A
Refueling		P + DW + B + RF	
Maximum Service Pressure	B	P_{max}	Level B
Continued Operating Transients		P + DW + TH + DFL	
		P + DW + TH + OBE[3]	
Reactor Trip Transients/Loss of Power		P + DW + B + TH + DFL[5] + ECCS[5]	
Cold Overpressure Protection		P + DW + B + TH + ECCS	
Maximum Service Pressure	C[4]	P_{max}	Level C
Design Basis Pipe Break		P + DW + B + DBPB[5] + DFL[5] + ECCS[5]	Level C[7]
Spurious ECCS Valve Actuation		P + DW + B + ECCS[5] + DFL[5]	Level C
Spurious RSV Actuation		P + DW + B + RSV[5] + DFL[5] + ECCS[5]	
SG Tube Rupture		P[8] + DW + DFL	
Maximum Service Pressure	D[9]	P_{max}	Level D
Main steam pipe break (MSPB) and feedwater pipe break (FWPB)		P + DW + MSPB/FWPB[5] + DFL[5]	
Pipe Breaks + SSE		P + DW + B + DFL[5] + SRSS(SSE + MSPB/FWPB/ECCS)[6]	
		P + DW + B + DFL[5] +SRSS(SSE + DBPB)[6] + ECCS[5]	
		P + DW + B + DFL[5] + SRSS(SSE + RSV)[6] + ECCS[5]	
SAM		SAM[10]	
Rod Ejection Accident		P[8] + DW	Level C[11]

Table 3.12-1: Required Load Combinations for Class 1 Piping (Continued)

Plant Event	Service Level	Load Combination(1)(12)	Allowable Limit
Pressure Test	Test	P + DW	NB-3657 Mandatory Appendix XIII, XIII-3600

Notes:

(1) When the method of analysis does not retain the sign of cyclic dynamic loads (e.g., reflected pressure waves or seismic), they are combined with other loads in the most conservative combination.

(2) The DFL for Service Level A are considered for Design Condition.

(3) The OBE loading is only applicable to fatigue analysis and is not required for Level B stress intensity and stress intensity range evaluations.

(4) If the total number of postulated occurrences for events where Service Level C limits are specified result in more than 25 stress cycles having an alternating stress intensity (S_{alt}) greater than the S_a value at 10^6 cycles determined from the applicable fatigue design curves given in ASME BPVC Section III Mandatory Appendix I, those events causing cycles in excess of 25 are analyzed to Level B Service Limits (NB-3113(b)).

(5) Applicable dynamic loads (DFL/DBPB/MSPB/FWPB/RSV/ECCS) are combined by absolute sum unless sufficient data are available for each load to understand the time history behavior of each compared to the others (i.e., when the time-phase relationships are known), in which case it is acceptable to use a less conservative approach that complies with NUREG-0484, Revision 1.

(6) The parenthesis indicate that SSE and DBPB/MSPB/FWPB/RSV/ECCS are combined using the square root sum of the squares method and do not indicate order of operations. DFLs resulting from pipe break or valve actuation events may be combined with DBPB/MSPB/FWPB/RSV/ECCS (either by absolute sum or SRSS depending on an evaluation consistent with Note 5) before combining with SSE by SRSS. A similar approach may be used for DFLs resulting from an SSE before combining with DBPB/MSPB/FWPB/RSV/ECCS.

(7) For a pipe break beyond the anchors of the analyzed system, but functionally in the same piping, the system may be qualified to Level D Service Limits. For example, a break in the CVCS injection piping outside of the NPM may be analyzed to Level D limits for the RCS injection piping.

(8) Dynamic load due to SG tube failure or rod ejection accident event is negligible.

(9) The rules in NB-3656(c) may be used as an alternative to NB-3656(a) to evaluate these conditions independent of other design and service loadings, except when it is necessary to demonstrate functional capability of the piping for Level D events.

(10) The range of the resulting moment and the amplitude of the longitudinal force, is evaluated in accordance with NB-3656(b)(4) and NB-3656(b)(5).

(11) Analyze to Level C Service Limits in accordance with NUREG-0800 Section 15.4.8, Acceptance Criterion 2.

(12) Applicable loads are defined in Section 3.12.5.3 and Table 3.9-2.

Table 3.12-2: Required Load Combinations for Class 2 and 3 Piping

Plant Event	Service Level	Load Combination(1)(12)	Allowable Limit
Design	-	P_{des} + DW + B + RF + DFL(2)	Design
Design External Pressure	-	P_{des} + DW + B + ECCS	Design
Refueling	A	P + DW + B + RF	Level A
Thermal Transients	A / B(3)	TH(4)	Level A and B
Maximum Service Pressure	B	P_{max}	Level B
Continued Operating Transients		P + DW + B + DFL	
Reactor Trip Transients / Loss of Power		P + DW + B + DFL(5) + ECCS(5)	
Cold Overpressure Protection		P + DW + B + ECCS	
Maximum Service Pressure	C	P_{max}	Level C
Design Basis Pipe Break		P + DW + B + DBPB(5) + DFL(5) + ECCS(5)	Level C(7)
Spurious ECCS Valve Actuation		P + DW + B + ECCS(5) + DFL(5)	Level C
Spurious RSV Actuation		P + DW + B + RSV(5) + DFL(5) + ECCS(5)	
SG Tube Rupture		P(8) + DW + B + DFL	
Maximum Service Pressure	D(9)	P_{max}	Level D
MSPB and FWPB		P + DW + B + MSPB/FWPB(5) + DFL(5)	
Pipe Breaks + SSE		P + DW + B + DFL(5) + SRSS(SSE + MSPB/FWPB/ECCS) (6)	
		P + DW + B + DFL(5) + SRSS(SSE + DBPB)(6) + ECCS(5)	
		P + DW + B + DFL(5) + SRSS(SSE + RSV)(6) + ECCS(5)	
SAM		SAM(10)	
Rod Ejection Accident		P(8) + DW + B	Level C(11)

Table 3.12-2: Required Load Combinations for Class 2 and 3 Piping (Continued)

Plant Event	Service Level	Load Combination(1)(12)	Allowable Limit
Pressure Test	Test	P + DW	NC/ND-3657

Notes:

(1) When the method of analysis does not retain the sign of cyclic dynamic loads (e.g., reflected pressure waves), they are combined with other loads in the most conservative combination.

(2) The DFL for Service Level A are considered for Design Condition.

(3) Evaluation of OBE loads (both inertial and SAM) is not required for Class 2 and 3 piping. There are no loads applicable to the single non-repeated anchor movement evaluation in NC/ND-3653.2(b).

(4) If Equation (10a) (NC/ND-3653.2(a)) cannot be met for the TH load, the load combination for Equation (11) (NC/ND-3653.2(c)) is P + DW + B + TH.

(5) Applicable dynamic loads (DFL/DBPB/MSPB/FWPB/RSV/ECCS) are combined by absolute sum unless sufficient data are available for each load to understand the time history behavior of each compared to the others (i.e., when the time-phase relationships are known), in which case it is acceptable to use a less conservative approach that complies with NUREG-0484, Revision 1.

(6) The parenthesis indicate that SSE and DBPB/MSPB/FWPB/RSV/ECCS are combined using the square root sum of the squares method and do not indicate order of operations. DFLs resulting from pipe break or valve actuation events may be combined with DBPB/MSPB/FWPB/RSV/ECCS (either by absolute sum or SRSS depending on an evaluation consistent with Note 5 above) before combining with SSE by SRSS. A similar approach may be used for DFLs resulting from an SSE before combining with DBPB/MSPB/FWPB/RSV/ECCS.

(7) For a pipe break beyond the anchors of the analyzed system but functionally in the same piping, the system may be qualified to Level D Service Limits (except where Branch Technical Position 3-4 containment penetration area stress limits govern). For example, a break in the CVCS injection piping outside of the NPM may be analyzed to Level D limits for the Class 3 CNTS CVC injection piping.

(8) Dynamic load due to SG tube failure or rod ejection accident is negligible.

(9) The rules in NC/ND-3655(c) may be used as an alternative to NC/ND-3655(a) to evaluate these conditions independent of other design and service loadings, except when it is necessary to demonstrate functional capability of the piping for Level D events.

(10) The range of the resulting moment and the amplitude of the longitudinal force, is evaluated in accordance with NC/ND-3655(b)(4) and NC/ND-3655(b)(5).

(11) Analyze to Level C in accordance with NUREG-0800 SRP Section 15.4.8, Acceptance Criterion 2.

(12) Applicable loads are defined in Section 3.12.5.3 and Table 3.9-2.

Table 3.12-3: Required Load Combinations for Class 1, 2, and 3 Supports

Plant Event	Service Level	Load Combination(1)(8)	Allowable Limit
Design	-	P + DW + B + TH + DFL(2) P + DW + B + TH + RF + F	Design
Static plus Friction	A	P + DW + B + TH + F	Level A
Normal Operating Transients		P + DW + B + TH + DFL	
Refueling		P + DW + B + TH + RF + F	
Static plus Friction	B	P + DW + B + TH + F	Level B
Continued Operating Transients		P + DW + B + TH + DFL	
Reactor Trip Transients / Loss of Power		P + DW + B + TH + DFL	
Reactor Trip w/ Delayed ECCS		P + DW + B + TH + DFL(4) + ECCS(4)	
Loss of Power		P + DW + B + TH + ECCS(3) + F	
Cold Overpressure Protection		P + DW + B + TH + ECCS(3) + F	
Static plus Friction	C	P + DW + B + TH + F	Level C
Spurious ECCS Valve Actuation		P + DW + B + TH + ECCS(4) + DFL(4)	
Spurious RSV Actuation		P + DW + B + TH + RSV(4) + DFL(4) + ECCS(4)	
SG Tube Rupture		P + DW + B + TH + DFL	
Design Basis Pipe Break		P + DW + B + TH + DBPB(4) + DFL(4) + ECCS(4)	Level C(6)
Static plus Friction	D	P + DW + B + TH + F	Level D
MSPB and FWPB		P + DW + B + TH + MSPB/FWPB(4) + DFL(4)	
Pipe Breaks + SSE		P + DW + B + TH + DFL(4) + SRSS(SSE + MSPB/FWPB/ECCS)(5)	
		P + DW + B + TH + DFL(4) + SRSS(SSE + DBPB)(5) + ECCS(4)	
		P + DW + B + TH + DFL(4) + SRSS(SSE + RSV)(5) + ECCS(4)	
Rod Ejection Accident(7)		P + DW + B + TH + F	Level C

Table 3.12-3: Required Load Combinations for Class 1, 2, and 3 Supports (Continued)

Plant Event	Service Level	Load Combination(1)(8)	Allowable Limit
Pressure Test	Test	P + DW	Test

Notes:

(1) When the method of analysis does not retain the sign of cyclic dynamic loads (e.g., reflected pressure waves or seismic), they are combined with other loads in the most conservative combination.

(2) The DFL for Service Level A are considered for Design Condition.

(3) The ECCS is included in combination with friction only if the ECCS loads have been applied statically in the detailed piping system stress analysis.

(4) Applicable dynamic loads (DFL/DBPB/MSPB/FWPB/RSV/ECCS) are combined by absolute sum unless sufficient data are available for each load to understand the time history behavior of each compared to the others (i.e., when the time-phase relationships are known), in which case it is acceptable to use a less conservative approach that complies with NUREG-0484, Revision 1.

(5) The parenthesis indicate that SSE and DBPB/MSPB/FWPB/RSV/ECCS are combined using the square root sum of the squares method and do not indicate order of operations. DFLs resulting from pipe break or valve actuation events may be combined with DBPB/MSPB/FWPB/RSV/ECCS (either by absolute sum or SRSS depending on an evaluation consistent with Note 4 above) before combining with SSE by SRSS. A similar approach may be used for DFLs resulting from an SSE before combining with DBPB/MSPB/FWPB/RSV/ECCS.

(6) For a pipe break beyond the anchors of the analyzed system but functionally in the same piping, the system may be qualified to Level D Service Limits. For example, a break in the CVCS injection piping outside of the NPM may be analyzed to Level D limits for the RCS injection piping.

(7) Analyze to Level C Service Limits in accordance with NUREG-0800 SRP Section 15.4.8, Acceptance Criterion 2.

(8) Applicable loads are defined in Section 3.12.5.3 and Table 3.9-2.

3.13 Threaded Fasteners (American Society of Mechanical Engineers Code Class 1, 2, and 3)

This section addresses the application of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1 (Reference 3.13-1), to the design of Class 1, 2, and 3 pressure-retaining threaded fasteners. Threaded fasteners and bolted connections, herein called threaded fasteners unless specified differently, include the bolts, studs, and nuts that are associated with Class 1, 2, and 3 pressure retaining joints.

The selection, design, fabrication, installation and inspection of threaded fasteners in the Class 1, 2, and 3 systems meet the criteria of 10 CFR 50.55a, including 10 CFR 50.55a(b)(4). Code cases (Regulatory Guide (RG) 1.84) are not used in the design of threaded fasteners in Class 1, 2, and 3 systems.

The threaded fastener design complies with General Design Criterion (GDC) 1, GDC 4, GDC 14, GDC 30, and GDC 31. Further discussion of compliance with the General Design Criteria are provided in this section.

- General Design Criterion 1 and GDC 30 are met as the bolting design is in conformance with the criteria of ASME BPVC, Section III and RG 1.65 as described below.
- General Design Criterion 4 is met by protecting the ASME Class 1, 2, and 3 threaded fasteners from the adverse impacts from lubricants and sealants and by using stainless steels or nickel-base alloys that are resistant to boric acid corrosion.
- General Design Criterion 14 is met by designing reactor coolant pressure boundary (RCPB) threaded fasteners to ASME Class 1 criteria.
- General Design Criterion 31 is met by ensuring the RCPB components are designed and fabricated with sufficient margin to ensure the RCPB behaves in a non-brittle manner and to minimize the probability of rapidly propagating fracture and gross rupture of the RCPB.

10 CFR 50, Appendix B, Criterion XIII, requires that measures be established to control the cleaning of material and equipment to prevent damage or deterioration. American Society of Mechanical Engineers NQA-1-2015 Part II, Subpart 2.1 provides quality assurance criteria for cleaning fluid systems and associated components that comply with 10 CFR 50 Appendix B. The design for threaded fasteners meets the cleaning criteria in ASME NQA-1-2015 Part II, Subpart 2.1.

3.13.1 Design Considerations

Class 1 pressure boundary threaded fasteners are designed in accordance with ASME BPVC, Section III (Reference 3.13-1), Subsection NB. Class 2 and 3 threaded fasteners are designed in accordance with Subsection NC and ND requirements, respectively.

3.13.1.1 Materials Selection

Applicable criteria used for material selection for ASME Class 1, 2, and 3 threaded fasteners are listed in Table 3.13-1, Criteria for Selection and Testing of Bolted Materials. Materials used for the threaded fasteners are selected for the associated environmental conditions for the lifetime of the plant. Only proven materials for the specific application and environment are used. If washers are used, they are fabricated from materials with mechanical properties that are compatible with the associated nuts.

The reactor coolant and containment vessel closure studs and nuts use SB-637 UNS N07718 (Alloy 718) instead of low alloy steels, as identified in Table 5.2-3, and Table 6.1-1. The selection of Alloy 718 instead of traditional low alloy steels is to prevent general corrosion when the bolting is submerged. Because of its resistance to general corrosion, the concerns addressed by RG 1.65 position 2(b) do not apply to Alloy 718. Alloy 718 is an austenitic, precipitation hardened, nickel-base alloy permitted for bolting materials by ASME BPVC Code Section lll (Reference 3.13-1), Subsection NB-2128.

Because Alloy 718 is a nonferrous material, ASME BPVC, Section III (Reference 3.13-1), Subsection NB-2311 exempts it from the fracture toughness test requirements in NB-2300. The minimum required room temperature yield strength of SB-637 Alloy 718 is 150 ksi, exceeding the 150 ksi maximum limit in RG 1.65 position 1(a)(i). Because Alloy 718 is nonferrous, it is not subject to the fracture toughness requirements in 10 CFR 50 Appendix G or RG 1.65 and the concern addressed by RG 1.65 position 1(a)(i) is not applicable to Alloy 718.

Alloy 718 is resistant to stress corrosion-cracking (SCC) when exposed to high temperature primary reactor coolant, although limited SCC is observed inside reactor vessel internals (Reference 3.13-3). However, SCC is unlikely for reactor vessel closure bolting because it is submerged at a much lower temperature than reactor coolant temperature. The solution treatment temperature range prior to precipitation hardening treatment is restricted to 1800 degrees F to 1850 degrees F in order to improve SCC resistance. The heat treatment is identical to SB-637 UNS N07718 except for the more restrictive solution temperature range. This heat treatment process provides better resistance to SCC and is within the limits of ASME Section II (Reference 3.13-2) material specification for Alloy 718.

The heat treatment for the mitigation of SCC is applied to Alloy 718 threaded fasteners in environments and applications where they are used in Class 1, 2, and 3 pressure retaining joints, including applications in low flow areas and low temperature systems.

Consistent with RG 1.65, lubricants are selected in accordance with the guidance in NUREG-1339 (Reference 3.13-4). Alloy 718 bolting material is in compliance with RG 1.65 requirements except for the requirements not applicable to Alloy 718 bolting as described above.

3.13.1.2 Special Materials Fabrication Processes and Controls

The criteria for mechanical property testing of threaded fasteners complies with the requirements of ASME BPVC, Section II (Reference 3.13-2), Part A and Part B, and also complies with ASME BPVC Section III (Reference 3.13-1) NB-2200, NC-2200, ND-2200, NB-2300, NC-2300, and ND-2300 as noted in Table 3.13-1 for ferritic threaded fasteners. Threaded fastener materials are chosen from proven materials for the specific application and environment and are used after evaluation of the potential for degradation, including galvanic corrosion and SCC. The bolting materials selected for the ASME Class 1, 2, and 3 threaded fasteners are discussed in Section 4.5, Section 5.2, and Section 6.1. Alloy 718 nuts for the reactor pressure vessel and containment vessel main flanges have surface treatments applied to reduce galling.

Fabrication and examination of threaded fasteners are performed in accordance with the criteria in Table 3.13-1 for ASME Code Class 1, 2, and 3 systems.

Lubricants used for the threaded fasteners covered by this section comply with the guidance in NUREG-1339 (Reference 3.13-4) to avoid galvanic corrosion and SCC.

3.13.1.3 Fracture Toughness Requirements for Threaded Fasteners Made from Ferritic Materials

The pressure-retaining Class 1, 2, and 3 components made of ferritic material meet the requirements of ASME BPVC, Section III (Reference 3.13-1), Subsections NB-2300, NC-2300 and ND-2300 respectively (Table 3.13-1). For pressure-retaining components of the reactor coolant pressure boundary, the requirements are supplemented by the additional requirements set forth in 10 CFR 50, Appendix G.

3.13.1.4 Pre-Service Inspection Requirements

Pressure boundary Class 1, 2, and 3 threaded fasteners are examined in accordance with ASME BPVC, Section XI (Reference 3.13-5), Subsections IWB-2200, IWC-2200 and IWD-2200 respectively for pre-service inspection. Containment vessel pressure retaining threaded fasteners are examined in accordance with ASME BPVC, Section XI, Subsection IWE-2200.

3.13.1.5 Certified Material Test Reports (Quality Assurance Records)

Pressure-retaining Class 1, 2, and 3 threaded fasteners are certified in accordance with Subsection NCA-3861 and Subsection NCA-3862 and are furnished with certified material test reports in accordance with the criteria of ASME BPVC, Section III (Reference 3.13-1) Subsections NB-2130, NC-2130 and ND-2130, respectively.

Material identification is required for Class 1, 2, and 3 threaded fasteners per ASME BPVC, Section III (Reference 3.13-1), Subsections NB-2150, NC-2150, ND-2150, respectively. Certified material test reports for ASME Section III

Class 1, 2, and 3 threaded fasteners are retained in accordance with 10 CFR 50.71.

3.13.2 Inservice Inspection Requirements

Inservice Inspection for ASME Class 1, 2, and 3 threaded fasteners is in accordance with the ASME BPVC, Section XI (Reference 3.13-5), as required by 10 CFR 50.55a, except where specific written relief is granted by the Nuclear Regulatory Commission.

COL Item 3.13-1: An applicant that references the NuScale Power Plant US460 standard design will provide an inservice inspection program for American Society of Mechanical Engineers Class 1, 2, and 3 threaded fasteners. The program will identify the applicable edition and addenda of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI and ensure compliance with 10 CFR 50.55a.

3.13.3 References

3.13-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section III, "Rules for Construction of Nuclear Facility Components," New York, NY.

3.13-2 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section II, "Materials," New York, NY.

3.13-3 McIlree, A.R., "Degradation of High Strength Austenitic Alloys X-750, 718 and A286 in Nuclear Power Systems," Proceedings of the 1st International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors," National Association of Corrosion Engineers, 1984.

3.13-4 U.S. Nuclear Regulatory Commission, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," NUREG-1339, June 1990.

3.13-5 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," New York, NY.

Table 3.13-1: American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III Criteria for Selection and Testing of Bolted Materials

Code Category		ASME Class 1 Criteria	ASME Class 2 Criteria	ASME Class 3 Criteria
Material Selection		NCA-1220 and NB-2128	NCA-1220 and NC-2128	NCA-1220 and ND-2128
Material test coupons and specimens for ferritic steel material (tensile test criteria)	Heat Treatment Criteria	NB-2210	NC-2210	ND-2210
	Test coupons requirements bolting and studing materials	NB-2221 NB-2224	NC-2221 NC-2224.3	ND-2221 ND-2224.3
Fracture toughness requirements	Materials to be impact tested	NB-2311	NC-2311	ND-2311
	Types of impact test	NB-2321	NC-2321	ND-2321
	Test coupons	NB-2322	NC-2322	ND-2322
	Acceptance standards	NB-2333	NC-2332.3	ND-2333
	Number of impact tests necessary	NB-2345	NC-2345	ND-2345
	Retesting	NB-2350	NC-2352	ND-2352
	Calibration of test equipment	NB-2360	NC-2360	ND-2360
Examination criteria for bolts, studs, and nuts		NB-2580	NC-2580	ND-2580
Certified material test report criteria		NCA-3860	NCA-3860	NCA-3860

Note 1: Section III paragraphs listed in this table represent those specified in the 2017 Edition of Section III.

Note 2: The threaded fasteners for the reactor vent valve and reactor recirculation valve connections are inspected per NB-2581 and NB-2584. Additionally the threaded fasteners are inspected as per NB-2586 after threading.

Appendix 3A Dynamic Simulation of the NuScale Power Module

3A.1 Seismic Analysis

The dynamic simulation of the NuScale Power Module (NPM) consists of a system model to determine the seismic dynamic response of the reactor pressure vessel, containment vessel, reactor internals and core support, reactor core, surrounding pool water, and structures, systems, and components (SSC) supported by the NPM. The dynamic simulation of the complete NPM system is performed using time history dynamic solution methods and a three dimensional (3-D) ANSYS software program (Section 3.9.1) finite element model. The NPM system model includes acoustic elements to represent the effects of fluid-structure interaction (FSI) due to pool water found between the CNV and pool floor and walls.

To account for possible dynamic coupling of the NPMs and the reactor building (RXB) system, a model of each of the NPMs is included in the RXB system model as described in TR-121515, NuScale Power Module Seismic Analysis (Reference 3A-1), and Section 3.7.2.

The RXB model applies the frequency domain soil-structure-fluid interaction seismic analysis methodology described in TR-0118-58005-P-A, Revision 2, Improvements in Frequency Domain Soil Structure Fluid Interaction (Reference 3A-2). Results from the RXB seismic system simulation include in-structure time histories at each NPM support location and the pool walls and floor surrounding the NPM. In-structure response spectra are also calculated. Results are shown in Section 3.7.2.

The detailed dynamic simulation of the NPM system is performed using a 3-D ANSYS model, using the in-structure time histories obtained from the RXB seismic simulation as input. The NPM dynamic simulation provides in-structure time histories and in-structure response spectra for qualification of equipment supported by the NPM and time histories at core support locations for seismic qualification of fuel assemblies.

The methodology and results of the seismic simulation of the NPM are provided in technical report TR-121515, NuScale Power Module Seismic Analysis (Reference 3A-1).

3A.2 Blowdown Simulation

The blowdown simulation addresses events caused by the failure of piping and the actuation of valves, including high-energy line breaks inside the containment vessel. These short-term transient events result in system internal pressure waves and asymmetric cavity pressurization waves external to the pipe break or valve outlet.

Short-term transient events require special treatment due to their rapidly changing thermal hydraulic conditions and resulting dynamic mechanical loads. In addition to the rapid nature of these transients, fluid-structure interactions are influential and are therefore also considered.

The methodology and results of the blowdown simulation of the NPM is provided in technical report TR-121517, NuScale Power Module Short-Term Transient Analysis (Reference 3A-3).

3A.3 References

3A-1 NuScale Power, LLC, “NuScale Power Module Seismic Analysis Technical Report,” TR-121515, Revision 0.

3A-2 NuScale Power, LLC, “Improvements in Frequency Domain Soil Structure Fluid Interaction Technical Report,” TR 0118 58005 A, Revision 2.

3A-3 NuScale Power, LLC, “NuScale Power Module Short Term Transient Analysis Technical Report,” TR-121517, Revision 0.

Appendix 3B Design Reports and Critical Section Details

This appendix summarizes the structural design and analysis of the Reactor Building (RXB) and Control Building (CRB). Section 3.8.4 and Section 3.8.5 describe these structures, their foundations, and the primary loads and load combinations. This appendix describes how those loads are combined and how the design is checked for adequacy. In addition, analysis and design of a selection of critical sections are described in detail. These sections are critical in that they represent parts of the structure that: (1) perform a safety-critical function, (2) are subjected to large stress demand, (3) are considered difficult to design or construct, or (4) are considered to be representative of the structural design. For example, the walls and basemat slab at the NuScale Power Module (NPM) bays satisfy the first three criteria.

Based on the above criteria for critical sections, the following structural members are presented in this section for the RXB:

Steel-Plate Composite (SC) Walls:

- at grid line 1 - West exterior structural wall
- at grid line 4 - Interior structural wall spanning from basemat to roof level
- at grid line B - Interior structural wall at the north side of pool spanning from basemat to roof level
- at grid line E - South exterior structural wall
- around the pool area - Interior structural walls around the pool area. Some of these walls are part of the SC walls at grid lines 4 and B
- between NPMs - Interior structural walls between NPMs within the pool area

Reinforced Concrete (RC) Slabs:

- RC-basemat slab at Elevation (EL) 25 ft
- RC-floor slab at EL 100 ft
- RC-roof slab at EL 187 ft

Miscellaneous:

- Reactor Building crane Corbel
- NPM skirt restraint
- NPM lug restraint

Based on the above criteria for critical sections, the following structural members are presented in this section for the CRB:

Reinforced Concrete Walls:

- at grid line 3 - Interior structural wall
- at grid line 5 - East exterior structural wall
- at grid line H - South exterior structural wall

Reinforced Concrete Slabs:

- RC-basemat slab at grade at EL.100 ft
- RC-floor slab at EL.123 ft

Table 3B-1 and Table 3B-2 outline the critical sections and details for the RXB. Table 3B-3 and Table 3B-4 outline the critical sections and details for the CRB.

Section 1.2 contains floor plan and section cut drawings of the RXB and CRB.

Section 3B.1 discusses the design methodology used for both buildings. Section 3B.2 provides the design report and critical section details for the RXB, and Section 3B.3 provides that information for the CRB.

3B.1 Methodology

Design of structural members is primarily based on ACI 349-13 (Reference 3B-3) and, as applicable, ACI 318-08 (Reference 3B-6) for RC members and AISC N690-18 (Reference 3B-4) and, as applicable, AISC 360-10 (Reference 3B-5) for SC walls. RC slabs and SC walls are discussed in the Building Design and Analysis Methodology for Safety Related Structures (Reference 3B-1). The design methodology for RC walls is not covered in Reference 3B-1 and is discussed in Section 3B.1.4.3.

ANSYS (Reference 3B-2) is used for the generation of finite element models, calculating structural response for static and dynamic loads as described in Sections 3.7 and 3.8, and post-processing of the analysis results. Soil 7, 9, and 11, along with soil 7 with soil separation are considered in design calculations as applicable. In this context, soil 7 with soil separation is treated as a separate new soil type.

The design calculations are performed to determine the section thickness, and reinforcement layout of RC members; and, wall thickness, thickness of steel faceplates and size and spacing of tie reinforcement for SC walls. Demand-to-capacity ratio (DCR) less than or equal to 1.0 is deemed acceptable to meet design requirements.

The DCR for different required strengths (design conditions) are calculated separately for each seismic input motion and the resulting DCR values are averaged for the same soil type. The averaged DCR values are enveloped for all soil types and load combinations.

3B.1.1 Structural Response

The following structural responses are calculated:

- Finite element forces at individual elements comprising walls, slabs and basemat. Element-based results are used to design SC walls and also to confirm the location of selected section cuts for design of RC members.
- Force resultants at selected section cuts. These section cuts are used to design RC members including floor slabs, walls, basemat, and roof.

The finite element forces are obtained in the local element coordinate system, shown in Figure 3B-1. Typically, the finite element responses are expressed in terms of local directions x, y, and z. In ANSYS, however, the finite element responses are expressed in terms of local directions 1, 2, and 3 (e.g., N11, M22, etc.). The equivalence is as follows:

- Local axis x = 1
- Local axis y = 2
- Local axis z = 3

In the ANSYS models, the finite element local coordinate system is defined such that the local 3 (z) axis is perpendicular to the plane and follows the positive direction of the global axes X, Y, or Z. Thus, for slabs, axis 1 (x) is along the global axis X (towards east), and axis 2 (y) is along global axis Y (towards north). For walls, axis 1 (x) follows global axis X or Y depending on the wall orientation, and axis 2 (y) follows global axis Z.

Section cuts resultant forces and moments are resolved at the geometric center of the cut plane. The force and moment resultants are obtained in the section cut local coordinate system, which is shown in Figure 3B-2. The local 1 axis is along the positive global axis that is parallel to the normal of the section cut plane. The local 2 axis is along the positive global axis that is parallel to the longitudinal direction of the section cut. The local 3 axis is the cross product of local 1 and local 2 axes. The demand calculated at each section cut using the local coordinate system. Depending on the section cut elements and nodes selection and the local coordinate system, the sign of the calculated outputs are reversed such that the positive axial load (F1) always corresponds to tension in the section cut.

- F1, the axial load (P) - positive is tension in the section cut.
- F2, the in-plane shear (V)
- F3, the out-of-plane shear (Vz)
- M1, the torsional moment (T)
- M2, the out-of-plane moment (M) - positive is tension at the surface that is on the positive side of the member section with respect to local 3 axis.
- M3, the in-plane moment (Mz)

The seismic response is calculated for the five in-layer motions (sets 1 through 5 in Table 3B-5) for soil types 7 and 11, and one in-layer motion (set 6 in Table 3B-5) for soil type 9 calculated at 85 ft below grade. The amplitude of the horizontal components (E and N) of the input motions are increased by 5 percent to account for accidental torsion in the structural system.

3B.1.2 Loads and Load Combinations

The loads and load combinations used in the design calculations are discussed in detail in Sections 3.7.1 and 3.8.4 for seismic and non-seismic loads. The design

calculations are performed using the load combinations from ACI 349 for design of RC members and load combinations from AISC N690 for SC walls.

The design calculations are also performed for the governing load combinations with reduced static loads to account for the adverse effects of reduced loads in accordance with Section 9.2.3 of ACI 349 for RC members and Section NB2.5d of AISC N690 for SC wall members.

For the RXB, the accident pressure, P_a, is considered through one global condition where 6 psid pressure is applied to the structural members around the pool region including the roof, and three isolated conditions where 11 psid is applied to the galleries between floors at EL 55 ft and 70 ft, 7 psid to the galleries between floors at EL 100 ft and 126 ft, and 7 psid is applied to the module heatup system (MHS) equipment room at EL 55 ft. The demands from the global condition are considered for design of all structural members. The demands from localized, isolated, cases of P_a are only considered for floors at EL 55 ft, 70 ft, 100 ft, and 126 ft and SC walls around these floors.

In design calculation of Reactor Building RC members, the thermal effects (T_o and T_a) are evaluated considering usable strength as discussed in Section 3B.1.4.6. In design calculation of SC walls, the thermal effects and accident pressure are considered as outlined by AISC N690.

3B.1.3 Cracked States Used in Load Combinations

The structural analyses are performed considering different cracked states associated with each load combination. The stiffness of members for out-of-plane bending is always considered as cracked and the cracking state due to in-plane demands is determined using the following seismic load combination:

$$D + 0.8L + E_{ss} \qquad \text{Eq. 3B-1}$$

The label CrkEs is assigned to the cracking state determined through the above load combination. The CrkEs cracking state is used in seismic (E_{ss}) and all individual static load cases (D, L, H, etc.) that are part of the extreme seismic load combination (LC6).

For the accident conditions, the out-of-plane bending stiffness of SC walls is further reduced to account for accident temperatures (T_a) per Section N9.2 of AISC N690 and all RC and SC-wall members are considered to be cracked for in-plane demands. The label CrkAll is assigned to this cracking state. CrkAll cracking state is used for seismic and also static load cases that are part of the seismic load combination LC9 except the load cases T_a, P_a, and R_a. For load cases T_a, P_a, and R_a, in addition to the CrkAll cracking state, the SC-wall members are further evaluated for axial cracking due to stresses from accident temperatures (T_a) and members determined to have axial cracking are assigned axial stiffness corresponding to that provided only by the steel faceplates. The label CrkAllTa is assigned to this cracking state. CrkAllTa cracking state is used

for T_a, P_a, and R_a that are part of the accident load combinations LC8 and LC9. For accident load combination LC8, the load cases that are not part of accident conditions (e.g. D, L, and H) are conservatively analyzed using the cracked state of members where uncracked state is used for in-plane actions and cracked state is used for out-of-plane bending. The label UnCrk is assigned to this cracking state. The UnCrk cracked state is also used for individual load cases that are part of all non-seismic and non-accident load combinations (e.g. wind and gravity load combinations).

Based on the above conditions, the following cracked states are used in the RXB design calculations:

1) UnCrk: This state represents the "uncracked" conditions. In this state, the out-of-plane bending is always considered as cracked for RC members and SC walls. This state is used in analysis of all load cases that are part of load combinations other than those with seismic (LC6 and LC9) and accident (LC8) load cases. This state is applicable to both RXB and CRB.

2) CrkAll: For this cracked state, all RC members and SC walls are assigned effective in-plane shear stiffness. The SC walls are assigned the lowest effective in-plane shear stiffness in accordance with N690. Furthermore, SC walls have additional flexibility in out-of-plane bending due to accident temperatures. This cracked state is used in analysis of all load cases (except T_a, R_a, and P_a) that are part of abnormal seismic load combination LC9. This state is only applicable to the RXB.

3) CrkAllTa: This cracked state is the same as the CrkAll cracked state with additional axial cracking in SC walls using stresses from accident temperature (T_a) case from the analysis model with CrkAll cracked state. This state is used in accident load cases (T_a, R_a, and P_a) that are part of accident load combination LC8 and abnormal seismic load combination LC9. This state is only applicable to the RXB.

4) CrkTo: This cracked state is the same as the UnCrk cracked state with additional axial cracking in SC walls and in-plane shear cracking in SC walls and RC members using stresses from operating temperature case (T_o) from the analysis model with UnCrk cracked state. This cracked state is used in operating temperature load case T_o in non-seismic load combinations. This cracked state is not used in design calculations as the operating temperatures are not included in design load combinations since accidental temperatures in LC8 and LC9 control the design. This cracked state is used in fragility calculations and is only applicable to the RXB.

5) CrkEs: This cracked state is the same as the UnCrk cracked state with in-plane cracking based on seismic combination ($D + 0.8L + E_{ss}$). This cracked state is used in the analysis of all individual static load cases (except T_o) and the seismic load case, E_{ss}, that are part of the extreme seismic load combination LC6. This cracked state is applicable to both RXB and CRB.

6) CrkEsTo: This cracked state is the same as the CrkEs cracked state with additional axial cracking in SC walls and in-plane shear cracking in SC walls and RC members using stresses from operating temperature (T_o) case from the analysis model with CrkEs cracked state. This cracked state is used in operating temperature load case T_o in extreme seismic load combination LC6. This cracked state is not used in design calculations as the operating temperatures are not included in design load combinations since accidental temperatures in LC8 and LC9 control the design. This cracked state is used in fragility calculations and is only applicable to the RXB.

   Table 3B-8 and Table 3B-9 provide the cracked states used in different load combinations for RC members and SC walls respectively in the RXB.

3B.1.4 Reinforced Concrete Wall and Slab Design Methodology

Design calculations are performed for the following design conditions defined in ACI 349:

- Axial force and out-of-plane moment interaction
- Out-of-plane shear
- In-plane shear

3B.1.4.1 Required Design Strength

The required design strengths of RC members are calculated using section cuts at critical regions of the members. The critical locations are determined using the guidelines in Reference 3B-1 for slabs, in Section 3B.1.4.3 for walls, and reviewing the contour plots of peak demand values for different load combinations.

3B.1.4.2 RC Slab Design Approach

The design approach for SC-I RC slabs is documented in Section 8 of TR-0920-71621-P-A-R1 (Reference 3B-1).

3B.1.4.3 RC Wall Design Approach

Structural walls are primarily designed for in-plane shear and in-plane moment. Walls are designed for the in-plane actions due to seismic loads using the special seismic requirements for special structural walls of ACI 349 Section 21.9 (Reference 3B-3).

The design of walls for out-of-plane forces follows the same requirements for conventional RC slabs outlined in in Section 8 of TR-0920-71621-P-A-R1 (Reference 3B-1).

In structural walls carrying low gravity loads, such as the massive walls encountered in safety-related structures, axial loads are mainly due to in-plane flexure resulting from lateral loads; and a small portion is due to gravity loads.

Thus, the axial load is considered in wall design by calculating the in-plane moment at critical section cuts, as shown in Section 3B.1.4.3.4.

For efficiency in the design process, the total axial load is included in the out-of-plane flexural design (i.e., P-M interaction). By calculating the axial load at discrete section cuts for out-of-plane flexure, including regions with large axial forces, the effect of the in-plane moment is included in the design.

Additionally, the in-plane moment is explicitly calculated in the following cases:

- At sections at the base of the wall. As shown in Section 3B.1.4.3.4, the section cuts for in-plane forces extend the entire wall length, which includes portions of perpendicular walls. Since the portions at the wall ends (boundaries) are usually not included in the section cuts for out-of-plane design (Section 3B.1.4.3.3), a confirmatory in-plane moment check is performed as explained in Section 3B.1.4.3.4.
- In cases where there is a need for wall boundary elements or special transverse reinforcement at the wall boundaries (Section 3B.1.4.3.9), the width of the wall boundaries is determined from a moment curvature analysis considering the in-plane moment and the axial load at that section of the member.

Similar to slabs, the effect of the torsional moment, T, is an increase in the out-of-plane shear force, V_z. Thus, by calculating the out-of-plane shear at discrete section cuts at critical member regions including regions with increased shear due to torsion, the effect of the torsional moment is included in the design process.

In summary, the required strengths for wall design are:

- Axial load, P, and out-of-plane moment, M, are used to design for axial-out-of-plane moment (P-M) interaction.
- Axial load, P, and out-of-plane shear, Vz, are used to design for axial-out-of-plane shear (P-Vz) interaction.
- The in-plane shear, V, is used to design for shear and shear-friction at slab-wall interfaces.
- The in-plane moment, Mz, and axial load, P, are used to verify in-plane moment capacity at wall sections including perpendicular walls and to assess the necessity of boundary elements.

Section 3B.1.4.3.3 through Section 3B.1.4.3.5 provide guidelines to determine critical sections where largest demand is expected in walls subjected to gravity and lateral loading. For clarity, critical locations are shown separately for each demand type, which are grouped as out-of-plane forces and in-plane forces.

The critical section cut locations shown in Section 3B.1.4.3.3 through Section 3B.1.4.3.5 are considered minimum locations to calculate structural demand; thus, additional section cuts are considered to investigate the

possibility of jigher demand outside these critical locations based on the peak element force results as discussed in Section 8.5.4 of Reference 3B-1.

3B.1.4.3.1 Criteria to Determine Slenderness Effects in Walls and Columns

Slenderness effects are not considered for walls and columns with clear height to smallest horizontal dimension ratio equal to or smaller than 7.3. This value is obtained as follows.

According to ACI 349 Section 10.10 (Reference 3B-3), slenderness effects are evaluated along two major axes in compression members. One method to account for slenderness effects is by magnification of moments obtained from the elastic analysis. Different procedures are followed for nonsway and sway stories. The floor slabs in the buildings are supported by massive walls in both directions, and thus, the floors are considered nonsway. For this condition, ACI 349 Equation 10-7 (Reference 3B-3) is used to determine whether slenderness effects can be considered as insignificant.

$$\frac{kl_u}{r} \leq 34 - 12\left(\frac{M_1}{M_2}\right)$$ ACI 349 (Reference 3B-2)

Where:

k is the effective length factor, which for elastic support condition is conservatively taken as 0.9 (Reference 3B-7).

r is the radius of gyration, which can be taken as 0.3h, where h is the smallest column dimension or the wall thickness.

M_2 and M_1 are the larger and smaller moments at the column ends, respectively.

l_u is the unsupported column height.

Examining ACI 349 Eq. 10-7 (Reference 3B-3), the smallest value of $34 - 12(M_1/M_2)$ is 22, which occurs when M_1=M_2. Thus, considering r = 0.3 h, and k = 0.9, the maximum aspect ratio below which the slenderness effects can be neglected is calculated as l_u / h = 7.3.

3B.1.4.3.2 Small Openings in Walls and Diaphragms

Openings in walls and diaphragms smaller than two times the member thickness are typically not modeled provided that the total amount of reinforcement required for the panel without the opening is maintained, shear requirements of ACI 318 Section 11.11.6 (Reference 3B-6) are

satisfied, and amount of reinforcement interrupted is outside column strips. The additional reinforcement is equally placed at either side of the opening.

For out-of-plane loading of walls and slabs without beams, openings of any size are permitted by ACI 318 Section 13.4 (Reference 3B-6), provided that the total amount of reinforcement required for the panel without the opening is maintained, shear requirements of ACI 318 Section 11.11.6 (Reference 3B-6) are satisfied, and amount of reinforcement interrupted is limited by location (e.g., inside and outside of column strips). This convention is known as the ACI rebar area replacement rule. However, ACI does not contain similar criteria for walls and diaphragms subject to in-plane loading. In these situations, walls and diaphragm segments generated by the opening and other boundaries are analyzed independently.

Small openings do not significantly change the distribution of in-plane forces. Thus, design practice (Reference 3B-8) is to use the ACI rebar area replacement rule for openings smaller than two times the member thickness.

3B.1.4.3.3 Section Cuts for Out-of-Plane Forces

Wall panels supported in all four edges by floor slabs or perpendicular walls behave similar to slabs when they are subject to out-of-plane forces. Thus, using the same critical locations described in Section 8.5.1 of Reference 3B-1, the section cut locations to obtain the demand due to out-of-plane forces in wall panels supported in all four edges are:

- For out-of-plane moment, at about the center of each of the slab-wall interfaces; and, at a section about the center of the wall short span (main direction of out-of-plane moment transfer), as shown in Figure 3B-3.
- For out-of-plane shear, at about the center of each of the slab-wall interfaces as shown in Figure 3B-4.

3B.1.4.3.4 Section Cuts for In-Plane Forces

The main function of structural walls is to resist lateral forces by horizontal in-plane shear and in-plane bending. Seismic forces from the floor slabs are transferred to the walls as in-plane shear that gradually increases from higher elevations to the lower levels.

For squat cantilever walls (i.e., walls with height to length ratio equal or less than 2) typically encountered in nuclear power plants, the largest in-plane shear and moment are found at the base of the wall. However, wall embedment may change the moment and shear distribution of a typical cantilever wall. Therefore, in-plane shear and in-plane moment is evaluated at section cuts at the base of the walls at each story.

The in-plane moment is equilibrated by a compression and tension couple or chord forces at the wall boundaries. Critical section cut locations to evaluate in-plane shear and chord forces for a one-story cantilever wall are shown in Figure 3B-5a. As shown, the section cuts for in-plane shear and chord forces include portions of perpendicular walls. At the section cut locations for chord forces, the out-of-plane moment is also calculated.

When the section cuts for out-of-plane moment do not include portions of perpendicular walls, a part of the chord force is missing from the axial load. Thus, a confirmatory check is performed where the in-plane moment demand obtained at the same section cut for in-plane shear (Figure 3B-5b) is verified to be below the capacity calculated considering all the longitudinal reinforcement present in the section. This confirmatory check is necessary since large axial forces may be present at the wall boundaries, thus increasing the in-plane moment at the wall.

3B.1.4.3.5 Section Cuts for Forces around Openings

Small openings, on the order of one or two wall thicknesses, are typically not modeled (Section 3B.1.4.3.2). For larger openings, the wall is designed to transfer the forces around the opening as described in this section.

In accordance with ACI 349, walls with openings are considered to be composed of vertical and horizontal wall segments (Figure 3B-6). A vertical wall segment is bounded horizontally by two openings or by an opening and an edge. Similarly, a horizontal wall segment is bounded vertically by two openings or by an opening and an edge.

Vertical wall segments transfer shear forces from the wall portion above to the wall portion below. Thus, additional section cuts are defined at the base of vertical wall segments. Figure 3B-7a shows additional section cuts to calculate in-plane shear and chord forces for the two vertical wall segments formed by the opening and the wall vertical edges.

Horizontal wall segments, on the other hand, behave more like beams in which the shear force is transfer horizontally from one end to the other end. Thus, additional section cuts are defined at each end of horizontal wall segments. Figure 3B-7a shows additional section cuts to calculate in-plane shear and chord forces for the horizontal wall segment formed by the opening and the wall top horizontal edge.

A confirmatory check is performed where the in-plane moment demand obtained at the same section cuts for in-plane shear (Figure 3B-7b) is verified to be below the capacity calculated considering all the longitudinal reinforcement obtained for in-plane shear and out-of-plane moment.

Out of-plane forces around openings are transferred in the same way as described in Section 8.5.3 of Reference 3B-1 for slabs. The section cut

locations to calculate the out-plane moment and out-of-plane shear around an opening in a wall are shown in Figure 3B-8.

3B.1.4.3.6 Minimum Reinforcement in Walls

In accordance with ACI 349 Section 14.3 (Reference 3B-3), the minimum ratio of longitudinal reinforcement area to gross concrete area, ρ_l, is 0.0015. The minimum ratio of transverse reinforcement area to gross concrete area, ρ_t, is 0.0025.

The longitudinal and transverse reinforcement spacing is defined to be less than three times the wall thickness, or 18 in.

For walls more than 10 in. thick, the reinforcement is provided in two layers on each face in each direction in accordance with ACI 349 Section 14.3.4 (Reference 3B-3).

3B.1.4.3.7 Minimum Concrete Cover

In agreement with ACI 318 Section 7.7.1 (Reference 3B-6), the minimum concrete cover for exterior walls or walls exposed to earth is defined as 2 inches. For interior walls not exposed to weather, the minimum concrete cover is defined as 1-1/2 inches for bars No. 14 or larger and 3/4 in. for smaller bars.

3B.1.4.3.8 Lateral Reinforcement for Compression Members

This section applies to columns and walls and slabs where longitudinal reinforcement is required for compressive strength (i.e., when the section fails in compression).

Lateral reinforcement for compression members is defined as spirals or ties. The requirements for spirals and ties are discussed in ACI 349 Sections 7.10.4 and 7.10.5, respectively (Reference 3B-3). The following description discusses main requirements for ties.

Ties are provided in compression members for the following reasons:

1) Ties restrain the longitudinal bars from buckling out through the surface

2) Properly detailed ties confine the concrete core, providing increased ductility.

3) Ties serve as shear reinforcement.

According to the first item listed above, ACI 349 Section 7.11 (Reference 3B-3) specifies that compression reinforcement in beams are to be enclosed by ties or stirrups satisfying the size and spacing

requirements for compression members in ACI 349 Section 7.10.5 (Reference 3B-3) and shown below.

In compression members, all bars are enclosed by lateral ties, at least No. 3 in size for longitudinal bars No. 10 or smaller, and at least No. 4 in size for larger and bundled bars.

The maximum vertical spacing of ties is given by ACI 349 Section 7.10.5.2 (Reference 3B-3) as:

s_{max} ≤ 16 x longitudinal bar diameter

≤ 48 x tie bar diameter

≤ least dimension of the member

ACI 349 Section 7.10.5.3 (Reference 3B-3) outlines the arrangement of ties in a column cross section. These arrangements are illustrated in ACI 349 Figure R7.10.5 (Reference 3B-3).

ACI 349 Section 7.10.5.4 (Reference 3B-3) requires that the bottom and top ties be placed not more than one-half of a tie spacing above or below the slab, respectively.

ACI 349 Section 7.9.1 (Reference 3B-3) requires that bar anchorages in connections of beams and columns be enclosed by ties, spirals, or stirrups. Generally, ties are most suitable for this purpose and are arranged in agreement with ACI 349 Section 7.10.5.4 (Reference 3B-3).

3B.1.4.3.9 Wall Boundary Elements

In agreement with ACI 349 Section 21.9.6 (Reference 3B-3), for walls subject to seismic loads, the necessity of boundary elements are investigated as follows:

- For walls and vertical wall segments with $\frac{h_w}{t_w} \leq 2$, boundary elements are not required
- For walls and vertical wall segments with $\frac{h_w}{t_w} > 2$, boundary elements are required if the maximum compression strain in the cross section analysis exceeds 0.002

Where boundary elements are required, the requirements of ACI 349 Section 21.9.6.4 (Reference 3B-3) are satisfied. In this case, the transverse reinforcement in the boundary elements is subject to similar requirements as for columns in moment frames resisting seismic forces. The extension of the boundary element (i.e., width) is determined based on the largest neutral axis depth, c, calculated for the factored axial force

and nominal moment strength consistent with the design forces. The value of c is calculated from a moment curvature analysis considering all provided reinforcement in the concrete section.

To avoid the transverse reinforcement requirements for boundary elements, the usable compression strain in walls with $\frac{h_w}{t_w} > 2$ is limited to 0.002.

Where boundary elements are not required, the following is satisfied:

- If the longitudinal reinforcement ratio at the wall boundary is greater than 430/fy, the boundary transverse reinforcement satisfy ACI 349 Sections 21.6.4.2 and 21.9.6.4(a) (Reference 3B-3). The boundary transverse reinforcement is the reinforcement located within the same width as calculated for boundary elements, obtained from a moment curvature analysis as explained above. The maximum longitudinal spacing of the transverse reinforcement in the boundary is defined to not exceed 10db or 12 in.; and,
- Except when Vu in the plane of the wall is less than $A_{cv}\sqrt{f'_c}$, the horizontal reinforcement terminating at the edges of walls without boundary elements defined shall have a standard hook engaging the edge reinforcement.

3B.1.4.3.10 Out-of-Plane Shear Capacity of Walls

From ACI 318 Sections 11.9 and 11.11 (Reference 3B-6), the shear strength of walls, slabs and basemat in the vicinity of columns, concentrated loads, or reactions is governed by the more severe of beam action (Section 8.11.1 of Reference 3B-1) or two-way action (Section 8.11.2 of Reference 3B-1).

The nominal out-of-plane shear strength is calculated for 1 foot of member section; that is, b_w =12 in.

3B.1.4.3.11 In-Plane Shear Capacity of Walls

Design for in-plane shear forces due to non-seismic loads is in accordance with ACI 318 Sections 11.9.2 through 11.9.9 (Reference 3B-6).

Design for in-plane shear forces due to seismic loads is in accordance with ACI 318 Section 21.9.4 (Reference 3B-6).

3B.1.4.3.12 In-Plane Shear Transfer through Shear Friction

Shear friction is evaluated across interfaces between two different volumes of concrete. In general, shear friction is evaluated at construction joints where fresh concrete is placed against previously hardened

concrete. In this case, a weak plane may exist at the joint or in concrete immediately adjacent to the joint. The construction process is reviewed to identify these locations. The nominal shear-friction capacity along the critical section is calculated per ACI 349 Section 11.6 (Reference 3B-3).

3B.1.4.3.13 Flexural Capacity

This section applies to the in-plane and out-of-plane bending of walls. These members are subject to combined flexure and axial forces; therefore, the design procedure for beam-column members in ACI 349 Chapter 10 (Reference 3B-3) is used. The combined axial-flexural load (P-M) strength is obtained through the construction of P-M interaction diagrams for bending in one direction. The calculations follow the basic design assumptions of ACI 349 Section 10.2 (Reference 3B-3) summarized below:

3B.1.4.3.14 Out-of-Plane Flexural Capacity of Walls

P-M interaction diagrams are calculated for one foot wide sections using the methodology described in ACI 349 Section 10.2 (Reference 3B-3). When the axial loads include collector forces, the usable compression strain is limited to 0.002, as explained in Section 3B.1.4.3.17.

3B.1.4.3.15 In-Plane Flexural Capacity of Walls

The P-M interaction diagrams are calculated for the entire wall section using the methodology described in ACI 349 Section 10 (Reference 3B-3). All provided longitudinal reinforcement in the section is included in the calculations.

From ACI 318 Section 14.3.6 (Reference 3B-6), vertical reinforcement in walls needed to resist in-plane moments due to non-seismic loads is enclosed by lateral ties if vertical reinforcement area is greater than 0.01 times gross concrete area. Ties are also used in walls where vertical reinforcement is required for axial strength (i.e., when the section fails in compression). In these situations, the requirements of Section 3B.1.4.3.8 are followed.

For walls subject to seismic loads, specific tie requirements are prescribed if the wall requires boundary elements, as shown in Section 3B.1.4.3.9.

3B.1.4.3.16 Slenderness Effects of Compression Members

Slenderness effects are evaluated for the out-of-plane direction of walls and columns with clear height to smallest horizontal dimension ratio higher than 7.3 (Section 3B.1.4.3.1). For these members, the slenderness effects

are ignored if the following equation is satisfied (ACI 349 Equation 10-7, Reference 3B-3):

$$\frac{kl_u}{r} \leq 34 - 12\left(\frac{M_1}{M_2}\right) \leq 40 \qquad \text{Eq. 3B-2}$$

Where:

k is the effective length factor obtained as described below.

r is the radius of gyration, which can be taken as 0.3h, where h is the column or wall dimension in the direction stability is being considered;

M_1 is the smaller factored end moment, taken as positive if the member is bent in single curvature, and negative if bent in double curvature;

M_2 is the larger factored end moment, taken always positive; and,

l_u is the unsupported height.

The effective length factor k can be conservatively taken equal to 1.0. More accurate values can be obtained using the Jackson and Moreland Alignment Charts for nonsway frames (Section 3B.1.4.3.1) shown in ACI 349 Figure R10.10.1.1.a (Reference 3B-3).

If Eq. 3B-2 is not satisfied, the design of the member is based on the factored forces and moments from any of the following procedures:

Nonlinear second-order analysis, satisfying ACI 349 Section 10.10.3 (Reference 3B-3);

Elastic second-order analysis, satisfying ACI 349 Section 10.10.4 (Reference 3B-3); or,

Moment magnification procedure, satisfying ACI 349 Section 10.10.5 (Reference 3B-3).

If the moment magnification procedure is used, walls (in their out-of-plane direction) and columns are designated as nonsway (Section 3B.1.4.3.1); thus, the moments can be magnified using ACI 349 Section 10.10.6 (Reference 3B-3).

3B.1.4.3.17 Diaphragm Collector Capacity

Collectors are designed considering axial (out-of-plane) flexure interaction. The P-M interaction diagram is calculated according to ACI 349 Section 10

(Reference 3B-3). Collectors are defined to have the same width as the thickness of the walls, and the same depth as the slabs. The required reinforcement is placed within this width. The usable compression strain is limited to 0.002 to avoid transverse reinforcement requirements per ACI 349 Section 21.11.7.5 (Reference 3B-3).

3B.1.4.3.18 Wall Boundary Elements

For walls subject to seismic loads, where boundary elements are required, the requirements of ACI 349 Section 21.9.6.4 (Reference 3B-3) are satisfied. Where boundary elements are not required, the requirements of ACI 349 Section 21.9.6.5 (Reference 3B-3) are satisfied.

3B.1.4.4 Basemat Foundation Design Force and Moments

The design forces and moments in RC basemat foundation are calculated using section cuts as discussed in Section 3B.1.1.

3B.1.4.5 Concrete Design for Thermal Loads

From the steady state thermal stress analysis described in Section 3.8.4, thermal loads (member demands) at section cuts across the member cross sections are obtained as described in Section 3B.1.1.

Loads resulting from thermal effects are self-relieving; i.e., thermal forces and moments are greatly reduced or completely relieved with the progress of concrete cracking and reinforcement yielding. For design, thermal out-of-plane moment and axial loads are converted to thermal strains. Concrete design is then calculated by comparing design demands from mechanical loads to “usable” strength (P-M interaction curves) which is calculated based on “usable” strains. The “usable” strains are calculated by subtracting the thermal strains from the allowable concrete and reinforcing steel strains. The procedure is explained in the following sections.

3B.1.4.5.1 Out-of-Plane Bending Strain and Axial Strain

Thermal out-of-plane bending and axial strains acting on a wall or slab section are used in calculating usable axial and out-of-plane bending strength (P-M interaction curves) as described in Section 3B.1.4.5.2. The axial and out-of-plane bending strain are calculated from the section forces as follows.

Linearized thermal bending stress, σ_{bx}, and thermal axial stress, σ_a, are calculated using equations Eq. 3B-3 and Eq. 3B-4:

$$\sigma_{bx} = \frac{M_{thx}\frac{h}{2}}{I_{xx}} \quad \text{Eq. 3B-3}$$

$$\sigma_a = \frac{P_{th}}{A} \quad \text{Eq. 3B-4}$$

where:

h = thickness of wall or slab

I_{xx} = Area moment of inertia of section cut about axis x

A = area of section cut

M_{thx} = bending moment about local section cut axis x, due to thermal effects

P_{th} = axial load in the section cut, due to thermal effects

Linearized thermal bending strain ε_{bx} and thermal axial strains ε_a are calculated using Eq. 3B-5 and Eq. 3B-6:

$$\varepsilon_{bx} = \frac{(1-v^2)\sigma_{bx}}{E} \quad \text{Eq. 3B-5}$$

$$\varepsilon_a = \frac{\sigma_a}{E} \quad \text{Eq. 3B-6}$$

where:

E =Young's modulus

v = Poisson's ratio

3B.1.4.5.2 Out-of-Plane Bending Strength and Axial Strength

Considering that increasing mechanical load results in additional cracking and yielding of reinforcement, thermal moment and thermal axial load are reduced with increasing mechanical load. The thermal moment M_{th}' and

thermal axial load P_{th}' at which the total moment M_u and total axial load P_u are equal to the design flexural strength, ΦM_n, and design axial load ΦP_n, are the thermal moment and axial load which satisfies the code strength requirement; that is:

$$M_{u_{tot}} = M_{mec} + M'_{th} \leq \Phi M_n \quad \text{Eq. 3B-7}$$

$$P_{u_{tot}} = P_{mec} + P'_{th} \leq \Phi P_n \quad \text{Eq. 3B-8}$$

Where M_{mec} and P_{mec} are the moment and axial mechanical loads (e.g., without the thermal effects)

M_{th}' and P_{th}', which are usually smaller than the out-of-plane moment and axial load calculated at a section cut (e.g., M_{thx} and P_{th}), are referred to as design thermal moment and design thermal axial load, respectively. M_{th}' and P_{th}' are calculated as follows:

- The nominal interaction curve for total strength, M_n vs P_n, is calculated as described in Section 8.12 of Reference 3B-1.
- Out-of-plane thermal moment M_{thx} and thermal axial force P_{th} acting on the section are taken from the steady state thermal stress analysis. Linearized thermal bending strain ε_{bx} and thermal axial strains ε_a are then calculated as described in Section 3B.1.4.5.1.
- At each calculated point of the nominal interaction curve, M_n vs P_n, the usable axial and bending strains are calculated by subtracting the thermal strain from the total allowable strains (for both concrete and reinforcing steel). Using the usable strains, the usable nominal interaction curve is determined, M_n' vs P_n'.
- Design thermal moment and axial load are calculated as the difference between the nominal and useable interaction curve, or:

$$M'_{th} = M_n - M'_n \quad \text{Eq. 3B-9}$$

$$P'_{th} = P_n - P'_n \quad \text{Eq. 3B-10}$$

With the calculated design thermal moment and axial load, the usable design flexural and axial strength is calculated as follows:

$$\Phi M'_n = \Phi M_n - M'_{th} \quad \text{Eq. 3B-11}$$

$$\Phi P'_n = \Phi P_n - P'_{th} \quad \text{Eq. 3B-12}$$

Finally, concrete design is conducted using mechanical moments and axial load alone and usable design flexural strength, ΦM_n, and usable design axial strength ΦP_n'.

Allowable mechanical load is not allowed to be increased due to presence of thermal load. Therefore total mechanical load must lie within both the design interaction curve (ΦM_n vs ΦP_n) and the usable design curve (ΦM_n' vs ΦP_n').

3B.1.5 Steel-Plate Composite Wall Design Methodology

Design calculations are performed to evaluate the following design conditions defined in AISC N690:

- Sxc, Sxt, Syc, and Syt: axial compression and tension checks in two orthogonal directions (x-direction corresponds to horizontal axial force acting on the vertical cross-sectional plane of walls and y-direction corresponds to vertical axial force acting on the horizontal cross-sectional plane of walls).
- Mx, My, and cMS: out-of-plane moment checks in two orthogonal directions and combined axial force and out-of-plane moment interaction checks (x-direction corresponds to out-of-plane bending about the vertical axis of walls and y-direction corresponds to out-of-plane bending about the horizontal axis of walls).
- SCxy, Vx, Vy, and cV: in-plane shear checks, out-of-plane shear checks in two orthogonal directions, and out-of-plane shear interaction checks (x-direction corresponds to out-of-plane shear on the vertical cross-sectional plane of walls and y-direction corresponds to out-of-plane shear on the horizontal cross-sectional plane of walls).

3B.1.5.1 Required Design Strength

AISC N690 provisions allow averaging the SC wall demand over panel sections to calculate the required strength for in-plane membrane forces, out-of-plane moments, and out-of-plane shear forces. The width and length of the panel sections is determined based on Section N9.2.5 of AISC N690 as follows:

- Away from openings and connection regions, the panel width and height should not exceed twice the section thickness
- In the vicinity of openings and in connection regions, the panel width and height should not exceed the section thickness

The design of the SC walls is performed using element-based demand values. This approach is conservative, as finite element models often show highly localized forces and moments that are not representative of the demand values in the members. Making use of this conservatism, the calculated DCR values over a single finite element are averaged with adjacent elements as needed per AISC N690 guidelines summarized above.

The following required strength demand values are calculated and used in the design:

- M_{rx} and M_{ry} are the required out-of-plane flexural strength per unit width in direction x and y respectively given in (kip-in/ft)
- M_{rxy} is the required twisting moment strength per unit width given in (kip-in/ft)
- S_{rx} and S_{ry} are required membrane axial strength per unit width in direction x and y given in (kip/ft)
- S_{rxy} is the required membrane in-plane shear strength per unit width (kip/ft)
- V_{rx} and V_{ry} are the required out-of-plane shear strength per unit width along edge parallel to the directions x and y respectively.

3B.1.5.2 Steel-Plate Composite Wall Design Approach

The design approach for Seismic Category I SC walls is documented in Section 6 of TR-0920-71621-P-A-R1 (Reference 3B-1).

3B.1.5.3 Steel-Plate Composite Wall to Reinforced Concrete Slab Connection Design Approach

The design approach for steel-plate composite walls to reinforced concrete slab connections is documented in Section 7 of TR-0920-71621-P-A-R1 (Reference 3B-1).

An alternate approach is presented in Section 3B.2.5. In this approach, the force demands from the RC slab to the SC wall are transferred through the connection region via a rebar connection to a development plate. The axial tension in the rebar is transferred from the RC slab to the SC wall through a load path consisting of reinforcing bar, mechanical bar couplers (CPLR), weld connection between the CPLR and development plate (DVLP PL), weld connection between DVLP PL and faceplates, and concrete bearing and breakout. The connection configuration for a single layer of rebar at the top and bottom of the interfacing RC slab is shown in Figure 3B-9. In the case where two layers of rebar are required at the top and bottom of the interfacing RC slab, the DVLP PL is sized to allow for the couplers of both layers to be welded. The required strength for the connection and components is the slab reinforcement specified tensile strength in accordance with the provisions of ACI 349 (Reference 3B-3). The rebar interfacing with the SC wall faceplate requires holes for the rebar to pass through.

3B.1.5.4 Steel-Plate Composite Walls and Thermal Effects

For accident thermal conditions, the out-of-plane bending stiffness of SC walls is reduced to account for accident temperatures (T_a) per Section N9.2 of AISC N690 and the in-plane shear stiffness is calculated using fully cracked

properties. Furthermore, the SC walls are evaluated for axial cracking due to stresses from accident temperatures, and members determined to have axial cracking are assigned axial stiffness corresponding to that provided only by the steel faceplates.

In agreement with AISC N690, all required strengths due to accident thermal loads are included in SC wall design.

3B.2 Reactor Building

3B.2.1 Design Report

3B.2.1.1 Structural Description and Geometry

The Reactor Building consists of reinforced concrete (RC) basemat and slabs and SC walls. The overall dimensions of the building are 231.5 ft, 155.5 ft, 171 ft in the east-west (X), north-south (Y), and vertical (Z) directions, respectively. Section 3.8.4 has a more detailed description of the RXB.

3B.2.1.2 Structural Material Requirements

The RXB design is based on the following material properties:

- Concrete
  - Compressive strength - 5 ksi (7 ksi for the floor slabs at EL 70 ft, 126 ft, and 146.5 ft and the roof slab).
  - Poisson's ratio - 0.17
  - Unit weight - 150 pcf
- Reinforcement
  - Yield stress - 60 ksi (ASTM A615 Grade 60 or ASTM A706 Grade 60)
  - Tensile strength - 90 ksi (A615 Grade 60), 80 ksi (A706 Grade 60)
  - Elongation - ASTMs A615 and A706
- Structural steel
  - Grade - ASTM A992 (W shapes), ASTM A500 Grade B (Tube Steel), ASTM A36 (plates)
  - Ultimate tensile strength - 65 ksi A992, 58 ksi A500 Grade B and A36
  - Yield stress - 50 ksi A992, 46 ksi A500 Grade B, 36 ksi A36
- Structural Steel used in SC Walls
  - Yield strength of faceplates and tie plates - 50 ksi
  - Ultimate strength of faceplates and tie plates - 65 ksi
  - Ultimate tensile strength of anchor connectors - 65 ksi
- Foundation soil

Section 3.8.5 and Section 3.7.1 describe soils considered in the design of the RXB.

3B.2.1.3 Structural Loads and Cracked States

The structural loads for the RXB are discussed in detail in Section 3.7.1 and Section 3.8.4 for seismic and non-seismic loads respectively.

The load combinations that are applicable to the RXB are summarized in Table 3B-6 for design of RC members and in Table 3B-7 for design of SC walls.

The cracked state of the analysis model for each individual load case is summarized in Table 3B-8 for each load combination used in design of RC members and Table 3B-9 for each load combination used in design of SC wall members.

3B.2.1.4 Structural Analysis and Design

- Design computations of critical sections

The design methodology of RXB related critical sections is discussed in Section 3B.1. Specific RXB critical sections analyzed are discussed in Section 3B.2.2 for walls, Section 3B.2.3 for slabs, Section 3B.2.4 for the NuScale Power Module bay, and Section 3B.2.5 for SC wall basemat and slab connections.

- Stability calculations

Section 3.8.5 addresses stability of the RXB.

3B.2.1.5 Summary of Results

The DCR values in Section 3B.2.2 through Section 3B.2.5 are the bounding values.

3B.2.1.6 Conclusion

The DCR values are less than or equal to 1.0; therefore, the critical sections meet the design requirements for the investigated loading conditions.

3B.2.2 Design Results - Steel-Plate Composite Walls

3B.2.2.1 Critical Sections in SC Walls

The design evaluation of SC walls is presented for the critical sections including SC walls at RX-1, RX-4, RX-B, RX-E, walls around pool area, and walls between NPMs. These SC wall members are shown in the ANSYS model of RXB in Figure 3B-10.

3B.2.2.2 Required Strengths

The required strengths for the SC wall design are conservatively calculated at each finite element (without averaging over panel sections). The calculations are performed for all active loads and load combinations discussed. As a reference, the peak contour plots of the calculated required strengths for the seismic load combination LC9_p for Baseline-Soil7 condition are shown for the critical SC wall sections in Figure 3B-11 through Figure 3B-16. In these peak contour plots, the load combination LC9 is calculated by adding the absolute maximum seismic response to static loads as follows:

$$\begin{aligned} LC9_p = D + F + 0.8L + H + (T_a + R_a + P_a) \\ + (Y_r + Y_j + Y_m) \\ + max\left|0.7E_{ss}\right| \end{aligned} \qquad \text{Eq. 3B-13}$$

and, similarly, LC9_n is calculated as follows:

$$\begin{aligned} LC9_n = D + F + 0.8L + H + (T_a + R_a + P_a) \\ + (Y_r + Y_j + Y_m) \\ - max\left|0.7E_{ss}\right| \end{aligned} \qquad \text{Eq. 3B-14}$$

In design evaluations, the full structural response time-history for all seismic input motions are used in seismic load combinations. Y_r, Y_j and Y_m loads, plus R_a, are not included in peak stress calculations.

3B.2.2.3 SC Wall Design Checks

The DCR values for each design condition are calculated at each finite element of the critical SC wall sections for all active load combinations. For the same soil type, the calculated DCR values for different design conditions are averaged over all seismic input motions for seismic load combinations and the averaged results are enveloped over all load combinations and all soil types. The resulting enveloped DCR values are shown in Figure 3B-17 through Figure 3B-22 and summarized in Table 3B-10. Figure 3B-23 shows the section view of SC walls at grid lines RX-1, RX-4, RX-B, and RX-E with typical rib tie plates and basemat connections. The DCR values summarized in Table 3B-10 are based on the results that are away from 1) joint regions where additional localized reinforcement is expected to affect the calculated DCR values and 2) locations with mathematical singularities, such as reentrant corners, where calculated stresses are unrealistically high due to limitations of finite element modeling with elastic material properties. Furthermore, DCR results are averaged, at few sub-regions, typically at few areas close to reentrant corners, among adjacent elements where one element has large DCR value while adjacent elements have lower DCR values per AISC N690 guidelines.

For each SC wall, the contour plots are provided in three rows:

- The first row shows the DCR contours for axial tension and axial compression (Sxc, Sxt, Syc, and Syt) design checks,
- The second row shows the DCR contours for out-of-plane moment (Mx and My) and combined axial and out-of-plane moment (cMS) design checks.
- The third row shows the DCR contours for in-plane shear (SCxy), out-of-plane shear (Vx and Vy), and interaction of out-of-plane shear (cV) design checks.

The maximum scale of each contour plot is set to 1.5 with red color. For completeness, the elements at the joint regions and reentrant corners around openings are included in the contour plots. As discussed earlier, the DCR values calculated at these regions are not representative of the overall structural performance of the members either they have typically larger capacities due to additional reinforcement or detailing around them (such as at the joint regions) or they are highly localized regions where numerical singularities cause large DCR values (such as at the reentrant corners).

As summarized in Table 3B-10, the DCR values calculated at SC walls are equal to or less than 1.0 and the critical sections of SC walls meet the design requirements.

3B.2.3 Design Results - Reinforced Concrete Slabs

3B.2.3.1 Critical Section in RC Slabs

The design evaluation of RC slabs is presented for the critical sections including basemat slab, floor slab at EL 100 ft and the roof slab.

3B.2.3.2 Required Strengths

The required strengths for the RC slabs are calculated at different section cuts defined on each member. The calculations are performed for all active loads and load combinations discussed. As a reference, the peak contour plots of the calculated required strengths for the seismic load combination LC6_p for Baseline-Soil7 condition are shown for all elements of the critical RC slab sections in Figure 3B-24 through Figure 3B-26. In these peak contour plots, the load combination LC6 is calculated by adding the absolute maximum seismic response to static loads as follows. These element-based contour plots are used to confirm that the location of section cuts matches with the location of the areas within the member where demand values are the largest. Based on these contour plots, as needed, additional section cuts are defined,

$$LC6_p = D + F + 0.8L + C_{cr} + H + (T_o + R_o) + max\left|E_{ss}\right| \quad \text{Eq. 3B-15}$$

and, similarly, LC6_n is calculated as follows:

$$LC6_n = D + F + 0.8L + C_{cr} + H + (T_o + R_o) - max\left|E_{ss}\right| \quad \text{Eq. 3B-16}$$

In design evaluations, the full structural response time-history for all seismic input motions are used in seismic load combinations. T_o, and R_o loads are not included in peak stress calculations.

3B.2.3.3 RC Slab Design Checks

Design calculations and evaluations of RC members are performed using demand values calculated at section cuts that are defined for critical regions on the members associated with different design conditions. The section cuts are initially defined based on the nominal locations provided in Section 8.5 of Reference 3B-1. These section cuts are then verified and expanded using the contour plots of peak demand values in each RC member for different load combinations. As an example, Figure 3B-24 through Figure 3B-26 show contour plots of peak combined demand values for the primarily controlling seismic load combination LC6 for Baseline-Soil7 condition. Review of the peak demand contour plots confirmed that the initially defined section cut locations typically match the regions with largest demand. There are some additional areas where large demand values are observed and at those areas additional section cuts are defined.

The full set of section cuts used in design calculations of the critical RC members of the RXB is shown in Figure 3B-27 through Figure 3B-29. In these figures, the section cuts labeled with a text box with white background color correspond to section cuts at nominal locations and the section cuts labeled with a text box with darker background color correspond to additional section cuts based on the review of peak demand contour plots. Table 3B-11 summarizes the properties and design parameters of each section cut.

The design calculations for out-of-plane demands (axial force - out of plane moment and axial force - out-of-plane shear) are performed by calculating the DCR value based on the required strengths and the section capacities calculated using the pre-defined reinforcement at each section (Table 3B-2). The geometry of the cross-sections in floor slabs and basemat is rectangular with equal reinforcement on both top and bottom surfaces. The roof slab is designed to be composite with W36x210 steel beams running in the north-south direction with 4 ft center-to-center spacing for out-of-plane bending demands in the north-south direction. For out-of-plane bending in the east-west direction, out-out-plane shear in both directions, and the in-plane shear in both directions only the roof slab is used to resist the demands.

The design calculations for in-plane shear demands are performed by calculating the required reinforcement to achieve a DCR value of 1.0 for the calculated required strengths at each section cut. The envelope of the

calculated in-plane shear reinforcement is added to the utilized amount of the longitudinal reinforcement defined for interaction of axial force and out-of-plane bending moment to determine the total longitudinal reinforcement at the section.

Table 3B-12 summarizes the results of design check for all section cuts for out-of-plane design conditions (PM and PV), Table 3B-13 summarizes results of design check for in-plane shear design condition.

For design of PM and PV conditions, the output provided in Table 3B-12 for each section cut includes the maximum DCR value, the governing load combination, and the governing soil type. For PV, the maximum allowed spacing, s_{max}, for shear ties is also provided. This maximum allowed spacing is used to limit the spacing that is pre-defined at each section to meet the code requirements. The reported DCR values correspond to averaged (over all seismic input motions for a given soil type) and enveloped (over all load combinations and soil type) values. The calculated DCR values are less than or equal to 1.0 for all section cuts.

For design of in-plane shear (IPS) condition, the output provided at each section cut includes the maximum DCR value, the required in-plane shear reinforcement, ro, the ratio of the required in-plane shear reinforcement to the maximum in-plane shear reinforcement, ro_max, associated with the maximum allowed in-plane shear capacity, the governing load combination, and the governing soil type. The in-plane shear reinforcement values reported under “ro” correspond to averaged (over all seismic input motions for a given soil type) and enveloped (over all load combinations and soil type) values. The IPS design calculations are performed to determine the required in-plane shear reinforcement, ro, to meet the design demands except for the two section cuts at the east and west edges of the roof slab. For cases where required in-plane shear reinforcement is calculated, the reported DCR values are typically equal to or very close to 1.0 and the ratio of ro/ro_max is typically less than 1.0. For the cases where in-plane shear reinforcement is specified, the reported ro is the smaller of the specified in-plane shear reinforcement and ro_max. For the two sections at the west and east edges of the roof slab, the in-plane shear reinforcement is defined to be the maximum in-plane shear reinforcement that is allowed by the code (ro/ro_max = 1.0). Based on the assessments, all sections within floor slab at EL 100 ft meet the in-plane shear design requirements with in-plane shear reinforcement less than the maximum allowed in-plane shear reinforcement (i.e., ro/ro_max < 1.0) and all sections within the roof slab meet the in-plane shear design requirements (DCRIP<1.0) for the specified in-plane shear reinforcement.

The additional in-plane shear reinforcement area, $A_{s_add_IP}$, at each section is calculated by subtracting the unutilized portion of the longitudinal reinforcement defined for axial forces and out-of-plane bending demands from the longitudinal reinforcement required for in-plane demands as follows:

$$A_{s_add_IP} = (A_{s_IP}/2) - (A_{s_{PM}})[1.0 - ave(DCR_{PM})] > 0.0 \qquad \text{Eq. 3B-17}$$

where, A_{s_IP} = ro[(t)(s)], t is the thickness of the section, s is the spacing of the reinforcement, A_{s_PM} is the longitudinal reinforcement at top or bottom of the section defined for axial force and out-of-plane bending (PM) demands, and ave (DCR_{PM}) is the average of the maximum demand-to-capacity ratios due to out-of-plane PM design condition from sections that are perpendicular to the direction of the in-plane shear demand (i.e., for in-plane shear demand in the north-south direction, the DCR_{PM} results from all sections running in the east-west direction are considered in the same member).

The additional required in-plane shear reinforcement areas are added to the longitudinal reinforcement defined for out-of-plane PM design conditions to determine the total longitudinal reinforcement at each member. A final set of design properties of critical sections of RC members is presented in Table 3B-14 together with the governing design condition and maximum DCR value. Figure 3B-30 through Figure 3B-31a show the layout of reinforcement within each critical section. The section views shown in Figure 3B-32 present the typical reinforcement layout in the basemat, the floor slab at EL 100 ft, and the roof slab. Based on these calculations and results, the critical sections of RC members meet the design requirements.

3B.2.4 NuScale Power Module Bay

3B.2.4.1 NuScale Power Module Skirt Support

A skirt restraint provides lateral support at the base of the NuScale Power Module (NPM). The NPM skirt restraint consists of four built-up stainless steel members bracing the NPM skirt in the lateral directions and a stainless steel annular bearing plate that supports the NPM in the vertical direction. Figure 3B-33 details a plan view of the typical NPM bay that shows the outline of the NPM skirt, the annular bearing plate, and lateral skirt restraints, all shown with pertinent geometry for the evaluation of these components. Figure 3B-34 shows the NPM section view at skirt support and Figure 3B-35 shows the elevation view of lower NPM bay.

The skirt has an outer diameter of approximately 141 inches, a width of 6 inches to be used when considering bearing loads, and a thickness of 4.5 inches, which transfers load into the adjacent braces (Figure 3B-33). The annular bearing plate has an inner diameter of 122 inches, a width of 12 inches, and thickness of 2 inches (Figure 3B-34).

The evaluation of the NPM skirt restraint is split into vertical analysis and lateral analysis. The vertical analysis evaluates the annular bearing plate for the vertical bearing loads. The lateral analysis evaluates the lateral braces based on the load path, starting with combined axial and bending of the braces, local bearing on the braces, evaluation of the braces connection to the SC walls, and local evaluation of the SC wall.

Loads and Load Combinations:

The NPM seismic skirt restraint is evaluated for the loading condition of a seismic event. The NPM seismic skirt bears on the skirt restraint horizontally, causing the skirt restraint to experience axial and bending forces. The NPM skirt also bears vertically on the steel plate underneath. The SC wall that has the weld plate that the brace is welded to also experience forces, all of which are evaluated.

The temperature effects on the skirt restraint and associated structural components are also considered. Per Section NB3.1 of AISC N690-18 (Reference 3B-4), a decrease should be taken into account in determining the design strength of the structural component that is exposed to sustained temperatures in excess of 250 degrees F. Due to a maximum analysis temperature of 212 degrees F, the material properties of the steel are unreduced. Since thermal expansion of these components is not restrained, there is no thermal load development. The annular bearing plate experiences internal stress development from thermal expansion, but plate buckling is resisted by the weight of the NPM.

The NPM skirt restraint is evaluated for the following load combinations:

Loading stage:	1.6L	(NB2-2)
Operational stage:	1.4D	(NB2-1)
	$D + E_{SS}$	(NB2-6)

Where,

D = dead load (self-weight of NPM)

E_{SS} = SSE seismic load

L = live load (NPM placement)

Load combination with SSE effects, NB2-6 governs the design.

Acceptance Criteria and Evaluation:

Seismic Category I structural steel and SC wall components are designed to AISC N690-18 and Seismic Category I concrete components are designed to ACI 349-13 (Reference 3B-3).

The vertical loads are resisted by the annual bearing plates, which are supported by the concrete basemat. Therefore, the annual bearing plate and concrete beneath are evaluated for the bearing load only. The lateral analysis of the NPM skirt restraint evaluates the lateral braces based on the load paths. Seismic loads are combined utilizing the 100-40-40 method. Table 3B-15

provides the DCR for the evaluated components and failure modes of the NPM skirt restraint.

3B.2.4.2 NuScale Power Module Lug Restraint

Lug restraints extending from the bay walls prevent the lateral movement of the NPM. To restrain the NPM in the lateral direction, two lug restraints are positioned on opposite sides of the NPM lug. To prevent movement, a wedge-jack system is attached to the lug restraints and has plates that are able to extend and retract. When the wedge-jack is extended, the NPM is restrained. When in the retracted position, the wedge-jack is considered completely flat and there is a gap to allow the NPM to be reinstalled by the Reactor Building crane (RBC). Three NPM lug restraint configurations are considered in the evaluation. These configurations are typical lug, end wall lug, and end bay lug as shown in Figure 3B-36. Figure 3B-37 shows the components of the typical lug restraint.

Exterior to the SC wall, the faceplate, restraint plates, intermediate restraint plates, stiffener plates, end-bearing plates, end-bearing stiffening plates, end-bearing intermediate built-up, and cover plates use stainless steel. Interior to the SC wall, the embedded plates use carbon steel with minimum yield strength of 65 ksi. Embedded shear stud is the Nelson headed stud conforming to AWS D1.1 Type B.

Loads and Load Combinations:

The NPM seismic lug restraint and associated components are evaluated for circumferential, radial, and vertical design seismic force from NPM lug. Note that the boundary conditions of the NPM and NPM lug restraint are such that the NPM lug restraint is only subject to seismic loads and no other loads. The design seismic loads are:

Circumferential:	1,500 kips
Radial:	+/- 112.5 kips
Vertical:	+/- 112.5 kips

Temperature effects on the lug restraint material strength and stiffness for the accident temperature is considered. The lug restraint steel components are allowed to expand along their surrounding parts. Therefore, thermal analysis of the lug restraints is not performed. At accident temperature, the interior portion of SC wall is evaluated at peak passive cooling temperature (227 degrees F). The AISC N690-18 Section NB3.3 states reduction in material properties should be investigated in temperature of excess of 250 degrees F. Therefore, no reduction in material properties is taken for portions interior to the SC wall.

The LRFD load combinations from AISC N690-18 Section NB2 are used for evaluation of the NPM seismic lug restraint components. Only dead load and

seismic load at accident temperature are applicable. The controlling load combination for the design condition is given below.

$D + E_{SS}$ (NB2-6)

Where,

D = dead load due to weight of the structural elements

E_{SS} = SSE seismic load

Acceptance Criteria and Evaluations:

The NPM lug restraint is a Seismic Category I component. Structural steel components are designed to the provisions of AISC N690-18 (Reference 3B-4). The concrete components are designed to the provisions of ACI 349-13 (Reference 3B-3).

The components in the NPM lug restraint are evaluated based on the load path. The first component evaluated is the welds connecting the wedge-jack to the lug restraint. The NPM shim bears uniformly on the wedge-jack at the center of the wedge-jack plate. The load path into the lug restraint considers the wedge-jack plate to be uniformly loaded. Following the connection to the lug restraint, the restraint stiffener plates and welded connection between the lug restraint and SC wall faceplate are evaluated. Therefore, the load is taken in the interior of the SC wall where the lug restraint anchorage to the SC wall concrete is evaluated. Within the SC wall, force transfers to the embedded shear studs. In addition to evaluation of force transfer to the embedded shear studs, the SC wall concrete and steel section components are evaluated. Table 3B-16 provides the DCR for the evaluated components and failure modes of the NPM lug restraint.

3B.2.4.3 Reactor Building Crane Corbel

The RBC corbel consists of two continuous stiffened ledges that support the RBC in both the horizontal and vertical directions. The two continuous stiffened ledges are at two locations in the RXB at EL 145 ft 6 inches, as shown in Figure 3B-38 and Figure 3B-39. These ledges are designed as stiffened seats with the RBC acting as a series of moving point loads along the crane rail on the top of the stiffened ledges.

The evaluation of the RBC corbel is split into three sections: downward load analysis, upward load analysis, and lateral analysis. The downward load analysis evaluates the RBC corbel based on the load path, starting with the combined axial and bending of the stiffeners, shear of the stiffeners, local buckling of the stiffeners, evaluation of the plate connections between each other and to the SC walls, and local evaluation of the SC wall. The upward load analysis evaluates the RBC corbel based on the load path, starting with shear of the stiffeners, evaluation of the plate connections between each other

and to the SC walls, and local evaluation of the SC wall. The lateral analysis evaluates the lateral braces based on the load path, starting with local buckling of the stiffeners, evaluation of the stiffener connections to the SC walls, and local evaluation of the SC wall.

Loads and Load Combinations:

The LRFD load combinations from AISC N690-18 (Reference 3B-4) Section NB2 are used for evaluation of the RBC corbel components. Only dead load, crane load, and seismic load are applicable. The controlling load combination for the design condition is given below.

$D + C + E_{SS}$ (NB2-6, Reference 3B-4)

where,

D = Dead load (self-weight of RBC corbel)

C = Crane load

E_{SS} = SSE seismic load

Load combination NB2-6 governs the design for the operation stage since it considers the SSE load that envelops all the N690 load combinations. This load combination corresponds to the extreme environmental load combinations.

The temperature effects on the RBC corbel and associated structural components are considered in RBC corbel evaluation. Section NB3.1 of N690-18 (Reference 3B-4) states that design strength of the structural component that is exposed to sustained temperature in excess of 250 degrees F should be reduced. Since the maximum analysis temperature is 227 degrees F, the material strengths are not decreased. The RBC corbel is structurally attached to the SC wall and expands with the wall during thermal expansion. The stresses are self-relieved through this mechanism. Thus, there is no thermal stress in the RBC corbel.

The RBC corbel components exterior to the SC wall module are made from ASTM A572 Grade 55 material.

Connection of carbon or low alloy steel uses 80 ksi weld filler material to itself and other carbon or low alloy steel. Other connections use 70 ksi weld filler metal.

Table 3B-17 provides the DCR and corresponding failure modes for the RBC corbel components.

3B.2.5 Steel-Plate Composite Wall to Reinforced Concrete Slab Connection

This section evaluates the concrete slab to external SC wall connection anchored by development plates (DVLP PL) welded to the faceplate (FP).

The scope of this evaluation is limited to the mechanical reinforcing bar coupler (CPLR), weld connection between CPLR and DVLP PL, DVLP PL concrete bearing, welded connection between DVLP PL and FP, and concrete breakout. Figure 3B-9 depicts components that are in scope.

3B.2.5.1 Design Requirements

A rigid connection is considered since reinforcing bars are provided in both top and bottom of RC slabs. In agreement with Section 12.6.4 of ACI 349 (Reference 3B-3), the required strength for the connection and components is the slab reinforcement's specified tensile strength.

The components of the DVLP PL anchor connection (CPLR, CPLR weld, DVLP, DVLP weld) are designed according to AISC N690-18 (Reference 3B-4) and AISC 360-16 (Reference 3B-5), and the reinforced concrete is designed according to ACI 349-3 (Reference 3B-3). All structural components are evaluated only for Load Resistance Factor Design (LRFD) load combinations LRFD strength capacities.

3B.2.5.2 Structural Material Requirements

The connection design is based on the following material properties:

- Concrete
  - Compressive strength - 5 ksi
- Reinforcement
  - Yield stress - 60 ksi (ASTM A615 Grade 60)
  - Tensile strength - 90 ksi (A615 Grade 60)
- Structural steel
  - Yield strength of faceplates and development plate - 50 ksi
  - Ultimate strength of faceplates and development plate - 65 ksi
- Welds
  - Weld filler material strength for E70 electrode - 70 ksi
- Couplers
  - *nVent LENTON C2/C3J Weldable Half-Coupler for Nuclear Applications* or similar
  - Specified outer dimension - no greater than 1.56 in. (#8 / #9 reinforcement) or 2 in. (#10 / #11 reinforcement)

3B.2.5.3 Mechanical Reinforcing Bar Coupler Evaluation

The CPLR is used to anchor the slab reinforcement to the DVLP PL that is welded to the opposing interior face of the SC wall module. The CPLR is designed as a Type 2 mechanical splice under ACI 349-13.

The generic dimensional specifications of the CPLR used in this connection are shown in Figure 3B-40 and Table 3B-18. These specifications bound the nVent LENTON C2/C3J Weldable Half-Coupler for Nuclear Applications. No further evaluation is required.

3B.2.5.4 Welded Mechanical Reinforcing Bar Coupler to Development Plate Evaluation

The welded connection between the CPLR and DVLP PL is designed to resist the slab reinforcement specified tensile strength in accordance with Section 12.6.4 of ACI 349 (Reference 3B-3). The connection is comprised of an all-round partial-joint-penetration (PJP) groove weld with a reinforcing all-around fillet weld (Figure 3B-9). The groove weld size ($\frac{\phi_1}{2} - \frac{\phi_2}{2}$) is determined by the specification of the mechanical coupler shown in Table 3B-18 and Figure 3B-40. The reinforcing filled weld size is chosen so that the combined weld strength is lower than the required strength.

3B.2.5.5 Development Plate Evaluation

The DVLP PL is used to anchor the CPLR into the SC wall by bearing on the concrete inside the SC wall. The design of the DVLP PL is evaluated in shear and bearing to resist the slab reinforcement's specified tensile strength in accordance with ACI 349, Section 12.6.4 (Reference 3B-3).

The DVLP PL are evaluated for anchoring a single layer of CPLR and a double layer as shown in Figure 3B-41. The generic properties are shown in Table 3B-19 with the tributary width of the DVLP PL (W_{DP}) calculated based on the geometry of the DVLP PL, CPLR, and weld.

3B.2.5.6 Welded Development Plate to Faceplate Evaluation

The welded connection between the DVLP PL to FP is not a structural element as its only purpose is to hold the DVLP PL in place during fabrication, transportation, and fit-up. The tensile load of the reinforcement bar is transferred through the DVLP PL bearing on the concrete inside the SC wall; therefore, no further evaluation is required.

3B.2.5.7 Concrete Breakout Evaluation

The tensile load of the reinforcement is transferred into the SC wall through the bearing of the DVLP PL on the concrete. Concrete breakout is restrained

by the opposite FP of the DVLP PL, and therefore, no further evaluation is required.

3B.2.5.8 Reinforced Concrete Slab to Steel-Plate Composite Wall Connection Results

The controlling failure mode is determined to be the CPLR weld to DVLP PL. This weld is evaluated for a single layer reinforcement as each welded coupler connection to development plate is evaluated individually.

The required strength for a single layer of slab reinforcement is shown in Table 3B-20. The total effective throat dimension of the combined PJP groove weld and fillet weld is also shown in Table 3B-20. This dimension is taken as the minimum dimension from the PJP root to the face of the fillet weld or:

$D_{\text{fillet Root}} + DP_{\text{JP Root}}$

The total weld strength calculated based on AISC 360, Section J, is shown in Table 3B-20 together with the resulting DCR.

The resulting DCRs are equal to or less than 1.0, which verifies the adequacy of this connection for the specified reinforcing bars.

3B.3 Control Building

3B.3.1 Design Report

3B.3.1.1 Structural Description and Geometry

The CRB is a surface-founded reinforced concrete structure. The CRB has two portions - an SC-I portion and a non-SC-I portion. This section addresses the design of the SC-I portion of the CRB. Section 3.8.4 describes the CRB. The CRB geometry and floor layout are shown in Figure 1.2-18 through Figure 1.2-21.

3B.3.1.2 Structural Material Requirements

The CRB design is based on the following material properties:

- Concrete
  - Compressive strength - 5 ksi
  - Modulus of elasticity - 4,031 ksi
  - Shear modulus - 1,722 ksi
  - Poisson's ratio - 0.17
- Reinforcement
  - Yield stress - 60 ksi (ASTM A615 Grade 60 or ASTM A706 Grade 60)
  - Tensile strength - 90 ksi (A615 Grade 60), 80 ksi (A706 Grade 60)

  - Elongation - ASTMs A615 and A706
- Structural steel
  - Grade - ASTM A992 (W shapes), ASTM A500 Grade B (tube steel), ASTM A36 (plates)
  - Ultimate tensile strength - 65 ksi A992, 58 ksi A500 Grade B and A36
  - Yield stress - 50 ksi A992, 46 ksi A500 Grade B, 36 ksi A36
- Foundation soil

Section 3.7.1 and Section 3.8.5 describe soils considered in design of the CRB.

3B.3.1.3 Critical Sections

For design, 10 regions are identified to represent the wall, slab and basemat critical sections. Each of these sections is defined as an ANSYS component in the finite element model. Table 3B-3 lists the regions and FEA components.

3B.3.1.4 Structural Loads

The structural loads for the CRB are discussed in Section 3.7.1 and Section 3.8.4 for seismic and non-seismic loads respectively.

The load combinations that are applicable to the CRB are summarized in Table 3B-21.

3B.3.1.5 Structural Analysis and Design

- Design Computations of Critical Elements

The design methodology of CRB related critical elements is discussed in Section 3B.1. Specific CRB critical elements analyzed are discussed in Section 3B.3.2 for walls and Section 3B.3.3 for slabs.

- Stability Calculations

Stability of the CRB is addressed in Section 3.8.5.

3B.3.1.6 Summary of Results

Sample contour plots showing maximum response locations are shown for each structural element in Figure 3B-41a through Figure 3B-41e. Table 3B-22 lists final design properties of the critical regions of the CRB.

Table 3B-23 provides the maximum DCR for each design check region for out-of-plane flexure, out-of-plane shear, and in-plane shear. The table also lists the load case number resulting in the maximum DCR as well as the soil type when applicable (soil type is not critical for static load case; consequently, the soil type is not listed for static load cases in the table). As shown in

Table 3B-23, all design check regions have no demand-to-capacity exceedances. In these tables the PMop refers to out-of-Plane flexural, PVop for out-of-plane shear, and Vip refers to In-Plane shear.

3B.3.1.7 Conclusions

The DCRs ratios presented are all less than or equal to 1.0. Therefore, the critical elements satisfy the design criteria for the loading investigated.

3B.3.2 Design Results - Walls

3B.3.2.1 Wall at Grid Line 3

The wall at grid line 3 is an interior structural wall running in the north-south direction between the basemat at EL. 100 ft and the slab at EL. 123 ft (region CRB_3_100) and between the slab at EL. 123 ft and the roof at EL. 150 ft 3 in. (region CRB_3_123). This wall is 3 feet thick. The ANSYS model elevation view is shown in Figure 3B-41f and Figure 3B-42.

The maximum DCRs and controlling load combination for the section cuts in region CRB_3_100 are shown in Table 3B-24 for out-of-plane flexure, out-of-plane shear, and in-plane shear. The maximum DCRs and controlling load combination for the section cuts in region CRB_3_123 are shown in Table 3B-25 for out-of-plane flexure, out-of-plane shear, and in-plane shear.

Based on these results and evaluations, the wall is acceptable.

3B.3.2.2 Wall at Grid Line 5

The wall at grid line 5 is the east exterior structural wall between the basemat at EL. 100 ft and the slab at EL. 123 ft (region CRB_5_100) and between the slab at EL. 123 ft and the roof at EL. 150 ft 3 in. (region CRB_5_123). This wall is 3 feet thick. The ANSYS model elevation view is shown in Figure 3B-43 and Figure 3B-44.

The maximum DCRs and controlling load combination for the section cuts in region CRB_5_100 are shown in Table 3B-26 for out-of-plane flexure, out-of-plane shear, and in-plane shear. The maximum DCRs and controlling load combination for the section cuts in region CRB_5_123 are shown in Table 3B-27 for out-of-plane flexure, out-of-plane shear, and in-plane shear.

Based on these results and evaluations, the wall is acceptable.

3B.3.2.3 Wall at Grid Line H

The wall at grid line H is the south exterior structural wall between the basemat at EL. 100 ft and the slab at EL. 123 ft (region CRB_H_100) and between the slab at EL. 123 ft and the roof at EL. 150 ft 3 in. (region CRB_H_123). This wall is 3 feet thick. The ANSYS model elevation view is shown in Figure 3B-45 and Figure 3B-46.

The maximum demand-to-capacity ratios and controlling load combination for the section cuts in region CRB_H_100 are shown in Table 3B-28 for out-of-plane flexure, out-of-plane shear, and in-plane shear, respectively. The maximum demand-to-capacity ratios and controlling load combination for the section cuts in region CRB_H_123 are shown in Table 3B-29 for out-of-plane flexure, out-of-plane shear, and in-plane shear.

Based on these results and evaluations, the wall is acceptable.

3B.3.3 Design Results - Slabs

3B.3.3.1 Basemat Foundation

The basemat is at grade level EL. 100 ft (region CRB_Basemat_100). The basemat is 5 feet thick. The ANSYS model plan view is shown in Figure 3B-47.

The maximum DCRs and controlling load combination for the section cuts in region CRB_Basemat_100 are shown in Table 3B-30 for out-of-plane flexure, out-of-plane shear, and in-plane shear.

Based on these results and evaluations, the basemat is acceptable.

3B.3.3.2 Slab Elevation 123 Feet

The slab at EL. 123 ft (regions CRB_Slab_123_1, CRB_Slab_123_2, and CRB_Slab_123_3) is 2 feet thick. The ANSYS model plan views are shown in Figure 3B-48 through Figure 3B-50.

The maximum DCRs and controlling load combination for the section cuts in region CRB_Slab_123_1 are shown in Table 3B-31 for out-of-plane flexure, out-of-plane shear, and in-plane shear. The maximum DCRs and controlling load combination for the section cuts in region CRB_Slab_123_2 are shown in Table 3B-32 for out-of-plane flexure, out-of-plane shear, and in-plane shear. The maximum DCRs and controlling load combination for the section cuts in region CRB_Slab_123_3 are shown in Table 3B-33 for out-of-plane flexure, out-of-plane shear, and in-plane shear.

Based on these results and evaluations, the slab is acceptable.

3B.4 References

3B-1 TR-0920-71621-P-A-R1, “Building Design and Analysis Methodology for Safety-Related Structures.”

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3B-4 American National Standards Institute/American Institute of Steel Construction, "Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities," ANSI/AISC N690-18, Chicago, IL.

3B-5 American National Standards Institute/American Institute of Steel Construction, "Specification for Structural Steel Buildings," ANSI/AISC 360-16, Chicago, IL.

3B-6 American Concrete Institute, "Building Code Requirements for Structural Concrete and Commentary," ACI 318-08, Farmington Hills, MI.

3B-7 MacGregor, J., and Wight, J., Reinforced Concrete Mechanics and Design. Prentice Hall, New Jersey, 2005.

3B-8 National Institute Standards Technology, “Seismic design of cast-in-place concrete diaphragms, chords, and collectors: A guide for practicing engineers,” Second Edition, GCR 16-917-42, NEHRP Seismic Design Technical Brief No. 3, produced by the Applied Technology Council for the National Institute of Standards and Technology, Gaithersburg, MD, October 2016.

3B-9 Ozkan, M. K., Ozgur, C., Ulku, E., and Hays, J., “Design of Nuclear Power Plant Concrete/Composite Structures Using Finite Element Analysis Results,” Transactions of the International Association for Structural Mechanics in Reactor Technology - SMiRT 23, Division IV: Paper ID# 430, Manchester, UK, 2015.

3B-10 Pacoste, C., Plos, M., Johanson, M., “Recommendations for finite element analysis for the design or reinforced concrete slabs,” Stockholm, Sweden, 2012.

3B-11 Moehle, Jack P., Ghodsi, Tony, Hooper, John D., Fields, David C., and Gedhada, Rajnikanth (2011), “Seismic design of cast-in-place concrete special structural walls and coupling beams: A guide for practicing engineers,” NEHRP Seismic Design Technical Brief No. 6, produced by the NEHRP Consultants Joint Venture, a partnership of the Applied Technology Council and the Consortium of Universities for Research in Earthquake Engineering, for the National Institute of Standards and Technology, Gaithersburgh, MD, NIST GCR 11-917-11REV-1.

Table 3B-1: Initial Design Properties of Critical SC Wall Sections of RXB

SC Wall	Total Wall Thickness (in.)	Faceplate Thickness (in.)	Anchor Diameter (in.)	Anchor Spacing[(a)] (in.)	Tie Plate Area (in^2)	Tie Plate Spacing[(a)] (in.)
RX-1	60	0.75	1.0	8.0	6.0	30.0
RX-4	48	0.75	1.0	8.0	6.0	24.0
RX-B_1	48	0.75	1.0	8.0	6.0	24.0
RX-B_2	78	0.75	1.0	8.0	6.0	39.0
RX-B_3	60	0.75	1.0	8.0	6.0	30.0
RX-E	48	0.75	1.0	8.0	6.0	24.0
Pool Walls	48	0.75	1.0	8.0	6.0	24.0
Walls between NPMs[(b)]	48	0.75	1.0	8.0	6.0	24.0

Notes:

a)Same spacing is used for horizontal and vertical directions

b)These walls are located on gridlines RX-4.3 and RX-4.6 and in output contour plots the labels Wall RX-4.3 and Wall RX-4.6 are used.

Table 3B-2: Initial Design Properties of Critical RC Slab Sections of RXB

RC Member	Sub-Region	Section Thickness (in.)	Clear Cover (in.)	Out-of-plane Bending Reinforcement (EFEW)	Out-of-plane Shear Reinforcement (EW)
Basemat	All	96[a]	3.0	2#11@12”	#4@12”
Floor EL 100’	All except Notes [b], [c], [d]	24	2.0	#9@12”	#4@12”
Floor EL 100’	Note [b] and [c]	24	2.0	#11@9”	#4@12”
Floor EL 100’	Note [c]	24	2.0	2#11@12”	#6@5”
Floor EL 100’	Note [d]	24	2.0	#9@9”	#4@12”
Floor EL 100’	Note [e]	36	2.0	2#11@9”	#4@12”
Roof	All except Note [f]	36	2.0	2#11@12”[f]	#4@12”

Notes:

a) The thickness of basemat is 96 in. (8 ft) except at the region within the pool area where an additional 12 in. layer is provided for non-structural reasons. This additional thickness is conservatively not considered in calculating section capacities for design evaluations.

b) The longitudinal reinforcement of #11@9" is provided in the north-south direction of slab regions between Gridlines RX-2 and RX-5 and the 24 in.-thick slab region surrounded by Gridlines RX-1, RX2.2, RX-B, and RX-D. At the same slab regions, the longitudinal reinforcement in the east-west direction is defined as #9@12" except at 120 in. (10 ft) wide strip of the slab region at the north side of Gridline RX-B and at the south of Gridline RX-D and between Gridlines RX-4 and RX-5 where #11@12 reinforcement is provided.

c) The longitudinal reinforcement of 2#11@12" is provided in the north-south direction at the narrow slab section between the pipe chase openings along Gridlines RX-B and RX-D. At the same location, the longitudinal reinforcement in the east-west direction is defined as #11@12". Similarly, the out-of-plane shear reinforcement of #6@5" is provided in the north-south direction and the out-of-plane shear reinforcement in the east-west direction is defined as #6@12.

d) The longitudinal reinforcement of #9@9" is provided in the east-west direction at the slab section between Gridlines RX-5 and RX-6 and Gridlines RX-B and RX-D. At the same location, the longitudinal reinforcement in the north-south direction is defined as #9@12".

e) The section of floor slab at EL 100 ft between Gridlines RX-C and RX-D and RX-2 and RX-2.4 is 36 in. (3 ft) thick and at this section the reinforcement is defined as 2#11@9" each-face-each-way (EFEW).

f) The typical reinforcement defined in the roof slab is 2#11@12" EFEW. The roof slab is designed as a composite section with W36x210 steel beams running in the north-south direction with 4 ft center-to-center spacing in the east-west direction.

Table 3B-3: Critical Wall and Slab Sections of CRB

Region	FEA Component	Description
CRB_3_100	RC_Wall_3_100	Interior Wall at Gridline 3, EL. 100’ to EL. 123’
CRB_3_123	RC_Wall_3_123	Interior Wall at Gridline 3, EL. 123’ to roof
CRB_5_100	RC_Wall_5_100	Exterior Wall at Gridline 5, EL. 100’ to EL. 123’
CRB_5_123	RC_Wall_5_123	Exterior Wall at Gridline 5, EL. 123’ to roof
CRB_H_100	RC_Wall_H_100	Exterior Wall at Gridline H, EL. 100’ to EL. 123’
CRB_H_123	RC_Wall_H_123	Exterior Wall at Gridline H, EL. 123’ to roof
CRB_Basemat_100	RC_Basemat_100	Basemat at EL. 100’
CRB_123_1	RC_Slab_123_1	Slab at EL. 123’, between gridlines 1-3-F-H
CRB_123_2	RC_Slab_123_2	Slab at EL. 123’, between gridlines 3-5-G-H
CRB_123_3	RC_Slab_123_3	Slab at EL. 123’, between gridlines 3-5-F-G

Table 3B-4: Critical Wall and Slab Sections of the CRB

Region	Section Thickness (in.)	Clear Cover (in.)
CRB_3_100	36	2.0
CRB_3_123	36	2.0
CRB_5_100	36	2.0
CRB_5_123	36	2.0
CRB_H_100	36	2.0
CRB_H_123	36	2.0
CRB_slab_123_1	24	2.0
CRB_slab_123_2	24	2.0
CRB_slab_123_3	24	2.0
CRB_Basemat_100	60	3.0

Table 3B-5: In-Layer Input Motion Data Sets used in Seismic Response Calculations

Soil Type	Direction		
	East-West	North-South	Vertical
Capitola: In-Layer Input Motion Data Set 1			
7	CP07E_L15_INLAY.dtn	CP07N_L15_INLAY.dtn	CP07V_L15_INLAY.dtn
11	CP11E_L15_INLAY.dtn	CP11N_L15_INLAY.dtn	CP11V_L15_INLAY.dtn
Chi-Chi: In-Layer Input Motion Data Set 2			
7	CC07E_L15_INLAY.dtn	CC07N_L15_INLAY.dtn	CC07V_L15_INLAY.dtn
11	CC11E_L15_INLAY.dtn	CC11N_L15_INLAY.dtn	CC11V_L15_INLAY.dtn
El Centro: In-Layer Input Motion Data Set 3			
7	EC07E_L15_INLAY.dtn	EC07N_L15_INLAY.dtn	EC07V_L15_INLAY.dtn
11	EC11E_L15_INLAY.dtn	EC11N_L15_INLAY.dtn	EC11V_L15_INLAY.dtn
Izmit: In-Layer Input Motion Data Set 4			
7	IZ07E_L15_INLAY.dtn	IZ07N_L15_INLAY.dtn	IZ07V_L15_INLAY.dtn
11	IZ11E_L15_INLAY.dtn	IZ11N_L15_INLAY.dtn	IZ11V_L15_INLAY.dtn
Yermo: In-Layer Input Motion Data Set 5			
7	YM07E_L15_INLAY.dtn	YM07N_L15_INLAY.dtn	YM07V_L15_INLAY.dtn
11	YM11E_L15_INLAY.dtn	YM11N_L15_INLAY.dtn	YM11V_L15_INLAY.dtn
Lucerne: In-Layer Input Motion Data Set 6			
9	LU09E_L15_INLAY.dtn	LU09N_L15_INLAY.dtn	LU09V_L15_INLAY.dtn

Table 3B-6: Load Combinations used in Design of RC Members in the RXB

Load Combination	ACI 349 Equation	Label used in the calculation
Normal Load Combinations		
$1.4(D + F + R_o) + T_o$	(9-1)	LC1
$1.2(D + F + T_o + R_o) + 1.6(L + H) + 1.4C_{cr} + 0.5(L_r \text{ or } S \text{ or } R)$	(9-2)	LC2
$1.2(D+ F + R_o) + 0.8(L + H) + 1.4C_{cr} + 1.6(L_r \text{ or } S \text{ or } R)$	(9-3)	LC3
Severe Environmental Load Combinations		
$1.2(D + F + R_o) + 1.6(L + H + E_o)$	(9-4)	LC4
$1.2(D + F + R_o) + 1.6(L + H + W)$	(9-5)	LC5
Extreme Environmental Load Combinations		
$D + F + 0.8L + C_{cr} + H + T_o + R_o + E_{ss}$	(9-6)	LC6
$D + F + 0.8L + H + T_o + R_o + (W_t \text{ or } W_h)$	(9-7)	LC7
$D + F + 0.8L + H + T_o + R_o + S_e$	(note [a])	
Abnormal Load Combinations		
$D + F + 0.8L + C_{cr} + H + (T_a + R_a + 1.2P_a)$	(9-8)	LC8
$D + F + 0.8L + H + (T_a + R_a + P_a) + (Y_r + Y_j + Y_m) + E_{ss}$	(9-9)	LC9
Reduced Load Combinations [b]		
$1.2(0.9D + F + T_o + R_o) + 1.6(L + H) + 1.4C_{cr} + 0.5S$	(9-2)	LC2b_1
$1.2(0.9D + F + T_o + R_o) + 1.6(0.9L + H) + 1.4C_{cr} + 0.5(0.9S)$	(9-2)	LC2b_2
$1.2(0.9D + F + T_o + R_o) + 1.6H + 1.4C_{cr}$	(9-2)	LC2b_3
$1.2(0.9D+ F + R_o) + 0.8(L + H) + 1.4C_{cr} + 1.6S$	(9-3)	LC3b_1
$1.2(0.9D+ F + R_o) + 0.8(0.9L + H) + 1.4C_{cr} + 1.6(0.9S)$	(9-3)	LC3b_2
$1.2(0.9D+ F + R_o) + 0.8H + 1.4C_{cr}$	(9-3)	LC3b_3
$0.9D + F + 0.8L + C_{cr} + H + T_o + R_o + E_{ss}$	(9-6)	LC6_1
$0.9D + F + 0.8(0.9L) + C_{cr} + H + T_o + R_o + E_{ss}$	(9-6)	LC6_2
$0.9D + F + C_{cr} + H + T_o + R_o + E_{ss}$	(9-6)	LC6_3
$0.9D + F + 0.8L + H + (T_a + R_a + P_a) + (Y_r + Y_j + Y_m) + E_{ss}$	(9-9)	LC9_1
$0.9D + F + 0.8(0.9L) + H + (T_a + R_a + P_a) + (Y_r + Y_j + Y_m) + E_{ss}$	(9-9)	LC9_2
$0.9D + F + H + (T_a + R_a + P_a) + (Y_r + Y_j + Y_m) + E_{ss}$	(9-9)	LC9_3
Load Combinations for Accident Pressure Cases at Galleries [c]		
$D + F + 0.8L + C_{cr} + H + 1.2P_a$	(9-8)	LC8a, LC8b, LC8c
$D + F + 0.8L + H + P_a + E_{ss}$	(9-9)	LC9a, LC9b, LC9c
$0.9D + F + 0.8L + H + P_a + E_{ss}$	(9-9)	LC9a_1, LC9b_1, LC9c_1
$0.9D + F + 0.8(0.9L) + H + P_a + E_{ss}$	(9-9)	LC9a_2, LC9b_2, LC9c_2
$0.9D + F + H + P_a + E_{ss}$	(9-9)	LC9a_3, LC9b_3, LC9c_3

Notes:

a) The load combination with extreme snow load is judged to be enveloped by the load combinations LC2 and LC3 and, hence, design calculations are not performed for this load combination.

b) These load combinations represent the version of the governing load combinations with reduced static loads to account for the adverse effects of reduced loads per Section 9.2.3 of ACI 349.

c) These load combinations are used to evaluate design of RXB floors at EL 55 ft, 70 ft, 100 ft, and 126 ft and SC walls around these floors for the localized effects of isolated accidental pressures at galleries and at module heatup system (MHS) equipment room. The subscripts "a," "b," and "c" correspond to localized accident pressure acting on different locations within the structure.

Table 3B-7: Load Combinations used in Design of SC walls in RXB

Load Combination	AISC N690 Equation	Label used in the calculation
Normal Load Combinations		
$1.4(D + F + R_o) + T_o + C_{cr}$	(NB2-1)	LC1
$1.2(D + F + T_o + R_o) + 1.6(L + H) + 1.4C_{cr} + 0.5(L_r \text{ or } S \text{ or } R)$	(NB2-2)	LC2
$1.2(D+ F + T_o + R_o) + 0.8(L + H) + 1.4C_{cr} + 1.6(L_r \text{ or } S \text{ or } R)$	(NB2-3)	LC3
Severe Environmental Load Combinations		
$1.2(D + F + R_o) + 0.8L + 1.6H + 0.2(L_r \text{ or } S \text{ or } R) + T_o + C_{cr} + 1.6E_o$	(NB2-5)	LC4
$1.2(D + F + R_o) + 0.8L + 1.6H + 0.5(L_r \text{ or } S \text{ or } R) + T_o + C_{cr} + W$	(NB2-4)	LC5
Extreme Environmental Load Combinations		
$D + F + 0.8L + H + C_{cr} + T_o + R_o + E_{ss}$	(NB2-6)	LC6
$D + F + 0.8L + H + T_o + R_o + (W_t \text{ or } W_h)$	(NB2-7)	LC7
$D + F + 0.8L + H + T_o + R_o + S_e$	(note [a])	
Abnormal Load Combinations		
$D + F + 0.8L + H + C_{cr} + T_a + R_a + 1.2P_a$	(NB2-8)	LC8
$D + F + 0.8L + H + (T_a + R_a + P_a) + (Y_r + Y_j + Y_m) + 0.7E_{ss}$	(NB2-9)	LC9
Reduced Load Combinations [b]		
$1.2(0.9D + 0.9F + T_o + R_o) + 1.6(L + H) + 1.4C_{cr} + 0.5S$	(NB2-2)	LC2b_1
$1.2(0.9D + 0.9F + T_o + R_o) + 1.4C_{cr} + 0.5S$	(NB2-2)	LC2b_2
$1.2(0.9D + 0.9F + T_o + R_o) + 1.6(L + H) + 1.4C_{cr}$	(NB2-2)	LC2b_3
$1.2(0.9D + 0.9F + T_o + R_o) + 1.4C_{cr}$	(NB2-2)	LC2b_4
$1.2(0.9D+ 0.9F + T_o + R_o) + 0.8(L + H) + 1.4C_{cr} + 1.6S$	(NB2-3)	LC3b_1
$1.2(0.9D+ 0.9F + T_o + R_o) + 1.4C_{cr} + 1.6S$	(NB2-3)	LC3b_2
$1.2(0.9D+ 0.9F + T_o + R_o) + 0.8(L + H) + 1.4C_{cr}$	(NB2-3)	LC3b_3
$1.2(0.9D+ 0.9F + T_o + R_o) + 1.4C_{cr}$	(NB2-3)	LC3b_4
$0.9D + 0.9F + 0.8L + H + C_{cr} + T_o + R_o + E_{ss}$	(NB2-6)	LC6_1
$0.9D + 0.9F + C_{cr} + T_o + R_o + E_{ss}$	(NB2-6)	LC6_2
$0.9D + 0.9F + 0.8L + H + (T_a + R_a + P_a) + (Y_r + Y_j + Y_m) + 0.7E_{ss}$	(NB2-9)	LC9_1
$0.9D + 0.9F + (T_a + R_a + P_a) + (Y_r + Y_j + Y_m) + 0.7E_{ss}$	(NB2-9)	LC9_2
Load Combinations for Accident Pressure Cases at Galleries [c]		
$D + F + 0.8L + H + C_{cr} + 1.2P_a$	(NB2-8)	LC8a, LC8b, LC8c
$D + F + 0.8L + H + P_a + 0.7E_{ss}$	(NB2-9)	LC9a, LC9b, LC9c
$0.9D + 0.9F + 0.8L + H + P_a + 0.7E_{ss}$	(NB2-9)	LC9a_1, LC9b_1, LC9c_1
$0.9D + 0.9F + P_a + 0.7E_{ss}$	(NB2-9)	LC9a_2, LC9b_2, LC9c_2

Notes:

a) The load combination with extreme snow load is judged to be enveloped by the load combinations LC2 and LC3 and, hence, design calculations are not performed for this load combination.

b) These load combinations represent the version of the governing load combinations with reduced static loads to account for the adverse effects of reduced loads per Section NB2.5d of AISC N690.

c) These load combinations are used to evaluate design of RXB floors at EL 55 ft, 70 ft, 100 ft, and 126 ft and SC Walls around these floors for the localized effects of isolated accidental pressures at galleries and at module heatup system (MHS) equipment room. The subscripts "a", "b", and "c" corresponds to localized accident pressure acting on different locations within the structure.

Table 3B-8: Cracked States used in Load Combinations for Design of RC Members in RXB

LC ID[a]	D	L	F	H	L_r	C_{cr}	S	R	T_o	R_o	W	E_o	E_{ss}	W_t	T_a	R_a	P_a	Y_r	Y_j	Y_m
LC1	UnCrk	-[b]	UnCrk	-	-	-	-	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC2b	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC3b	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	-	UnCrk	-	-	-	-	-	-	-	-	-	-
LC4	CrkEo	CrkEo	CrkEo	CrkEo	-	-	-	-	-	CrkEo	-	CrkEo	-	-	-	-	-	-	-	-
LC5	UnCrk	UnCrk	UnCrk	UnCrk	-	-	-	-	-	UnCrk	UnCrk	-	-	-	-	-	-	-	-	-
LC6	CrkEs	CrkEs	CrkEs	CrkEs	-	CrkEs	-	-	CrkEsTo	CrkEs	-	-	CrkEs	-	-	-	-	-	-	-
LC7	UnCrk	UnCrk	UnCrk	UnCrk	-	-	-	-	CrkTo	UnCrk	-	-	-	UnCrk	-	-	-	-	-	-
LC8[c]	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	-	-	-	-	-	-	-	-	CrkAllTa	CrkAllTa	CrkAllTa	-	-	-
LC9[d]	CrkAll	CrkAll	CrkAll	CrkAll	-	-	-	-	-	-	-	-	CrkAll	-	CrkAllTa	CrkAllTa	CrkAllTa	CrkAll	CrkAll	CrkAll
LC2b_1	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC2b_2	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC2b_3	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC3b_1	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	-	UnCrk	-	-	-	-	-	-	-	-	-	-
LC3b_2	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	-	UnCrk	-	-	-	-	-	-	-	-	-	-
LC3b_3	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	-	UnCrk	-	-	-	-	-	-	-	-	-	-
LC6_1	CrkEs	CrkEs	CrkEs	CrkEs	-	CrkEs	-	-	CrkEsTo	CrkEs	-	-	CrkEs	-	-	-	-	-	-	-
LC6_2	CrkEs	CrkEs	CrkEs	CrkEs	-	CrkEs	-	-	CrkEsTo	CrkEs	-	-	CrkEs	-	-	-	-	-	-	-
LC6_3	CrkEs	CrkEs	CrkEs	CrkEs	-	CrkEs	-	-	CrkEsTo	CrkEs	-	-	CrkEs	-	-	-	-	-	-	-
LC9_1[d]	CrkAll	CrkAll	CrkAll	CrkAll	-	-	-	-	-	-	-	-	CrkAll	-	CrkAllTa	CrkAllTa	CrkAllTa	CrkAll	CrkAll	CrkAll
LC9_2[d]	CrkAll	CrkAll	CrkAll	CrkAll	-	-	-	-	-	-	-	-	CrkAll	-	CrkAllTa	CrkAllTa	CrkAllTa	CrkAll	CrkAll	CrkAll
LC9_3[d]	CrkAll	CrkAll	CrkAll	CrkAll	-	-	-	-	-	-	-	-	CrkAll	-	CrkAllTa	CrkAllTa	CrkAllTa	CrkAll	CrkAll	CrkAll

Notes:

a) Load combination labels as defined in Table 3B-6.

b) The character "-" is placed under the individual load case if that load case is not present in the corresponding load combination.

c) The same cracked state is also used for load combinations LC8a, LC8b, and LC8c.

d) The same cracked state is also used for load combinations LC9a, LC9b, LC9c, LC9a_1, LC9a_2, LC9a_3, LC9b_1, LC9b_2, LC9b_3, LC9c_1, LC9c_2, and LC9c_3

Table 3B-9: Cracked States used in Load Combinations for Design of SC Walls in RXB

LC ID[a]	D	L	F	H	L_r	C_{cr}	S	R	T_o	R_o	W	E_o	E_{ss}	W_t	T_a	R_a	P_a	Y_r	Y_j	Y_m
LC1	UnCrk	_[b]	UnCrk	-	-	UnCrk	-	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC2b	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC3b	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC4	CrkEo	CrkEo	CrkEo	CrkEo	-	CrkEo	CrkEo	-	CrkEoTo	CrkEo	-	CrkEo	-	-	-	-	-	-	-	-
LC5	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	UnCrk	-	-	-	-	-	-	-	-	-
LC6	CrkEs	CrkEs	CrkEs	CrkEs	-	CrkEs	-	-	CrkEsTo	CrkEs	-	-	CrkEs	-	-	-	-	-	-	-
LC7	UnCrk	UnCrk	UnCrk	UnCrk	-	-	-	-	CrkTo	UnCrk	-	-	-	UnCrk	-	-	-	-	-	-
LC8[c]	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	-	-	-	-	-	-	-	-	CrkAllTa	CrkAllTa	CrkAllTa	-	-	-
LC9[d]	CrkAll	CrkAll	CrkAll	CrkAll	-	-	-	-	-	-	-	-	CrkAll	-	CrkAllTa	CrkAllTa	CrkAllTa	CrkAll	CrkAll	CrkAll
LC2b_1	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC2b_2	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC2b_3	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC2b_4	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC3b_1	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC3b_2	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC3b_3	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC3b_4	UnCrk	UnCrk	UnCrk	UnCrk	-	UnCrk	UnCrk	-	CrkTo	UnCrk	-	-	-	-	-	-	-	-	-	-
LC6_1	CrkEs	CrkEs	CrkEs	CrkEs	-	CrkEs	-	-	CrkEsTo	CrkEs	-	-	CrkEs	-	-	-	-	-	-	-
LC6_2	CrkEs	CrkEs	CrkEs	CrkEs	-	CrkEs	-	-	CrkEsTo	CrkEs	-	-	CrkEs	-	-	-	-	-	-	-
LC9_1[d]	CrkAll	CrkAll	CrkAll	CrkAll	-	-	-	-	-	-	-	-	CrkAll	-	CrkAllTa	CrkAllTa	CrkAllTa	CrkAll	CrkAll	CrkAll
LC9_2[d]	CrkAll	CrkAll	CrkAll	CrkAll	-	-	-	-	-	-	-	-	CrkAll	-	CrkAllTa	CrkAllTa	CrkAllTa	CrkAll	CrkAll	CrkAll

Notes:

a) Load combination labels as defined in Table 3B-7.

b) The character "-" is placed under the individual load case if that load case is not present in the corresponding load combination.

c) The same cracked state is also used for load combinations LC8a, LC8b, and LC8c.

d) The same cracked state is also used for load combinations LC9a, LC9b, LC9c, LC9a_1, LC9a_2, LC9b_1, LC9b_2, LC9c_1, and LC9c_2

Table 3B-10: Maximum DCR Values[a] (enveloped over all load combinations and soil types) at critical Sections of SC Walls in the RXB for All Design Conditions

SC Wall	Design Conditions										
	Sxc	Sxt	Syc	Syt	Mx	My	cMS	SCxy	Vx	Vy	cV
Wall RX1	0.3	0.7	0.3	0.7	0.7	0.7	1.0	0.6	0.5	0.7	0.3
Wall RX4	0.3	0.3	0.5	0.8	0.3	0.3	0.8	0.6	0.3	0.3	0.3
Wall RX4.3[b]	0.2	0.2	0.5	0.7	0.2	0.2	0.7	0.5	0.3	0.3	0.3
Wall RX4.6[b]	0.2	0.2	0.5	0.6	0.2	0.2	0.6	0.5	0.3	0.3	0.3
Wall RXB	0.3	0.7	0.3	0.5	0.4	0.7	0.8	0.4	0.5	0.5	0.3
Wall RXE	0.2	0.5	0.3	0.5	0.3	0.3	0.7	0.3	0.3	0.5	0.3
Pool Walls	0.3	0.3	0.3	0.3	0.4	0.5	0.5	0.3	0.5	0.5	0.3

Notes:

a)The DCR values are taken from areas away from joint regions and reentrant corners. Furthermore, DCR results are averaged among adjacent elements where one element has large DCR value while adjacent elements have lower DCR values.

b)SC walls between NPMs

Table 3B-11: Properties of Section Cuts Used in Design Calculations of Critical RC Members in RXB

Member	CutID	Dir_DC[a]	t (in.)	L (in.)	L/t[b]	cc[c] (in.)	OOP Bending[d]			OOP Shear Rebar[f]
							Local TopRebar[e]	Local BotRebar[e]	d (in.)	
Basemat-EL25	Cut101	EW_OOP	96	276	2.9	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut102	EW_OOP	96	287	3.0	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut103	EW_OOP	96	287	3.0	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut104	EW_OOP	96	293	3.0	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut105	EW_OOP	96	311	3.2	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut106	EW_OOP	96	287	3.0	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut107	EW_OOP	96	302	3.1	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut108	EW_OOP	96	294	3.1	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut109	EW_OOP	96	294	3.1	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut110	EW_OOP	96	294	3.1	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut111	EW_OOP	96	294	3.1	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut112	EW_OOP	96	294	3.1	3.0	2#11@12"	2#11@12"	90.9	#4@12"
	Cut126	NS_OOP	96	305	3.2	3.0	2#11@12"	2#11@12"	89.5	#4@12"
	Cut127	NS_OOP	96	286	3.0	3.0	2#11@12"	2#11@12"	89.5	#4@12"
	Cut128	NS_OOP	96	305	3.2	3.0	2#11@12"	2#11@12"	89.5	#4@12"
	Cut129	NS_OOP	96	296	3.1	3.0	2#11@12"	2#11@12"	89.5	#4@12"
	Cut130	NS_OOP	96	287	3.0	3.0	2#11@12"	2#11@12"	89.5	#4@12"
	Cut131	NS_OOP	96	305	3.2	3.0	2#11@12"	2#11@12"	89.5	#4@12"
	Cut132	NS_OOP	96	290	3.0	3.0	2#11@12"	2#11@12"	89.5	#4@12"
	Cut133	NS_OOP	96	290	3.0	3.0	2#11@12"	2#11@12"	89.5	#4@12"
Floor-EL100	Cut501	EW_OOP	24	86	3.6	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut502	EW_OOP	36	123	3.4	2.0	2#11@9"	2#11@9"	31.9	#4@12"
	Cut503	EW_OOP	36	123	3.4	2.0	2#11@9"	2#11@9"	31.9	#4@12"
	Cut504	EW_OOP	24	96	4.0	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut505	EW_OOP	24	96	4.0	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut506	EW_OOP	24	96	4.0	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut507	EW_OOP	24	102	4.2	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut508	EW_OOP	24	96	4.0	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut509	EW_OOP	24	96	4.0	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut510	EW_OOP	24	82	3.4	2.0	#11@9"	#11@9"	21.3	#4@12"

Table 3B-11: Properties of Section Cuts Used in Design Calculations of Critical RC Members in RXB (Continued)

Member	CutID	Dir_DC[a]	t (in.)	L (in.)	L/t[b]	cc[c] (in.)	OOP Bending[d]			OOP Shear Rebar[f]
							Local TopRebar[e]	Local BotRebar[e]	d (in.)	
Floor-EL100 (continued)	Cut511	EW_OOP	24	108	4.5	2.0	2#11@12"	2#11@12"	19.9	#6@5"
	Cut512	EW_OOP	24	108	4.5	2.0	2#11@12"	2#11@12"	19.9	#6@5"
	Cut513	EW_OOP	24	82	3.4	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut514	EW_OOP	24	82	3.4	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut526	NS_OOP	24	95	4.0	2.0	#9@12"	#9@12"	20.3	#4@12"
	Cut527	NS_OOP	24	73	3.0	2.0	#9@12"	#9@12"	20.3	#4@12"
	Cut528	NS_OOP	24	77	3.2	2.0	#9@12"	#9@12"	20.3	#4@12"
	Cut529	NS_OOP	24	77	3.2	2.0	#9@12"	#9@12"	20.3	#4@12"
	Cut530	NS_OOP	24	77	3.2	2.0	#9@12"	#9@12"	20.3	#4@12"
	Cut531	NS_OOP	24	96	4.0	2.0	#9@9"	#9@9"	20.3	#4@12"
	Cut532	NS_OOP	24	69	2.9	2.0	#9@9"	#9@9"	20.3	#4@12"
	Cut533	NS_OOP	24	89	3.7	2.0	#9@12"	#9@12"	20.3	#4@12"
	Cut534	NS_OOP	24	89	3.7	2.0	#9@12"	#9@12"	20.3	#4@12"
	Cut535	NS_OOP	24	83	3.5	2.0	#11@12"	#11@12"	19.9	#4@12"
	Cut551	EW_IP	24	294	12.3	2.0	#9@12"	#9@12"	20.3	#4@12"
	Cut552	EW_IP	24	186	7.8	2.0	#9@12"	#9@12"	20.3	#4@12"
	Cut553	EW_IP	24	432	18.0	2.0	#9@12"	#9@12"	20.3	#4@12"
	Cut576	NS_IP	24	420	17.5	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut577	NS_IP	24	196	8.2	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut578	NS_IP	24	420	17.5	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut579	NS_IP	24	420	17.5	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut580	NS_IP	24	420	17.5	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut581	NS_IP	24	420	17.5	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut582	NS_IP	24	420	17.5	2.0	#11@9"	#11@9"	21.3	#4@12"
	Cut583	NS_IP	24	420	17.5	2.0	#11@9"	#11@9"	21.3	#4@12"
Roof	Cut801[g]	EW_OOP	36	118	3.3	2.0	2#11@12"	2#11@12"	31.9	#4@12"
	Cut802[g]	EW_OOP	36	104	2.9	2.0	2#11@12"	2#11@12"	31.9	#4@12"
	Cut803[g]	EW_OOP	36	135	3.8	2.0	2#11@12"	2#11@12"	31.9	#4@12"
	Cut804[g]	EW_OOP	36	135	3.8	2.0	2#11@12"	2#11@12"	31.9	#4@12"

Table 3B-11: Properties of Section Cuts Used in Design Calculations of Critical RC Members in RXB (Continued)

Member	CutID	Dir_DC[a]	t (in.)	L (in.)	L/t[b]	cc[c] (in.)	OOP Bending[d]			OOP Shear Rebar[f]
							Local TopRebar[e]	Local BotRebar[e]	d (in.)	
Roof (continued)	Cut805[g]	EW_OOP	36	135	3.8	2.0	2#11@12"	2#11@12"	31.9	#4@12"
	Cut806[g]	EW_OOP	36	143	4.0	2.0	2#11@12"	2#11@12"	31.9	#4@12"
	Cut807[g]	EW_OOP	36	143	4.0	2.0	2#11@12"	2#11@12"	31.9	#4@12"
	Cut808[g]	EW_OOP	36	126	3.5	2.0	2#11@12"	2#11@12"	31.9	#4@12"
	Cut826	NS_OOP	36	129	3.6	2.0	2#11@12"	2#11@12"	30.5	#4@12"
	Cut827	NS_OOP	36	141	3.9	2.0	2#11@12"	2#11@12"	30.5	#4@12"
	Cut828	NS_OOP	36	141	3.9	2.0	2#11@12"	2#11@12"	30.5	#4@12"
	Cut829	NS_OOP	36	129	3.6	2.0	2#11@12"	2#11@12"	30.5	#4@12"
	Cut830	NS_OOP	36	158	4.4	2.0	2#11@12"	2#11@12"	30.5	#4@12"
	Cut831	NS_OOP	36	129	3.6	2.0	2#11@12"	2#11@12"	30.5	#4@12"
	Cut832	NS_OOP	36	129	3.6	2.0	2#11@12"	2#11@12"	30.5	#4@12"
	Cut876	NS_IP	36	870	24.2	2.0	2#11@12"	2#11@12"	31.9	#4@12"
	Cut877	NS_IP	36	870	24.2	2.0	2#11@12"	2#11@12"	31.9	#4@12"

Notes:

a) Direction of section cut and the type of design condition that the section cut is used. For example, EW_OOP corresponds to a section cut running in the east-west direction and it is used for design calculations for out-of-plane actions.

b) The length-to-thickness ratio of section cuts are nominally limited to 3 for out-of-plane design conditions (small deviations are due to finite element mesh size in the analysis models) and for in-plane design conditions no limit is applied as it is determined based on the full member dimension along the section cut.

c) Clear cover

d) The reinforcement for out-of-plane bending is defined such that the reinforcement running in the north-south direction is located outside of the reinforcement running in the east-west direction. The effective depth of reinforcement is measured from the geometric center of the reinforcement layers in one direction to the exterior surface of the section. The same reinforcement is used in each face.

e) The longitudinal reinforcement is defined for axial force and OOP bending design evaluations. The longitudinal reinforcement listed for section cuts defined for IP evaluations represent the longitudinal reinforcement defined in that member for OOP design evaluation and only provided for reference. The reinforcement required for in-plane demands are specifically calculated at each section and added to the utilized amount of the longitudinal reinforcement defined for axial force and OOP bending demands. The longitudinal reinforcement defined for OOP design evaluations is running perpendicular to the orientation of the section cut whereas the longitudinal reinforcement listed for IP design evaluations is running parallel to the orientation of the section cut.

f) Out-of-plane shear reinforcement is defined to be the same in both horizontal directions.

g) These sections are considered to be composite with W36x210 steel beams running in the north-south direction with 4 ft center-to-center spacing in the east-west direction.

Table 3B-12: Design Evaluation of Critical Sections of RC Members in RXB for Out-of-Plane PM and PV Design Conditions

Member	CutID	Design for PM			Design for PV				
		DCR	LC[(a)]	S_ID[(b)]	DCR	ro[(c)]	s_{mx}[(d)]	LC[(a)]	CS_ID[(b)]
Basemat-EL25	Cut101	0.28	LC1	Static	0.37	0.01667	24	LC6	Baseline_Soil11
	Cut102	0.24	LC1	Static	0.73	0.01667	24	LC5_29	Static
	Cut103	0.40	LC1	Static	0.67	0.01667	24	LC9	Baseline_Soil11
	Cut104	0.41	LC1	Static	0.33	0.01667	24	LC6	Baseline_Soil11
	Cut105	0.13	LC5_8	Static	0.67	0.01667	24	LC1	Static
	Cut106	0.40	LC1	Static	0.73	0.01667	24	LC6	Baseline_Soil11
	Cut107	0.46	LC1	Static	0.69	0.01667	24	LC6	Baseline_Soil11
	Cut108	0.13	LC5_27	Static	0.66	0.01667	24	LC9	Baseline_Soil11
	Cut109	0.52	LC1	Static	0.37	0.01667	24	LC9_3	Soil_Separation_Soil7
	Cut110	0.52	LC1	Static	0.46	0.01667	24	LC6	Baseline_Soil11
	Cut111	0.14	LC5_23	Static	0.65	0.01667	24	LC9	Baseline_Soil11
	Cut112	0.33	LC1	Static	0.72	0.01667	24	LC6	Soil_Separation_Soil7
	Cut126	0.11	LC5_23	Static	0.30	0.01667	24	LC6	Soil_Separation_Soil7
	Cut127	0.25	LC1	Static	0.37	0.01667	24	LC6	Baseline_Soil11
	Cut128	0.25	LC1	Static	0.32	0.01667	24	LC1	Static
	Cut129	0.31	LC1	Static	0.62	0.01667	24	LC9	Baseline_Soil11
	Cut130	0.41	LC1	Static	0.32	0.01667	24	LC6	Baseline_Soil11
	Cut131	0.21	LC1	Static	0.34	0.01667	24	LC6	Baseline_Soil7
	Cut132	0.20	LC1	Static	0.43	0.01667	24	LC1	Static
	Cut133	0.17	LC1	Static	0.52	0.01667	24	LC6	Baseline_Soil11

Table 3B-12: Design Evaluation of Critical Sections of RC Members in RXB for Out-of-Plane PM and PV Design Conditions (Continued)

Member	CutID	Design for PM			Design for PV				
		DCR	LC[a]	S_ID[b]	DCR	ro[c]	s_{mx}[d]	LC[a]	CS_ID[b]
Floor-EL100	Cut501	0.44	LC9	Soil_Separation_Soil7	0.53	0.01667	11	LC9	Soil_Separation_Soil7
	Cut502	0.92	LC9	Soil_Separation_Soil7	0.90	0.01667	16	LC9	Soil_Separation_Soil7
	Cut503	0.27	LC9_3	Soil_Separation_Soil7	0.33	0.01667	16	LC9	Baseline_Soil9
	Cut504	0.69	LC9	Soil_Separation_Soil7	0.66	0.01667	11	LC9	Baseline_Soil9
	Cut505	0.56	LC9	Baseline_Soil9	0.10	0.01667	11	LC9_3	Soil_Separation_Soil7
	Cut506	0.45	LC9	Baseline_Soil9	0.55	0.01667	11	LC9	Baseline_Soil9
	Cut507	0.48	LC6	Baseline_Soil7	0.81	0.01667	11	LC9	Baseline_Soil9
	Cut508	0.39	LC6	Soil_Separation_Soil7	0.13	0.01667	11	LC6	Baseline_Soil7
	Cut509	0.72	LC9	Soil_Separation_Soil7	0.60	0.01667	11	LC9	Soil_Separation_Soil7
	Cut510	0.75	LC9	Baseline_Soil9	0.17	0.01667	11	LC9	Baseline_Soil9
	Cut511	0.42	LC9	Baseline_Soil9	0.77	0.08800	5	LC9	Baseline_Soil9
	Cut512	0.41	LC6	Baseline_Soil7	0.66	0.08800	5	LC9	Baseline_Soil11
	Cut513	0.66	LC9	Baseline_Soil9	0.15	0.01667	11	LC9	Baseline_Soil9
	Cut514	0.60	LC9	Baseline_Soil9	0.74	0.01667	11	LC9	Baseline_Soil9
	Cut526	0.89	LC9	Soil_Separation_Soil7	0.37	0.01667	10	LC9	Baseline_Soil9
	Cut527	0.29	LC9_3	Soil_Separation_Soil7	0.14	0.01667	10	LC9_3	Baseline_Soil9
	Cut528	0.65	LC9	Soil_Separation_Soil7	0.28	0.01667	10	LC6	Baseline_Soil7
	Cut529	0.33	LC9	Baseline_Soil7	0.32	0.01667	10	LC6	Baseline_Soil7
	Cut530	0.34	LC9	Baseline_Soil7	0.31	0.01667	10	LC6	Baseline_Soil7
	Cut531	0.53	LC9	Baseline_Soil7	0.24	0.01667	10	LC9	Soil_Separation_Soil7
	Cut532	0.78	LC9	Soil_Separation_Soil7	0.90	0.01667	10	LC9	Soil_Separation_Soil7
	Cut533	0.43	LC9	Baseline_Soil9	0.43	0.01667	10	LC9	Baseline_Soil9
	Cut534	0.53	LC9	Baseline_Soil9	0.48	0.01667	10	LC9	Baseline_Soil9
	Cut535	0.86	LC9	Baseline_Soil9	0.11	0.01667	10	LC9	Baseline_Soil9

Table 3B-12: Design Evaluation of Critical Sections of RC Members in RXB for Out-of-Plane PM and PV Design Conditions (Continued)

Member	CutID	Design for PM			Design for PV				
		DCR	LC[a]	S_ID[b]	DCR	ro[c]	s_{mx}[d]	LC[a]	CS_ID[b]
Roof	Cut801	0.52	LC9_1	Soil_Separation_Soil7	0.02	0.01667	17	LC9_1	Baseline_Soil7
	Cut802	0.27	LC9_1	Soil_Separation_Soil7	0.43	0.01667	17	LC6	Baseline_Soil7
	Cut803	0.29	LC9_1	Soil_Separation_Soil7	0.49	0.01667	17	LC6	Baseline_Soil7
	Cut804	0.70	LC9_1	Baseline_Soil7	0.11	0.01667	17	LC9_1	Baseline_Soil9
	Cut805	0.30	LC9_1	Baseline_Soil7	0.48	0.01667	17	LC6	Soil_Separation_Soil7
	Cut806	0.55	LC9_3	Baseline_Soil9	0.08	0.01667	17	LC9_3	Baseline_Soil11
	Cut807	0.26	LC9_3	Baseline_Soil9	0.41	0.01667	17	LC6	Soil_Separation_Soil7
	Cut808	0.47	LC9_1	Baseline_Soil7	0.03	0.01667	17	LC9_1	Baseline_Soil9
	Cut826	0.20	LC9_3	Baseline_Soil9	0.20	0.01667	16	LC6	Baseline_Soil7
	Cut827	0.47	LC9_3	Baseline_Soil9	0.05	0.01667	16	LC9_3	Baseline_Soil9
	Cut828	0.58	LC9_1	Baseline_Soil7	0.06	0.01667	16	LC9_1	Baseline_Soil7
	Cut829	0.20	LC7_55_5	Static	0.14	0.01667	16	LC7_55_5	Static
	Cut830	0.62	LC9_1	Baseline_Soil7	0.06	0.01667	16	LC9_1	Soil_Separation_Soil7
	Cut831	0.20	LC9_1	Baseline_Soil7	0.20	0.01667	16	LC6	Baseline_Soil7
	Cut832	0.26	LC6	Baseline_Soil7	0.09	0.01667	16	LC9_3	Baseline_Soil9

Notes:
a)The load combination for the governing design condition
b)Soil Type. The label "Static" is used when design is governed by a non-seismic load combination.
c)Ratio of shear reinforcement (Av/s)
d)Maximum allowable spacing for out-of-plane shear reinforcement

Table 3B-13: Design Evaluation of Critical Sections of RC Members in RXB for In-Plane Shear

Member	CutID	Design for in-plane shear						
		DCR	ro[a]	A_{s_IP}[b] (in^2/ft)	ro/ro_max[c]	$A_{s_add_IP}$[d] (in^2/ft) (EF)	LC[e]	S_ID[f]
Floor-EL100	Cut551	-[g]	0.00279	0.80	0.39	-	LC9	Soil_Separation_Soil7
	Cut552	-	0.00071	0.20	0.10	-	LC9_3	Soil_Separation_Soil7
	Cut553	-	0.00196	0.57	0.28	-	LC9	Baseline_Soil7
	Cut576	-	0.00280	0.81	0.40	-	LC9_3	Soil_Separation_Soil7
	Cut577	-	0.00287	0.83	0.41	-	LC9	Soil_Separation_Soil7
	Cut578	-	0.00450	1.30	0.64	-	LC9	Soil_Separation_Soil7
	Cut579	-	0.00405	1.17	0.57	-	LC9_3	Soil_Separation_Soil7
	Cut580	-	0.00073	0.21	0.10	-	LC9_3	Soil_Separation_Soil7
	Cut581	-	0.00099	0.29	0.14	-	LC9_1	Soil_Separation_Soil7
	Cut582	-	0.00144	0.41	0.20	-	LC6	Baseline_Soil7
	Cut583	-	0.00164	0.47	0.23	-	LC6	Soil_Separation_Soil7
Roof	Cut876	0.94[h]	0.00837	3.61	1.00	0.23	LC6_3	Baseline_Soil7
	Cut877	0.92[h]	0.00837	3.61	1.00	0.23	LC6	Baseline_Soil7

Notes:

a) Ratio of in-plane shear reinforcement ($A_{s_IP}/[(t)(s)]$)

b) Total in-plane shear reinforcement required to resist the in-plane shear demand (ro[(t)(s)])

c) Ratio of total required in-plane shear reinforcement to the maximum allowable in-plane shear reinforcement

d) Additional in-plane shear reinforcement area at each face. Calculated as $A_{s_add_IP} = (A_{s_IP}/2)-(A_{s_PM})(1.0\text{-ave(DCRPM)})>0.0$, where A_{s_IP} = ro[(t)(s)], t is the thickness of the section, s is the spacing of the reinforcement, A_{s_PM} is the longitudinal reinforcement at top or bottom of the section defined for out-of-plane PM demand, and ave(DCRPM) is average of the maximum demand-to-capacity ratios due to out-of-plane PM design condition from all sections that are perpendicular to the direction of the in-plane shear demand (i.e. for in-plane shear demand in the north-south direction, the DCR_{PM} results from all sections running in the east-west direction are considered in the same member). The symbol "-" is used at a section where un-utilized amount of the longitudinal reinforcement defined for out-of-plane PM demand is more than the in-plane shear reinforcement required at that section.

e) Load combination that governs the in-plane shear design at the section

f) Soil type for the governing design condition. The label "Static" is used when design is governed by a non-seismic load combination.

g) At these section cuts, the required in-plane shear reinforcement ratio, ro, is calculated to meet the design requirement. For this reason, the DCR value is theoretically is equal to 1.0. The ratio ro/ro_max can be used to assess the utilization of the section in terms of its maximum in-plane shear capacity per code. As discussed in Section 3B.2.3.3, the required in-plane shear reinforcement is added on the longitudinal reinforcement specified for out-of-plane bending design conditions. The final provided longitudinal reinforcement is typically larger than the required in-plane shear reinforcement, ro, and, hence, the corresponding DCR for in-plane shear for these sections will be less than 1.0.

h) At these section, the DCR_{IP} is calculated considering the maximum allowable in-plane shear reinforcement that can be provided at the section per ACI 349 requirements.

Table 3B-14: Final Design Properties of Critical Sections of RC Members in RXB and the Maximum DCR Value with the Governing Design Condition

RC Member	Reinf. Layout	Section Thick. (in.)	Clear Cover (in.)	Typical Reinforcement[b]				DCR (governing design condition)
				OOP Bending (EFEW)[c]	OOP Shear (EW)[d]	Add. IP Shear (EF) (in^2/ft)[e]	Total Long. (EF)	
Basemat	Figure 3B-30	96[a]	3.0	2#11@12	#4@12	-	2#11@12" (E-W) 2#11@12" (N-S)	0.73 (OOP-Shear)
Floor EL 100	Figure 3B-31	24[c]	2.0	#9@12	#4@10	-	#9@12" (E-W) #11@9" (N-S)	0.92(PM)
Roof	Figure 3B-31a	36	2.0	2#11@12	#4@12	0.23	2#11@12" (E-W) 2#11@9" (N-S)	0.94 (IP-Shear)

Notes:

a) The thickness of basemat is 96 in. (8 ft) except at the region within the pool area where an additional 12 in. layer is provided for non-structural reasons. This additional thickness is conservatively not considered in calculating section capacities for design evaluations.

b) The reinforcement provided in this table is "typical" reinforcement and may show variation in different sub-regions of the members as shown in Figure 3B-30 through Figure 3B-31a.

c) The longitudinal reinforcement is defined such that the reinforcement running in the north-south direction is located outside of the reinforcement running in the east-west direction.

d) Out-of-plane shear reinforcement is defined to be the same in both horizontal directions except at specific regions shown in Figure 3B-30 through Figure 3B-31a. In slabs with 24 in. thickness, the spacing of reinforcement is limited by d/2.

e) The additional required reinforcement for in-plane shear demand is summarized in Table 3B-13.

Table 3B-15: Demand-to-Capacity Ratio (DCR) for the NPM Skirt Restraint

Analysis Type	Failure Mode	DCR
Vertical Analysis	Bearing on annular steel plate	0.03
	Bearing on concrete	0.28
	Flexural buckling of guide element	0.72
Lateral Analysis	Compression flange yielding of skirt restraint	0.18
	Tension flange yielding of skirt restraint	0.18
	Flexural buckling of skirt restraint	0.24
	Torsional and flexural-torsional buckling of skirt restraint	0.24
	Combined flexure and compression of bumper	0.41
	Bearing on steel bumper	0.22
	Weld between built-up web and flanges	0.69
	Weld between built-up beam and connecting plate	0.85
	Base metal of the weld between built-up beam and connecting plate	0.65
	Weld between stiffeners and built-up flanges	0.80
	Base metal of the weld between stiffeners and built-up flanges	0.27
	Weld between connecting plate and faceplate	0.84
	Base metal for the weld between connecting plate and faceplate	0.69
	Punching shear in bay wall	0.77
	Embedded plate stress	0.55
	Weld between SC wall faceplate and embedded plate	0.81
	Base metal for the weld between SC wall faceplate and embedded plate	0.42

Table 3B-16: Demand-to-Capacity Ratio (DCR) for the NPM Lug Restraint

Component	Failure Mode	DCR
Wedge-Jack to Lug Restraint Connection	Bearing strength of top and bottom restraint plates (all lug restraints)	0.264
	Weld connection between wedge-jack and lug restraint (all lug restraints)	0.575
Top and Bottom Restraint Stiffener Plate	Combined compression and flexure (typical lug restraint) (A-A)	0.234
	Combined compression and flexure (typical lug restraint) (B-B)	0.698
	Shear of top/bottom restraint stiffener plate (typical lug restraint) (B-B)	0.267
	Shear of top/bottom restraint stiffener plate (typical lug restraint) (A-A)	0.252
	Combined compression and flexure (back-of-bay and end bay) (A-A)	0.727
	Combined compression and flexure (back-of-bay and end bay) (B-B)	0.867
	Shear of top/bottom restraint stiffener plate (back-of-bay and end bay) (B-B)	0.260
	Shear of top/bottom restraint stiffener plate (back-of-bay and end bay) (C-C)	0.361
Stiffener Plate	Weld connection (typical lug restraint)	0.741
	Flexure (typical lug restraint)	0.044
	Weld connection (back-of-bay and end bay)	0.727
	Flexure (back-of-bay and end bay)	0.085
Restraint Stiffener/ Embedded Plate and SC Wall Faceplate Connection	Single PJP weld connection (typical lug restraint)	0.884
	Single PJP weld connection (back-of-bay and end bay)	0.849
	Shear strength of restraint plate at intermediate section (back-of-bay and end bay)	0.398
	Flexure strength of restraint plate at intermediate section (back-of-bay and end bay)	0.732
Top/Bottom Restraint	Weld connection between top/bottom restraint plates and intermediate restraint plate (typical lug restraint)	0.816
	Weld connection between top/bottom restraint plates and intermediate restraint plate (back-of-bay and end bay)	0.753
	Connection between top/bottom restraint stiffener plates and end-bearing stiffening plates (typical lug restraint)	0.852
	Connection between top/bottom restraint stiffener plates and end-bearing stiffening plates (back-of-bay and end bay)	0.782
End-Bearing Intermediate Built-up W-Shape	Compressive strength of end-bearing intermediate built-up (typical lug restraint)	0.558
	Compressive strength of end-bearing intermediate built-up (back-of-bay and end bay)	0.453
Concrete	Bearing strength of interior SC wall concrete (typical lug restraint)	0.753
	Bearing strength of interior SC wall concrete (back-of-bay and end bay)	0.611
Embedded Steel Components in SC Wall	Shear strength of steel components interior to SC wall (typical lug restraint)	0.186
	Flexural strength of steel components interior to SC wall (typical lug restraint)	0.163
	Shear strength of steel components interior to SC wall (back-of-bay and end bay)	0.078
	Flexural strength of steel components interior to SC wall (back-of-bay and end bay)	0.097
Shear stud embedment	Stud strength (for all lug restraints)	0.881

Table 3B-17: Demand-to-Capacity Ratio (DCR) for the RBC Corbel

Component	Loading Direction	Failure Mode	DCR
Transverse Stiffener	Downward	Shear	0.49
Base metal of weld between transverse stiffener and seat stiffener	Downward	Shear	0.46
Transverse Stiffener Bolted Connection	Downward	Shear	0.52
Seat Stiffener	Downward	Combined axial and flexural	0.49
Weld between seat stiffener and SC wall faceplate	Downward	Shear	0.47
Weld between top plate and seat stiffener	Downward	Shear	0.51
Weld between seat stiffener and SC wall faceplate	Downward	Shear	0.51
Weld between top plate and rib plate	Downward	Shear	0.49
Base metal of weld between top plate and rib plate	Downward	Shear	0.52
Weld between bottom plate and seat stiffener	Downward	Shear	0.52
Weld between Top Plate and SC Wall Faceplate	East-West	Shear	0.47
Top Plate to Top Plate Splice Weld	East-West	Shear	0.51

Table 3B-18: Coupler Dimensional Specifications

Bar Size	Φ_1 (in.)	Φ_2 (in.)
#8	1.56	1.0
#9	1.56	0.94
#10	2.0	0.94
#11	2.0	1.13

Table 3B-19: DVLP PL Dimensional Properties

Bar Size	#8		#9		#10		#11	
Properties	**Single**	**Double**	**Single**	**Double**	**Single**	**Double**	**Single**	**Double**
W_{DP} (in.)	5.25	5.25	5.5	5.5	5.75	5.75	6.25	6.25
H_{DP} (in.)	5	10	5	10	5	10	5	10
t_{DP} (in.)	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5

Table 3B-20: Connection Design Results

Bar Size	Required Strength (kip)[1]	Total Effective Throat (in.)	Total Weld Strength (kip)	DCR
#8	71.1	0.552	75.86	0.94
#9	90	0.705	100.42	0.90
#10	114.3	0.817	122.64	0.93
#11	140.4	0.882	153.08	0.92

[1] Corresponds to reinforcement in a single layer. The weld strength is evaluated for a single layer reinforcement as each welded coupler connection to development plate is evaluated individually

Table 3B-21: Load Combinations for Design of RC Members in CRB

Load Combination	ACI 349 Equation Number	Load Combination ID used in Design Scripts
1.4D	9-1	LC1
1.2D+1.6L+0.5S	9-2	LC2b
0.9D+1.6L+0.5S		LC2b_1(1)
0.9D+1.44L+0.45S		LC2b_2(1)
1.2D+0.8L+1.6S	9-3	LC3b
0.9D+0.8L+1.6S		LC3b_1(1)
0.9D+0.72L+1.44S		LC3b_2(1)
1.2D+1.6L+1.6W	9-5	LC5(2)
D+0.8L+Ess	9-6	LC6
0.9D+0.8L+Ess		LC6N(1)
0.9D+0.72L+Ess		LC6Mx(1)
0.9D+Ess		LC6Mn(1)
D+0.8L+(Wt or Wh)	9-7	LC7
0.9D+0.8L+(Wt or Wh)		LC7N
0.9D+0.72L+0.9(Wt or Wh)		LC7Mx

Notes:

1. Load combinations are related to reduced dead load effect as per 9.2.3 of ACI 349 -13.
2. Load combination 9-4 is not applicable as it includes operating basis earthquake (OBE). Analysis and evaluation for OBE for the design is not required for SC-I structures since OBE is established as 1/3 of the safety shutdown earthquake (SSE).
3. For ACI 349 Eq 9-7 load combination, $W_t = W_{wt}+0.5W_{pt}+W_{mt}$ and $W_h=W_{wh}+W_{mh}$. Where W_{wt} is tornado wind, W_{pf} is tornado atmospheric pressure and W_{mt} is tornado missile, W_{wh} is hurricane wind and W_{mh} is hurricane missile

Table 3B-22: Final Design Properties of Critical Wall and Slab Sections of the CRB

Component	Mop Reinforcement EFEW $A_{sMop}(in^2/ft)$	Required Vip reinforcement $A_{sIP}(in^2/ft)=r_t*h*12"$	Additional Vip Reinforcement EFEW $A_{sIP_add}(in^2/ft)=A_{sIP}/2-A_{sMop}*(1-maxDCR_{PM})$	Total Required Reinforcement EFEW $As_{tot}(in^2/ft)=A_{sMop}+A_{sIP_Add}$	Total Provided Longitudinal Reinforcement EFEW As_{tot_prov}	Out of Plane Shear Reinforcement Required/ Provided (in^2/ft)
CRB_3_100(1)	1#8@12"=0.79	0.00394*36*12=1.70	0.67	1.46	1#11@12"	-
CRB_3_123(1)	1#8@12"=0.79	0.00098*36*12=0.42	0	0.79	1#8@12"	-
CRB_5_100(1)	1#9@12"=1.00	0.00115*36*12=0.49	0.15	1.15	1#10@12"	-
CRB_5_123(1)	1#8@12"=0.79	-	-	0.79	1#8@12"	-
CRB_H_100(1)	1#11@12"=1.56	-	-	1.56	1#11@12"	-
CRB_H_123(1)	1#10@12"=1.27	-	-	1.27	1#10@12"	-
CRB_slab_123_1(2)	2#9@12"=2.00	-	-	2.00	2#10@12"	-
CRB_slab_123_2(2)	2#9@12"=2.00	0.00044*24*12=0.13	0	2.00	2#10@12"	-
CRB_slab_123_3(2)	2#9@12"=2.00	-	-	2.00	2#10@12"	-
CRB_Basemat_100(2)	1#11@12"=1.56	-	-	1.56	1#11@12"	0.0018/ #3@12@12(3)

Notes:
1. For all walls, the vertical reinforcement is on the inside and the horizontal reinforcement is on the outside.
2. For all slabs, and the basemat, NS reinforcement is on the inside and EW reinforcement is on the outside.
3. Required out of plane reinforcement is the maximum needed for the component.

Table 3B-23: CRB Summary of Maximum Demand-to-Capacity Ratio (DCR)

Component	PMop		Vip		PVop	
	Max DCR	Load Comb, Soil Type	Max DCR	Load Comb, Soil Type	Max DCR	Load Comb, Soil Type
CRB_3_100	0.77	LC6Mn/ Soil7	1.00(1)	LC6/ Soil7	0.28	LC6/ Soil11
CRB_3_123	0.39	LC6N/ Soil7	1.00(4)	LC6/ Soil7	0.23	LC6/ Soil9
CRB_5_100	0.90	LC6Mn/ Soil7	1.00(1)	LC6/ Soil7	0.32	LC6/ Soil7
CRB_5_123	0.80	LC6/ Soil7	0.84(3)	LC6/ Soil7	0.34	LC6/ Soil9
CRB_H_100	0.98	LC6/ Soil7	0.83(3)	LC6Mn/ Soil9	0.87	LC6/ Soil7
CRB_H_123	0.69	LC7N_2_2/ Soil11	0.60(3)	LC6/ Soil9	0.91	LC7_3_2/ Soil11
CRB_slab_123_1	0.67	LC6/ Soil7	0.89(3)	LC6/ Soil7	0.81	LC6/ Soil7
CRB_slab_123_2	0.89	LC6/ Soil7	1.00(4)	LC6/ Soil7	0.87	LC6/ Soil7
CRB_slab_123_3	0.50	LC6/ Soil7	0.83(3)	LC6Mn/ Soil7	0.74	LC6/ Soil7
CRB_Basemat_100	0.93	LC6/ Soil11	0.40(3)	LC6/ Soil11	1.00(2)	LC6/ Soil11

Notes:

1. The DCR values are conservative as they only reflect the IP shear capacity of the concrete. The DCR does not take into account the additional reinforcement added due to the additional IP shear reinforcement calculated in Table 3B-22. The actual DCR is less than 1.0.
2. The actual value of the OOP shear DCR is conservative as it does not take into account the actual additional reinforcement added to resist OOP shear. The actual DCR is less than 1.0.
3. The interaction ratio listed in the table is conservative as it is based on the concrete section capacity only neglecting the contribution of the excess flexural reinforcement to resist IP shear unless additional IP shear is added.
4. The actual value of the IP shear DCR for these components is conservative as it does not take credit for the contribution of the excess flexural reinforcement to resist IP shear. The conservatism is demonstrated as follows:
   - From Table 3B-25, DCR =1.0 is reported for SC-8, SC-9, and SC-15. The maximum DCR for PMop for these section cuts is 0.18. Table 3B-22 demonstrates no additional IP shear reinforcement required based on the maximum DCR for PMop of 0.39. Thus this is conservative and actual DCR <1.0.
   - From Table 3B-32, DCR=1.0 occurs in a region where PMop does not control the design. PMop for SC-5 which is in the same region of SC-9 has DCR of 0.56 for PMop. Thus, the actual maximum DCR reported for IP is <1.0.

Table 3B-24: Demand-to-Capacity Ratio (DCR) and Result Summary for CRB_3_100

PMop				PVop					PVip			
Cut_ID	DCR	LC_ID	Soil Type	DCR	Av/s,ro,rot	max_spc	LC_ID	Soil Type	DCR	Av/s,ro,rot	LC_ID	Soil Type
sc_7	0.77	LC6Mn	Soil7	0.28	0.00000	15.94	LC6	Soil11	N/A	N/A	-	-
sc_4	0.70	LC6Mn	Soil7	0.17	0.00000	15.94	LC6Mn	Soil7	N/A	N/A	-	-
sc_9	0.44	LC6	Soil7	0.06	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_10	0.27	LC6	Soil7	0.04	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_8	0.20	LC6	Soil7	0.09	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_6	0.16	LC6	Soil7	0.13	0.00000	15.94	LC6Mn	Soil9	N/A	N/A	-	-
sc_3	0.16	LC6	Soil7	0.14	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_5	0.14	LC1	Soil11	0.09	0.00000	15.94	LC6Mn	Soil7	N/A	N/A	-	-
sc_2	0.12	LC6	Soil7	0.05	0.00000	15.94	LC6Mn	Soil7	N/A	N/A	-	-
sc_18	0.11	LC1	Soil11	0.19	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_1	0.08	LC6	Soil11	0.12	0.00000	15.94	LC6Mn	Soil9	N/A	N/A	-	-
sc_19	0.04	LC6Mn	Soil7	0.11	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_11	N/A	-	-	N/A	N/A	0.00	-	-	1.00	0.00145	LC6	Soil7
sc_12	N/A	-	-	N/A	N/A	0.00	-	-	1.00	0.00350	LC6	Soil7
sc_13	N/A	-	-	N/A	N/A	0.00	-	-	1.00	0.00394	LC6	Soil7
sc_14	N/A	-	-	N/A	N/A	0.00	-	-	1.00	0.00272	LC6	Soil7
sc_15	N/A	-	-	N/A	N/A	0.00	-	-	1.00	0.00173	LC6	Soil7

Table 3B-25: Demand-to-Capacity Ratio (DCR) and Result Summary for CRB_3-123

PMop				PVop					PVip			
Cut_ID	DCR	LC_ID	Soil Type	DCR	Av/s,ro,rot	max_spc	LC_ID	Soil Type	DCR	Av/s,ro,rot	LC_ID	Soil Type
sc_1	0.39	LC6N	Soil7	0.23	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_2	0.17	LC6	Soil7	0.01	0.00000	15.94	LC6Mn	Soil9	0.83	0.00001	LC6	Soil7
sc_3	0.33	LC6Mn	Soil7	0.17	0.00000	15.94	LC6Mn	Soil9	N/A	N/A	-	-
sc_4	0.12	LC6N	Soil9	0.10	0.00000	15.94	LC6	Soil7	0.97	0.00011	LC6	Soil7
sc-5	0.10	LC6N	Soil9	0.09	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc-6	0.36	LC6	Soil7	0.21	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc-7	0.07	LC6	Soil9	0.07	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_8	0.18	LC6Mn	Soil9	0.03	0.00000	15.94	LC6N	Soil9	1.00	0.00094	LC6	Soil7
sc_9	0.07	LC6	Soil9	0.22	0.00000	15.94	LC6	Soil9	1.00	0.00098	LC6	Soil7
sc-10	0.08	LC6	Soil9	0.22	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc-11	0.10	LC1	Soil11	0.20	0.00000	15.94	LC6N	Soil9	N/A	N/A	-	-
sc-12	0.03	LC1	Soil11	0.05	0.00000	15.94	LC6Mn	Soil7	N/A	N/A	-	-
sc-13	0.12	LC6N	Soil9	0.22	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc-14	0.16	LC6	Soil9	0.14	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc-15	N/A	-	-	N/A	N/A	0.00	-	-	1.00	0.00034	LC6	Soil7

Table 3B-26: Demand-to-Capacity Ratio (DCR) and Result Summary for CRB_5_100

PMop				PVop					PVip			
Cut_ID	DCR	LC_ID	Soil Type	DCR	Av/ s,ro,rot	max_spc	LC_ID	Soil Type	DCR	Av/s,r0,rot	LC ID	Soil Type
sc_9	0.90	LC6Mn	Soil7	0.27	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_12	0.86	LC6Mn	Soil7	0.32	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_11	0.45	LC6Mn	Soil9	0.18	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_5	0.42	LC6Mn	Soil9	0.18	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_7	0.30	LC5_3	Soil11	0.19	0.00000	15.94	LC7_7_2	Soil11	N/A	N/A	-	-
sc_6	0.24	LC6N	Soil9	0.07	0.00000	15.94	LC5_4	Soil11	N/A	N/A	-	-
sc_1	0.19	LC6N	Soil9	0.14	0.00000	15.94	LC6Mn	Soil9	N/A	N/A	-	-
sc_13	0.17	LC6	Soil7	0.17	0.00000	15.94	LC1	Soil11	N/A	N/A	-	-
sc_2	0.15	LC6	Soil11	0.07	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_3	0.13	LC6Mn	Soil9	0.06	0.00000	15.94	LC6	Soil11	N/A	N/A	-	-
sc_8	0.12	LC6	Soil11	0.12	0.00000	15.94	LC5_3	Soil11	N/A	N/A	-	-
sc_4	0.11	LC6	Soil9	0.06	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_10	N/A	-	-	N/A	N/A	0.00	-	-	1.00	0.00115	LC6	Soil7

Table 3B-27: Demand-to-Capacity Ratio (DCR) and Result Summary for CRB_5_123

PMop				PVop					PVip			
Cut_ID	DCR	LC_ID	Soil Type	DCR	Av/s,ro,rot	max_spc	LC_ID	Soil Type	DCR	Av/s,ro,rot	LC_ID	Soil Type
sc_1	0.80	LC6	Soil7	0.34	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_9	0.57	LC6	Soil7	0.26	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_4	0.53	LC6	Soil7	0.24	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_2	0.45	LC6Mn	Soil7	0.21	0.00000	15.94	LC6Mn	Soil9	N/A	N/A	-	-
sc_10	0.37	LC6	Soil7	0.20	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_6	0.33	LC6N	Soil7	0.14	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_5	0.30	LC6	Soil9	0.32	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_7	0.25	LC6Mn	Soil9	0.11	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_3	0.20	LC6N	Soil9	0.14	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_8	N/A	-	-	N/A	N/A	0.00	-	-	0.84	0.00000	LC6	Soil7

Table 3B-28: Demand-to-Capacity Ratio (DCR) and Result Summary for CRB_H_100

PMop				PVop					PVip			
Cut_ID	DCR	LC_ID	Soil Type	DCR	Av/s,ro,rot	max_spc	LC_ID	Soil Type	DCR	Av/s,ro,rot	LC_ID	Soil Type
sc_6	0.98	LC6	Soil7	0.87	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_16	0.91	LC6	Soil7	0.72	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_7	0.68	LC6	Soil7	0.58	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_5	0.64	LC7N_1_1	Soil11	0.70	0.00000	15.94	LC7_1_1	Soil11	N/A	N/A	-	-
sc_14	0.54	LC6	Soil7	0.52	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_15	0.51	LC6Mn	Soil7	0.18	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_13	0.47	LC6Mn	Soil7	0.22	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_1	0.32	LC7N_2_1	Soil11	0.44	0.00000	15.94	LC7N_2_1	Soil11	N/A	N/A	-	-
sc_3	0.29	LC7N_2_1	Soil11	0.11	0.00000	15.94	LC6Mn	Soil7	N/A	N/A	-	-
sc_11	0.26	LC6Mn	Soil7	0.07	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_9	0.23	LC6N	Soil7	0.10	0.00000	15.94	LC6Mn	Soil7	N/A	N/A	-	-
sc_2	0.20	LC5_3	Soil11	0.19	0.00000	15.94	LC7_7_2	Soil11	N/A	N/A	-	-
sc_10	0.18	LC5_4	Soil11	0.17	0.00000	15.94	LC7_8_2	Soil11	N/A	N/A	-	-
sc_20	0.17	LC6	Soil7	0.33	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_4	0.13	LC6	Soil9	0.16	0.00000	15.94	LC7_7_2	Soil11	N/A	N/A	-	-
sc_21	0.07	LC6	Soil7	0.65	0.00000	15.94	LC7N_2_1	Soil11	N/A	N/A	-	-
sc_12	0.06	LC6	Soil7	0.09	0.00000	15.94	LC6	Soil9	N/A	N/A	-	-
sc_8	N/A	-	-	N/A	N/A	0.00	-	-	0.83	0.00000	LC6Mn	Soil9
sc_17	N/A	-	-	N/A	N/A	0.00	-	-	0.83	0.00000	LC6	Soil7
sc_18	N/A	-	-	N/A	N/A	0.00	-	-	0.78	0.00000	LC6N	Soil9
sc_19	N/A	-	-	N/A	N/A	0.00	-	-	0.73	0.00000	LC6N	Soil9

Table 3B-29: Demand-to-Capacity Ratio (DCR) and Result Summary for CRB_H_123

PMop				PVop					PVip			
Cut_ID	DCR	LC_ID	Soil Type	DCR	Av/s,ro,rot	max_spc	LC_ID	Soil Type	DCR	Av/s,ro,rot	LC_ID	Soil Type
sc_3	0.69	LC7N_2_2	Soil11	0.91	0.00000	15.94	LC7_3_2	Soil11	N/A	N/A	-	-
sc_1	0.67	LC6	Soil7	0.50	0.00000	15.94	LC7_1_2	Soil11	N/A	N/A	-	-
sc_4	0.42	LC6Mn	Soil7	0.23	0.00000	15.94	LC6Mn	Soil7	N/A	N/A	-	-
sc_18	0.41	LC6	Soil7	0.19	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_6	0.39	LC6	Soil7	0.46	0.00000	15.94	LC7N_2_2	Soil11	N/A	N/A	-	-
sc_13	0.34	LC6	Soil9	0.24	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_10	0.31	LC6	Soil7	0.25	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_2	0.27	LC6Mn	Soil7	0.23	0.00000	15.94	LC6Mn	Soil7	N/A	N/A	-	-
sc_5	0.26	LC6N	Soil9	0.09	0.00000	15.94	LC7_8_2	Soil11	N/A	N/A	-	-
sc_11	0.24	LC6Mn	Soil9	0.19	0.00000	15.94	LC6Mn	Soil7	N/A	N/A	-	-
sc_19	0.23	LC6Mn	Soil7	0.37	0.00000	15.94	LC7_2_1	Soil11	N/A	N/A	-	-
sc_14	0.22	LC6N	Soil9	0.08	0.00000	15.94	LC7_7_1	Soil11	N/A	N/A	-	-
sc_16	0.21	LC6	Soil7	0.19	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_9	0.17	LC6N	Soil7	0.10	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_20	0.11	LC6Mn	Soil7	0.17	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_17	0.09	LC6Mn	Soil7	0.16	0.00000	15.94	LC6	Soil7	N/A	N/A	-	-
sc_12	0.07	LC6Mn	Soil9	0.23	0.00000	15.94	LC7N_1_1	Soil11	N/A	N/A	-	-
sc_7	0.05	LC6	Soil9	0.19	0.00000	15.94	LC7_7_2	Soil11	N/A	N/A	-	-
sc_8	N/A	-	-	N/A	N/A	0.00	-	-	0.60	0.00000	LC6	Soil9
sc_15	N/A	-	-	N/A	N/A	0.00	-	-	0.49	0.00000	LC6	Soil9

Table 3B-30: Demand-to-Capacity Ratio (DCR) and Result Summary for CRB Basemat

PMop				PVop					PVip			
Cut_ID	DCR	LC_ID	Soil Type	DCR	Av/s,ro,rot	max_spc	LC_ID	Soil Type	DCR	Av/s,ro,rot	LC_ID	Soil Type
sc_14	0.93	LC6	Soil11	0.18	0.00000	24.00	LC6	Soil11	N/A	N/A	-	-
sc_17	0.75	LC6	Soil11	0.10	0.00000	24.00	LC6	Soil11	N/A	N/A	-	-
sc_8	0.60	LC1	Soil11	1.00	0.00177	24.00	LC6	Soil11	N/A	N/A	-	-
sc_11	0.56	LC1	Soil11	0.95	0.00000	24.00	LC6	Soil11	N/A	N/A	-	-
sc_18	0.47	LC6N	Soil11	0.26	0.00000	24.00	LC6	Soil7	N/A	N/A	-	-
sc_15	0.47	LC6	Soil11	0.09	0.00000	24.00	LC6	Soil11	N/A	N/A	-	-
sc_9	0.46	LC6	Soil11	0.30	0.00000	24.00	LC6	Soil11	N/A	N/A	-	-
sc_10	0.37	LC6	Soil11	0.34	0.00000	24.00	LC6	Soil11	N/A	N/A	-	-
sc_16	0.29	LC6	Soil11	0.63	0.00000	24.00	LC6	Soil11	N/A	N/A	-	-
sc_13	0.18	LC6	Soil11	0.20	0.00000	24.00	LC6	Soil7	N/A	N/A	-	-
sc_12	0.17	LC6Mn	Soil7	1.00	0.00360	24.00	LC6	Soil7	N/A	N/A	-	-
sc_1	N/A	-	-	N/A	N/A	0.00	-	-	0.25	0.00000	LC6	Soil7
sc_2	N/A	-	-	N/A	N/A	0.00	-	-	0.14	0.00000	LC6	Soil7
sc_3	N/A	-	-	N/A	N/A	0.00	-	-	0.18	0.00000	LC6	Soil7
sc_4	N/A	-	-	N/A	N/A	0.00	-	-	0.23	0.00000	LC6	Soil7
sc_5	N/A	-	-	N/A	N/A	0.00	-	-	0.39	0.00000	LC6	Soil11
sc_6	N/A	-	-	N/A	N/A	0.00	-	-	0.16	0.00000	LC6	Soil7
sc_7	N/A	-	-	N/A	N/A	0.00	-	-	0.40	0.00000	LC6	Soil11

Table 3B-31: Demand-to-Capacity Ratio (DCR) and Result Summary for CRB_123_1

PMop				PVop					PVip			
Cut_ID	DCR	LC_ID	Soil Type	DCR	Av/s,ro,rot	max_spc	LC_ID	Soil Type	DCR	Av/s,ro,rot	LC_ID	Soil Type
sco_4	0.57	LC6	Soil7	0.65	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_14	0.50	LC6	Soil7	0.52	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sco_5	0.49	LC6	Soil7	0.64	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_3	0.39	LC6	Soil7	0.09	0.00000	9.94	LC6	Soil9	N/A	N/A	-	-
sc_9	0.13	LC6	Soil7	0.05	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_7	0.12	LC6	Soil7	0.04	0.00000	9.94	LC6	Soil9	N/A	N/A	-	-
sc_15	0.12	LC6	Soil7	0.10	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_8	0.11	LC6	Soil7	0.02	0.00000	9.94	LC6	Soil9	N/A	N/A	-	-
sc_6	0.11	LC6	Soil7	0.02	0.00000	9.94	LC6Mn	Soil9	N/A	N/A	-	-
sc_16	0.11	LC6	Soil7	0.07	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
scv_1	N/A	-	-	0.72	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
scv_2	N/A	-	-	0.77	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
scv_4	N/A	-	-	0.63	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
scv_5	N/A	-	-	0.62	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_10	N/A	-	-	N/A	N/A	0.00	-	-	0.89	0.00000	LC6	Soil7
sc_11	N/A	-	-	N/A	N/A	0.00	-	-	0.67	0.00000	LC6	Soil9
sc_12	N/A	-	-	N/A	N/A	0.00	-	-	0.54	0.00000	LC6	Soil9
sc_13	N/A	-	-	N/A	N/A	0.00	-	-	0.51	0.00000	LC6	Soil9
sc_18	N/A	-	-	N/A	N/A	0.00	-	-	0.55	0.00000	LC6	Soil7
sco_4	0.57	LC6	Soil7	0.65	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_14	0.50	LC6	Soil7	0.52	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sco_5	0.49	LC6	Soil7	0.64	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-

Table 3B-32: Demand-to-Capacity Ratio (DCR) and Result Summary for CRB_123_2

PMop				PVop					PVip			
Cut_ID	DCR	LC_ID	Soil Type	DCR	Av/s,ro,rot	max_spc	LC_ID	Soil Type	DCR	Av/s,ro,rot	LC_ID	Soil Type
sc_4	0.89	LC6	Soil7	0.87	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sco_2	0.79	LC6	Soil7	0.81	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sco_1	0.60	LC6	Soil7	0.67	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sco_5	0.56	LC6	Soil7	0.66	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_3	0.50	LC6	Soil7	0.13	0.00000	9.94	LC6	Soil9	N/A	N/A	-	-
sc_12	0.19	LC6	Soil7	0.03	0.00000	9.94	LC6	Soil9	N/A	N/A	-	-
sc_11	0.19	LC6	Soil7	0.02	0.00000	9.94	LC6	Soil11	N/A	N/A	-	-
sc_10	0.13	LC6	Soil7	0.15	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_13	0.09	LC6	Soil7	0.03	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
scv_1	N/A	-	-	0.65	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
scv_2	N/A	-	-	0.80	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
scv_4	N/A	-	-	0.85	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
scv_5	N/A	-	-	0.64	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_6	N/A	-	-	N/A	N/A	0.00	-	-	0.97	0.00005	LC6	Soil7
sc_7	N/A	-	-	N/A	N/A	0.00	-	-	0.54	0.00000	LC6	Soil9
sc_8	N/A	-	-	N/A	N/A	0.00	-	-	0.57	0.00000	LC6	Soil9
sc_9	N/A	-	-	N/A	N/A	0.00	-	-	1.00	0.00044	LC6	Soil7

Table 3B-33: Demand-to-Capacity Ratio (DCR) and Result summary for CRB_123_3

PMop				PVop					PVip			
Cut_ID	DCR	LC_ID	Soil Type	DCR	Av/s,ro,rot	max_spc	LC_ID	Soil Type	DCR	Av/s,ro,rot	LC_ID	Soil Type
sco_4	0.50	LC6	Soil7	0.72	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sco_2	0.25	LC6	Soil7	0.57	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_10	0.21	LC6	Soil7	0.03	0.00000	9.94	LC6	Soil9	N/A	N/A	-	-
sc_3	0.15	LC6	Soil7	0.60	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sco_1	0.10	LC6	Soil7	0.19	0.00000	9.94	LC6Mn	Soil9	N/A	N/A	-	-
sco_5	0.09	LC6	Soil7	0.17	0.00000	9.94	LC6Mn	Soil7	N/A	N/A	-	-
scv_1	N/A	-	-	0.20	0.00000	9.94	LC6	Soil9	N/A	N/A	-	-
scv_2	N/A	-	-	0.55	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
scv_4	N/A	-	-	0.74	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
scv_5	N/A	-	-	0.19	0.00000	9.94	LC6	Soil7	N/A	N/A	-	-
sc_6	N/A	-	-	N/A	N/A	0.00	-	-	0.56	0.00000	LC6	Soil9
sc_7	N/A	-	-	N/A	N/A	0.00	-	-	0.28	0.00000	LC6	Soil9
sc_8	N/A	-	-	N/A	N/A	0.00	-	-	0.23	0.00000	LC6	Soil9
sc_9	N/A	-	-	N/A	N/A	0.00	-	-	0.83	0.00000	LC6Mn	Soil7

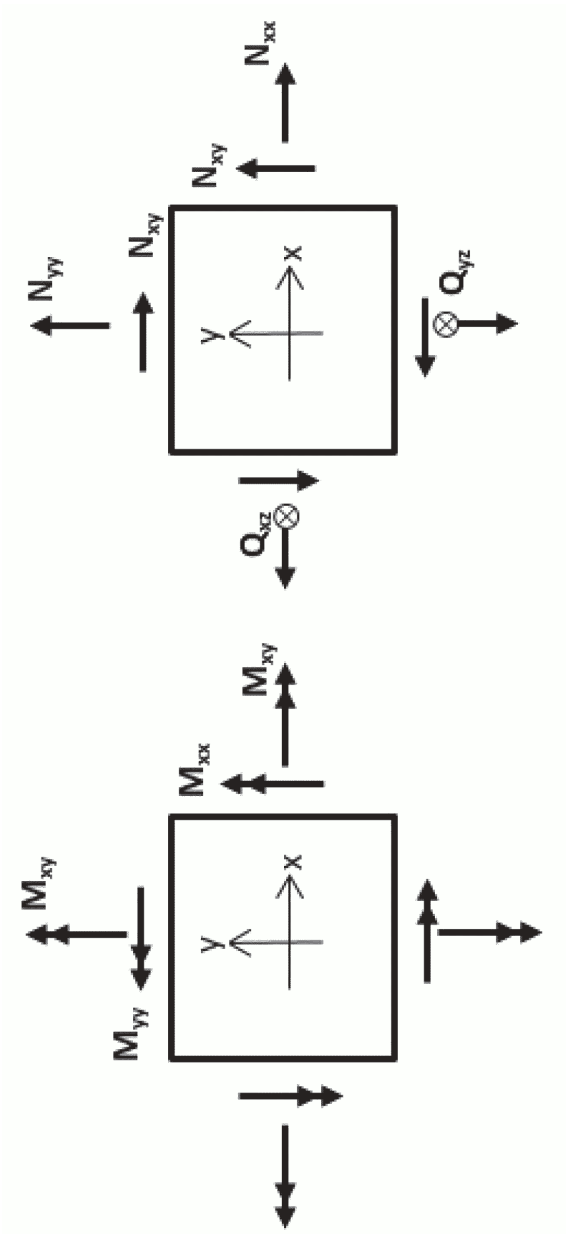


Figure 3B-1: Finite Element Forces and Moments per Unit Length, in the Local Element Coordinate System

Figure 3B-2: Force and Moment Resultants Obtained at a Section Cut, in the Section Cut Local Coordinate System

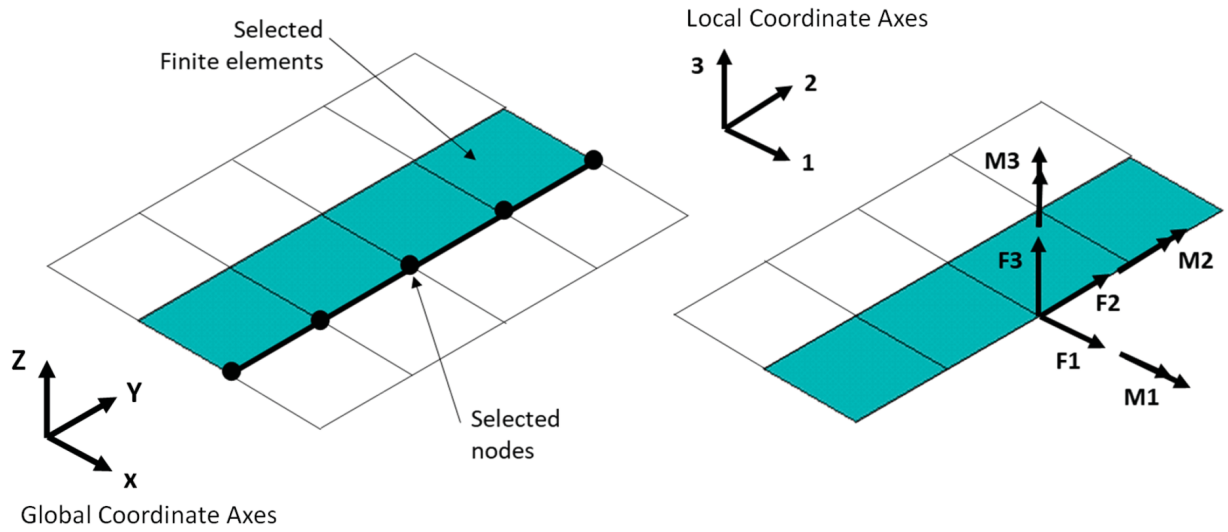


Figure 3B-3: Section Cut Locations to Determine Out-of-Plane Moment Demand in a Wall Panel due to Gravity and Frame Action

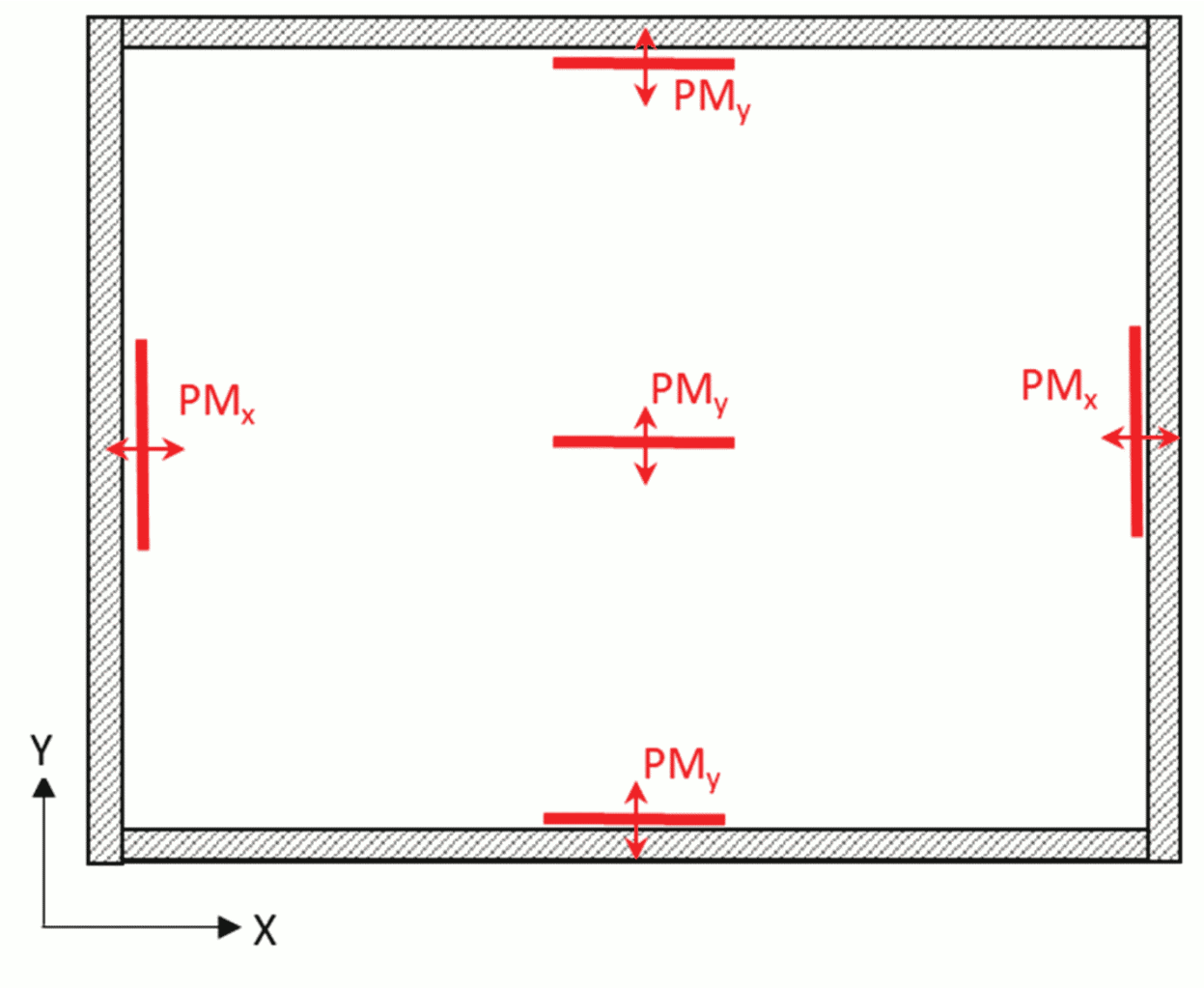
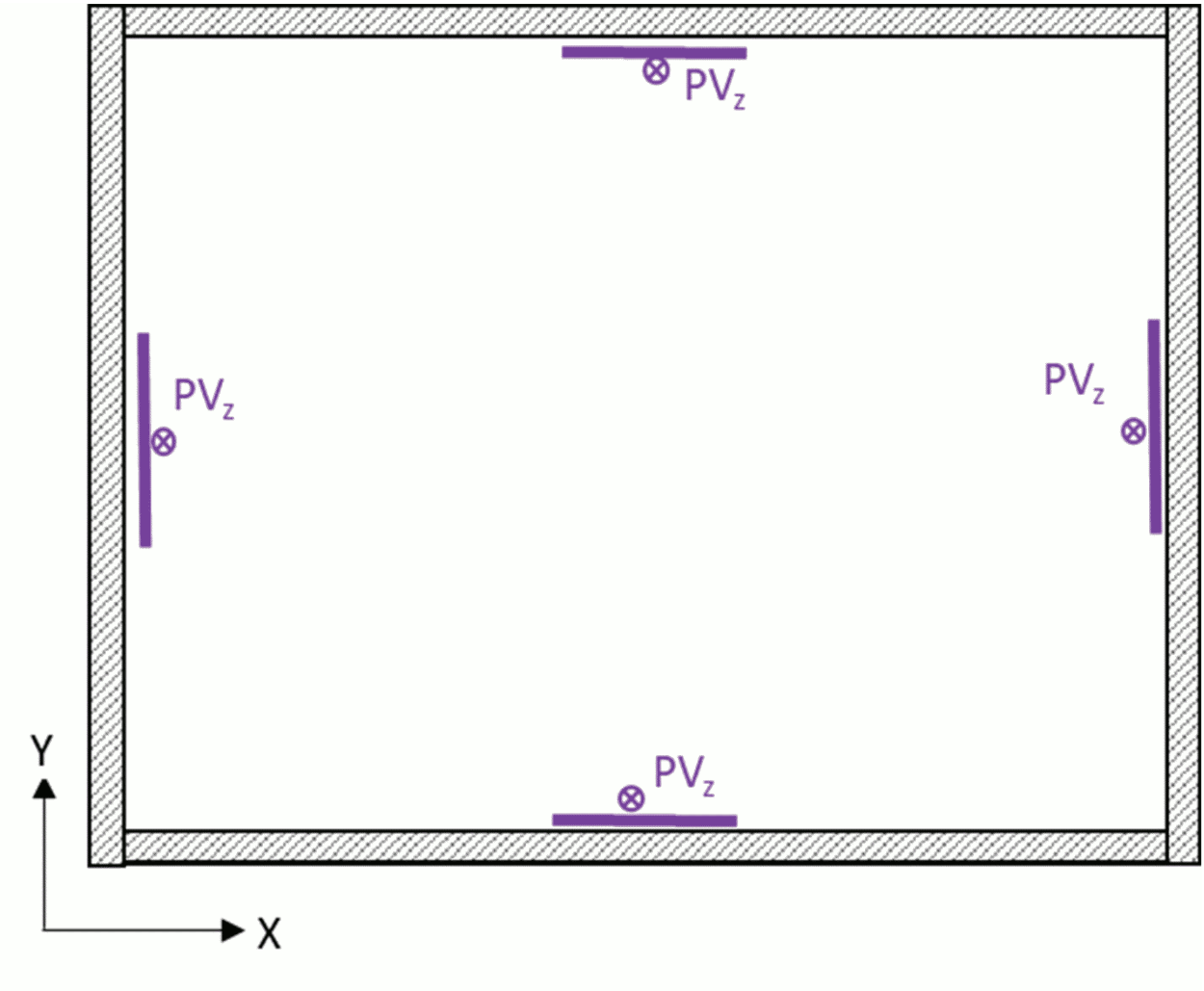
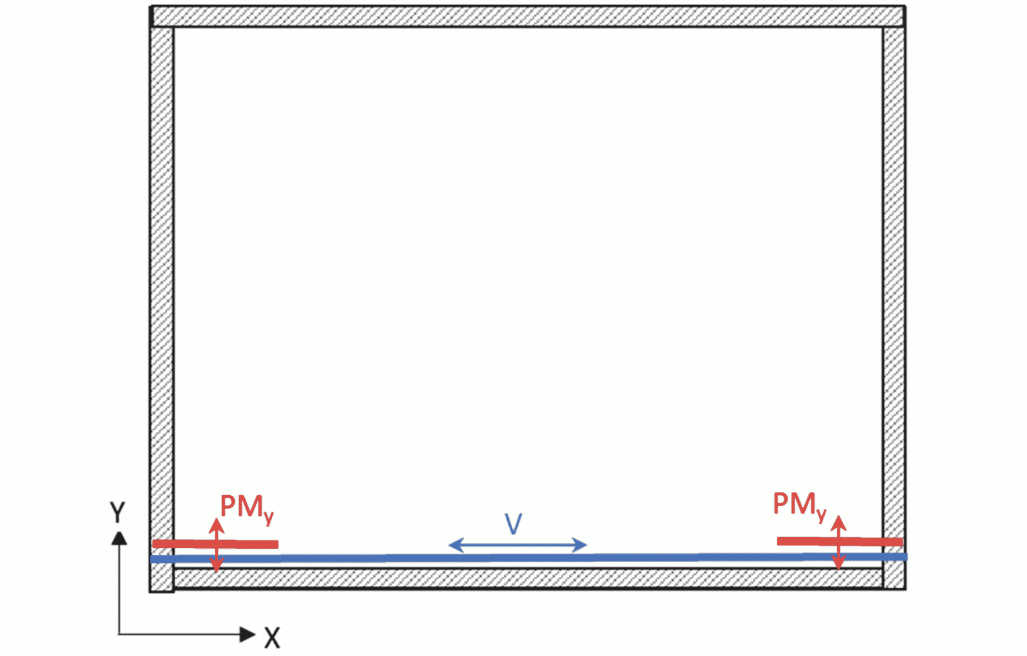


Figure 3B-4: Section Cut Locations to Determine Out-of-Plane Shear Demand in a Wall Panel due to Gravity and Frame Action

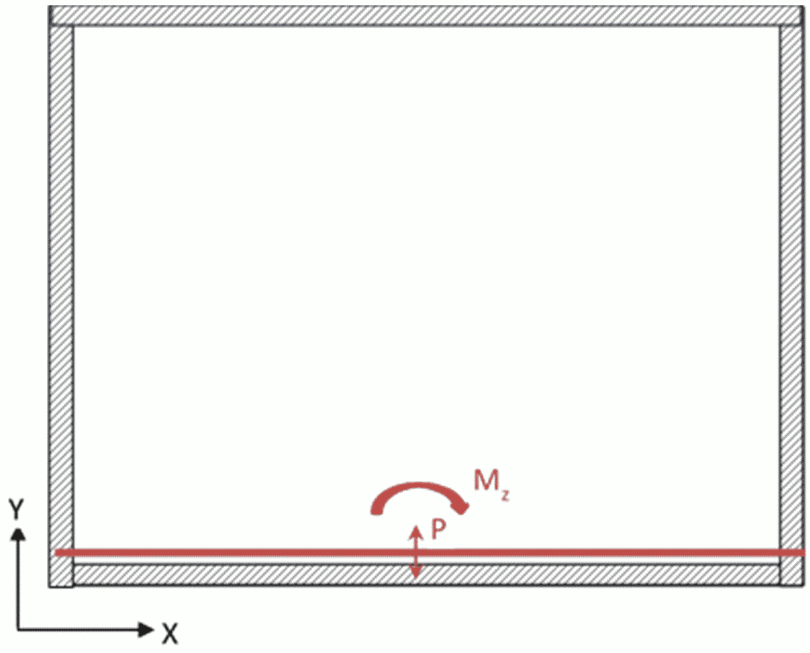


In-plane moment is considered by direct calculation of chord forces
In-plane moment is explicitly calculated at a section cut

Figure 3B-5: Section Cut Locations to Determine In-Plane Shear and In-Plane Moment in a Wall Panel

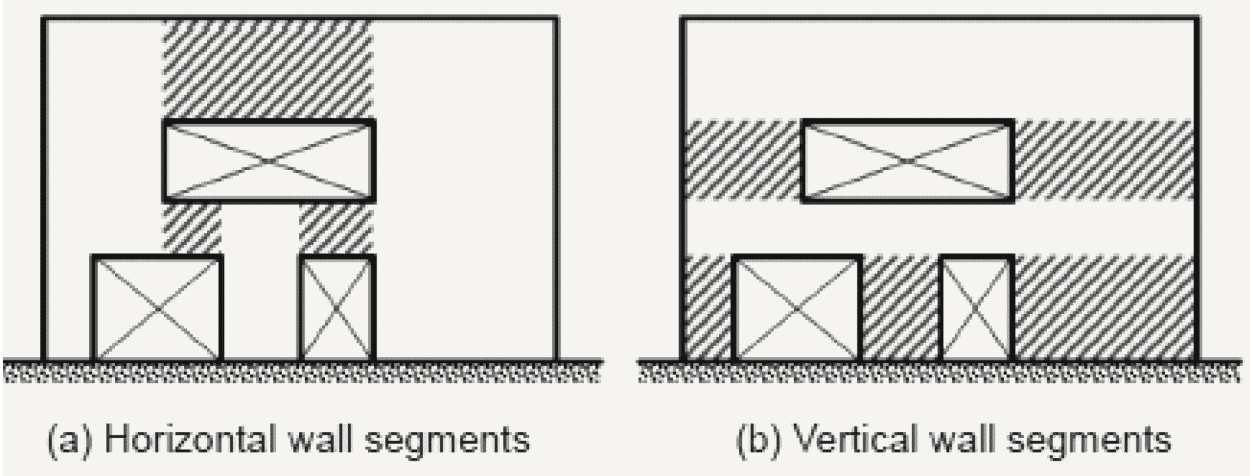


a. In-plane moment is considered by direct calculation of chord forces



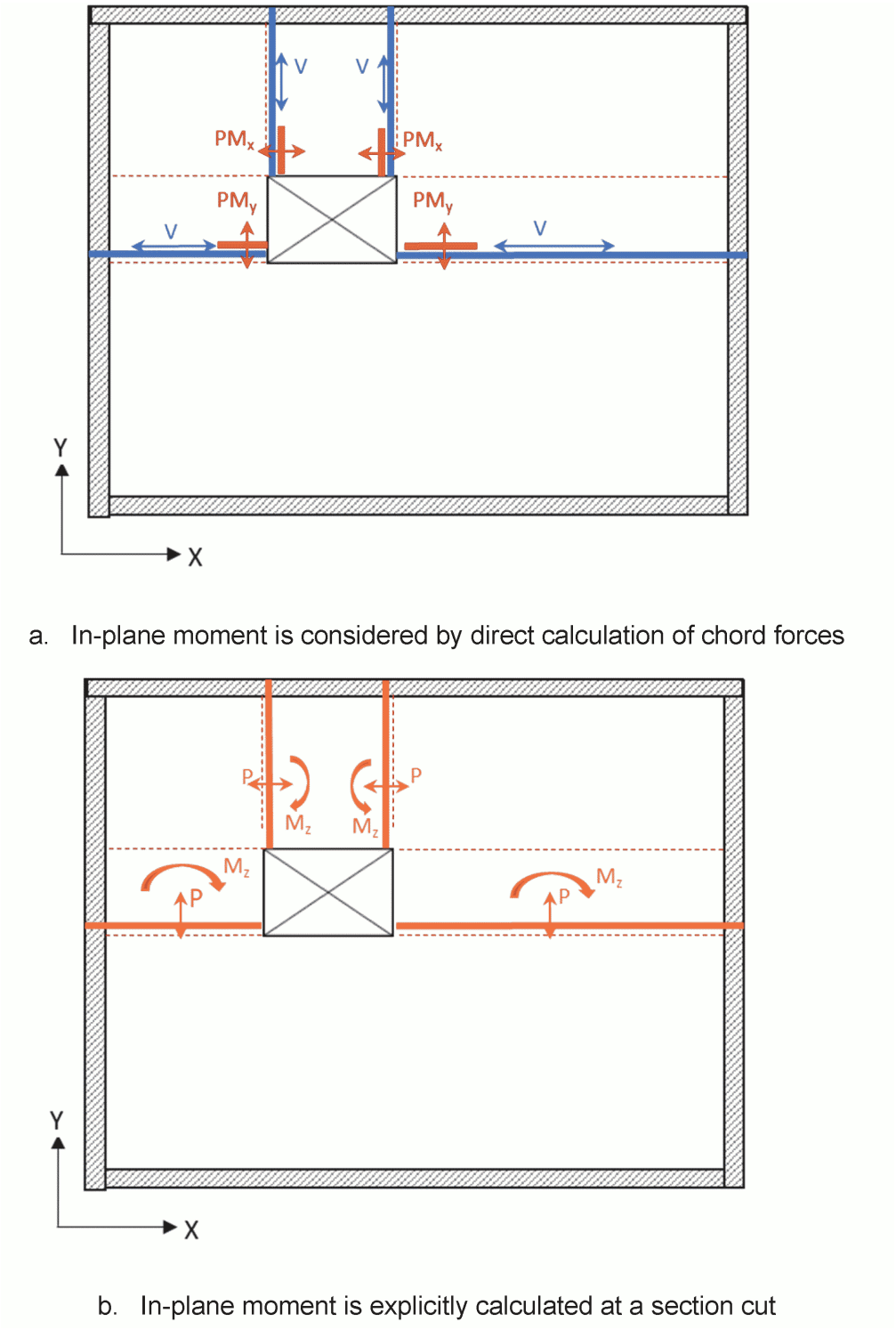
b. In-plane moment is explicitly calculated at a section cut

Figure 3B-6: Examples of Vertical and Horizontal Wall Segments (from Reference 3B-11)



In-plane moment is considered by direct calculation of chord forces
In-plane moment is explicitly calculated at a section cut

Figure 3B-7: Section cut locations to determine in-plane shear and in-plane moment in horizontal and vertical wall segments



a. In-plane moment is considered by direct calculation of chord forces

b. In-plane moment is explicitly calculated at a section cut

Figure 3B-8: Section Cut Locations to Determine Out-of-Plane Moment and Shear around an Opening in a Wall Panel

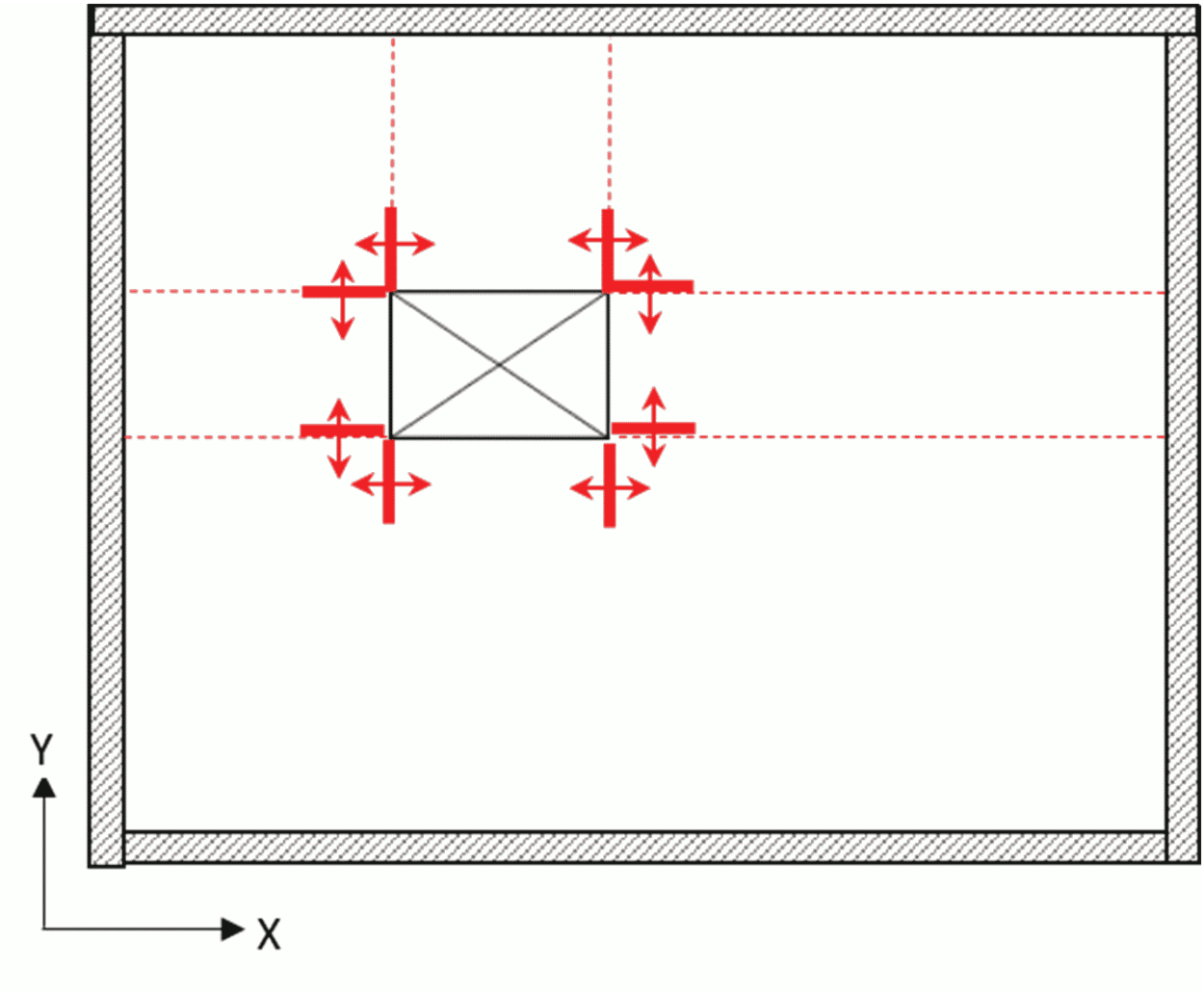
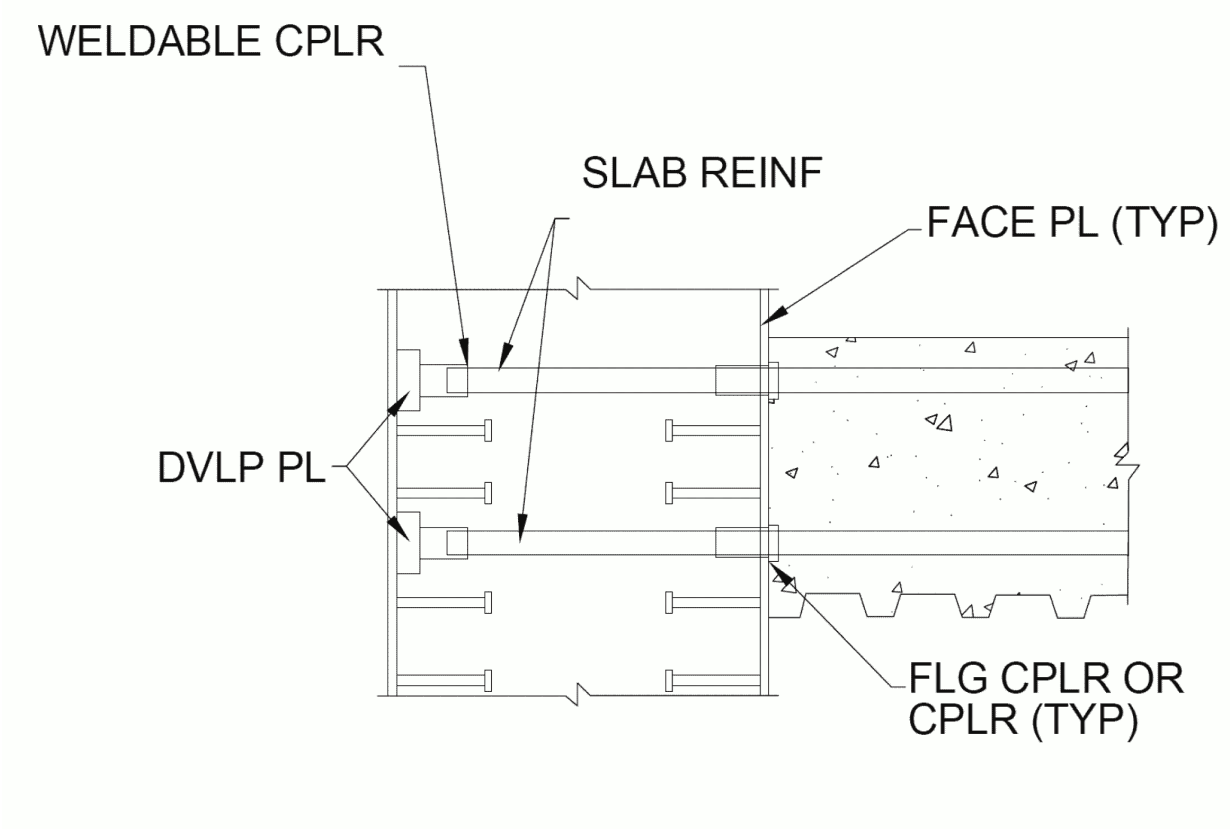


Figure 3B-9: RC Slab to SC Wall Connection Configuration - Single Layer of Rebar



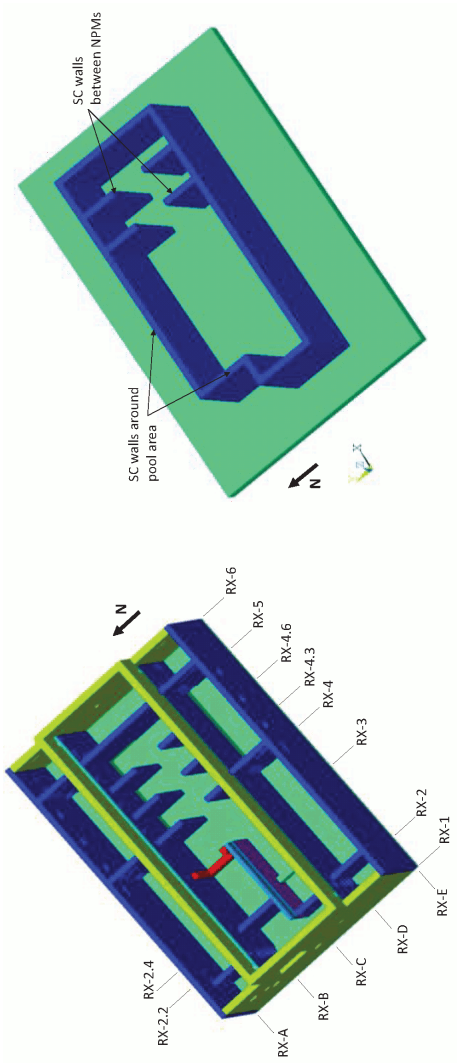


Figure 3B-10: SC Walls used in Design Calculations

Figure 3B-11: Wall RX1 Peak Contour Plots of Required Strengths for Load Combination LC9_p (force unit kip/ft and moment unit kip-in/ft)

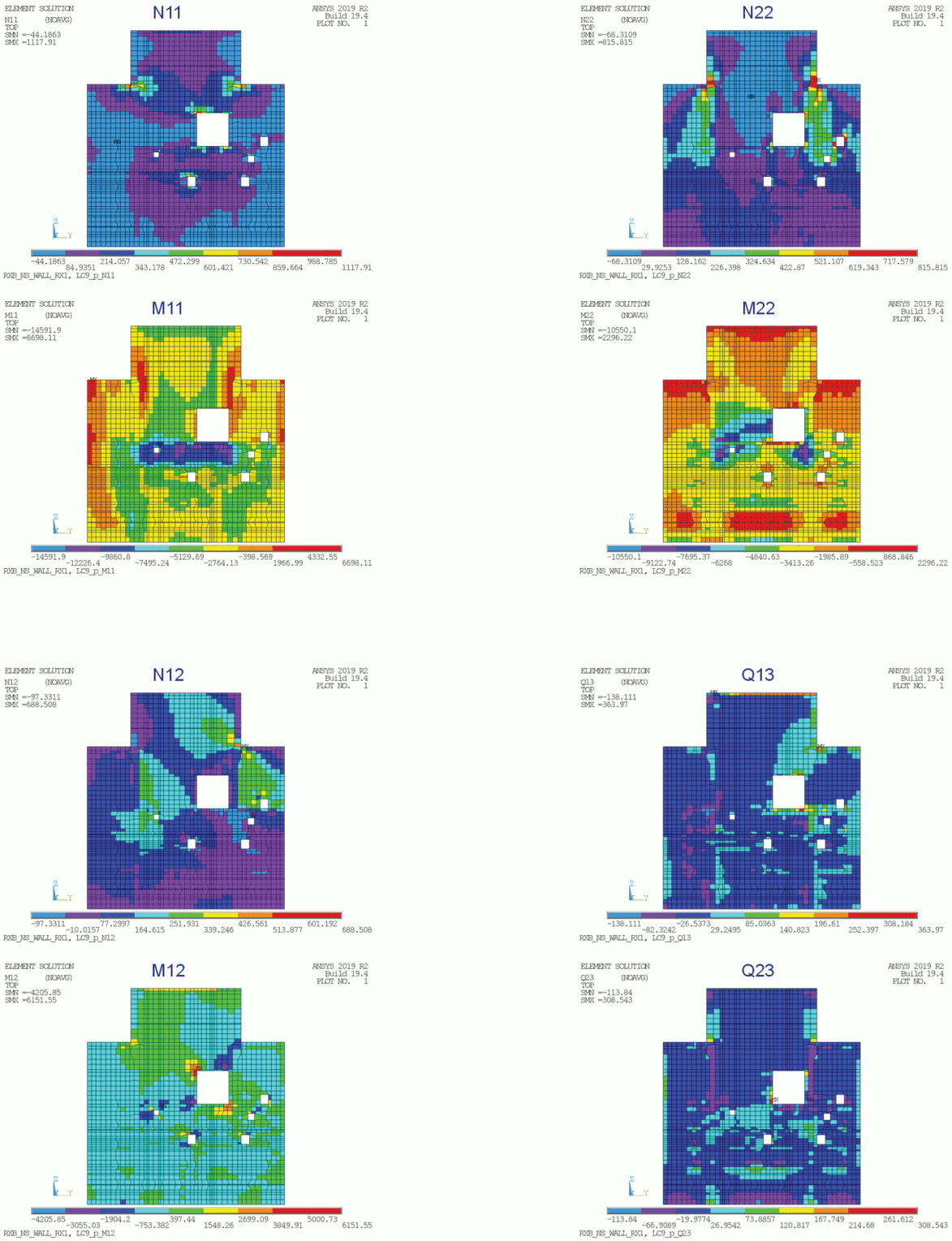


Figure 3B-12: Wall RX4 Peak Contour Plots of Required Strengths for Load Combination LC9_p (force unit kip/ft and moment unit kip-in/ft)

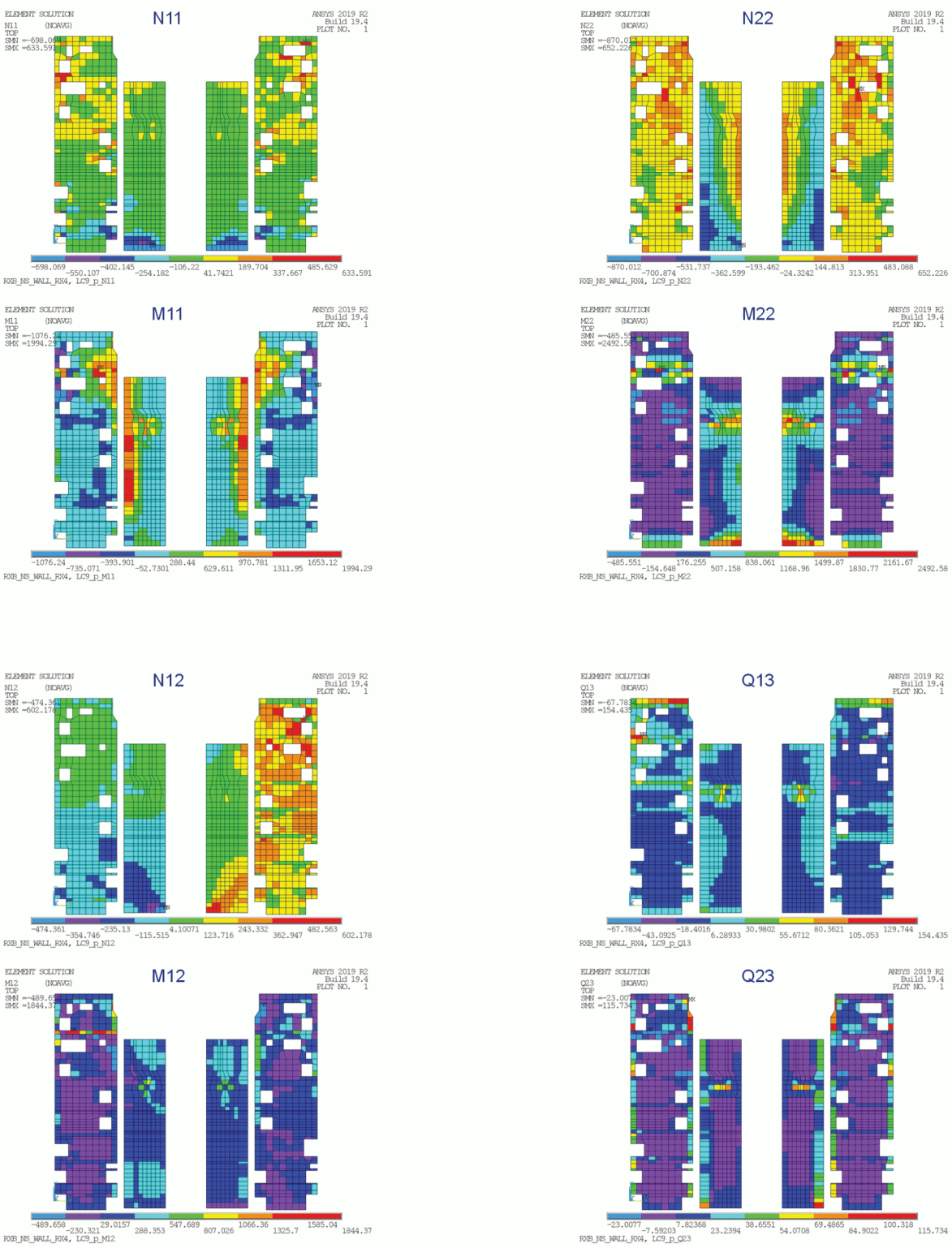


Figure 3B-13: Wall RX4.3 Peak Contour Plots of Required Strengths for Load Combinations LC9_p (force unit kip/ft and moment unit kip-in/ft)

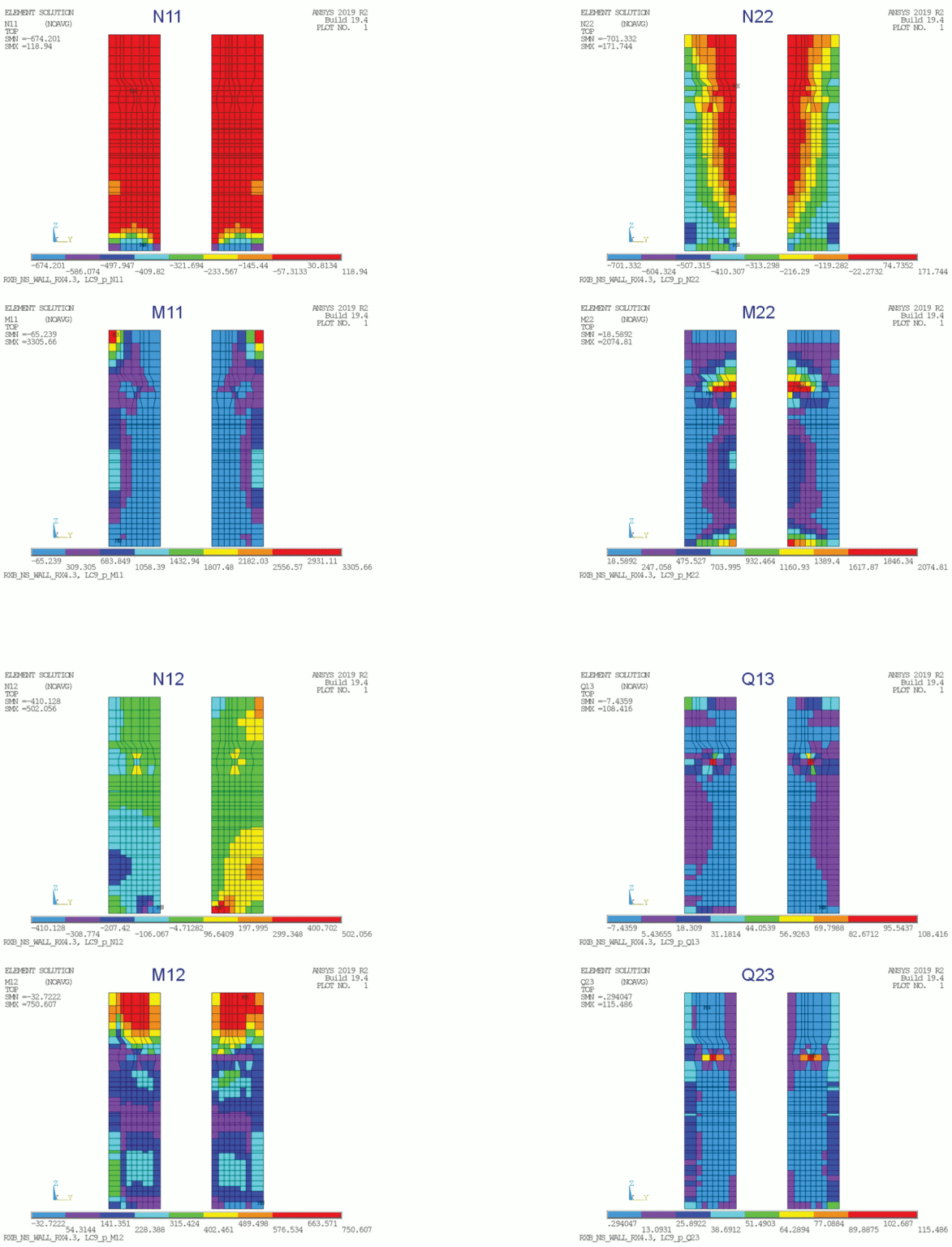


Figure 3B-14: Wall RX4.6 Peak Contour Plots of Required Strengths for Load Combination LC9_p (force unit kip/ft and moment unit kip-in/ft)

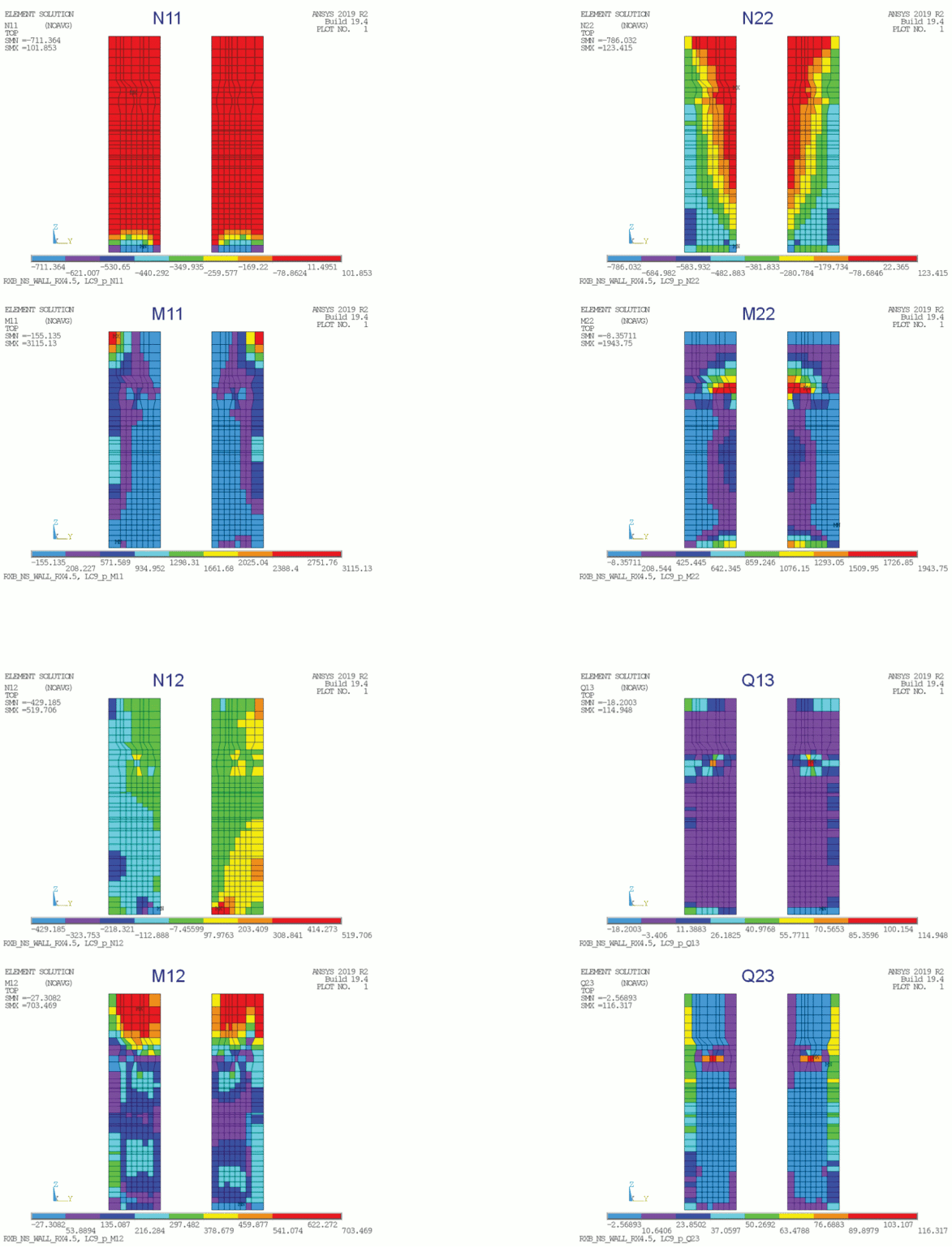


Figure 3B-15: Wall RXB Peak Contour Plots of Required Strengths for Load Combination LC9_p (force unit kip/ft and moment unit kip-in/ft)

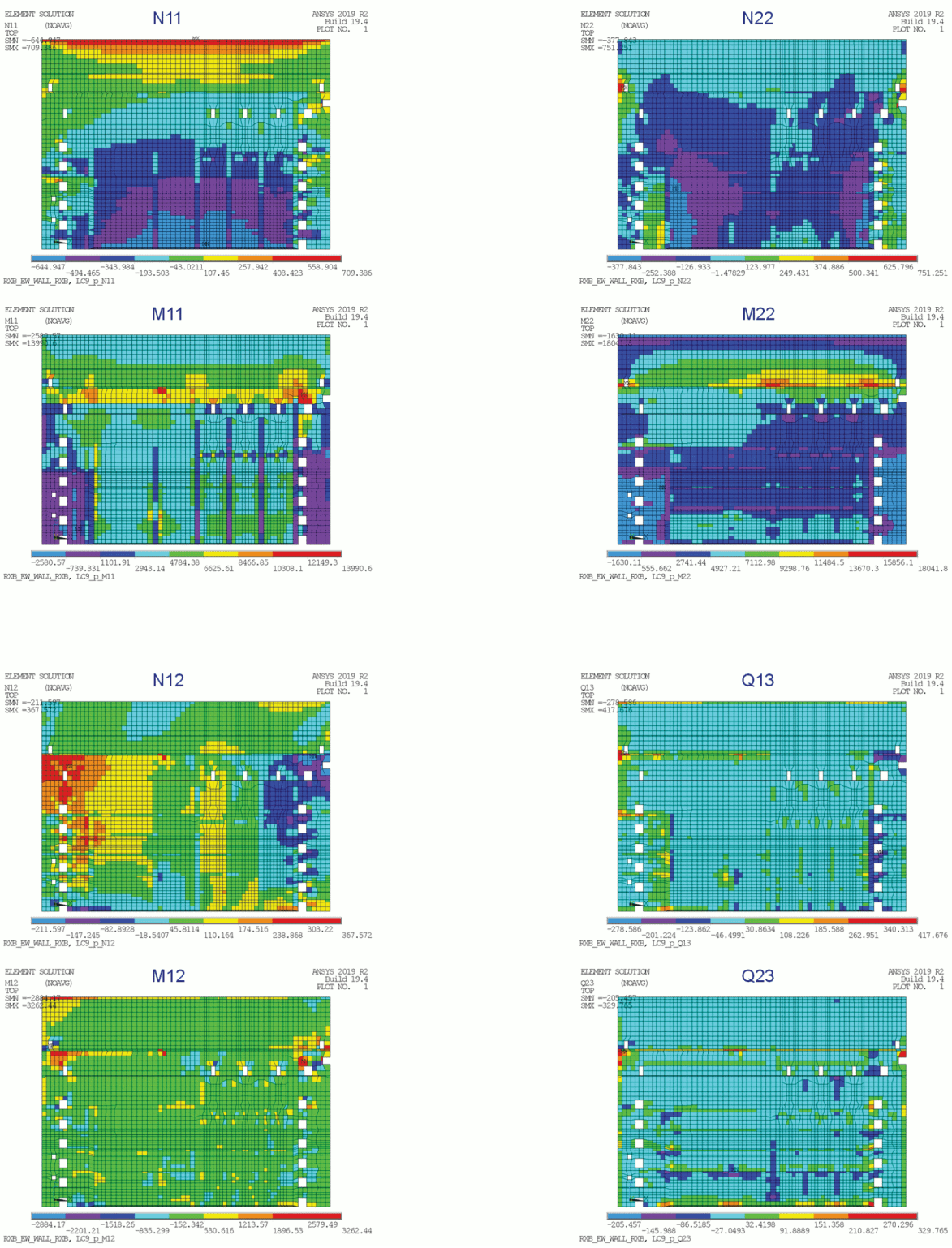
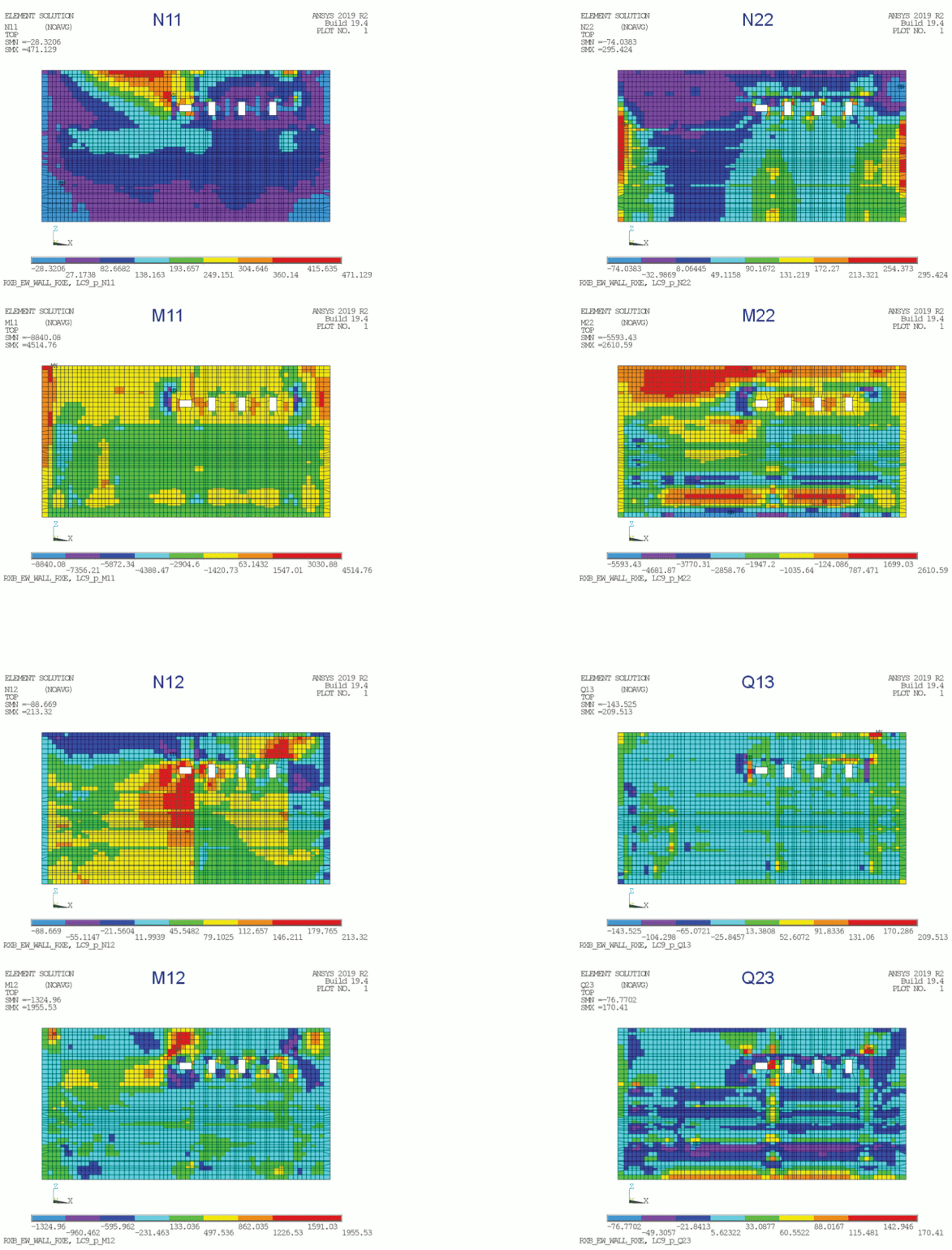


Figure 3B-16: Wall RXE Peak Contour Plots of Required Strengths for Load Combination LC9_p (force unit kip/ft and moment unit kip-in/ft)



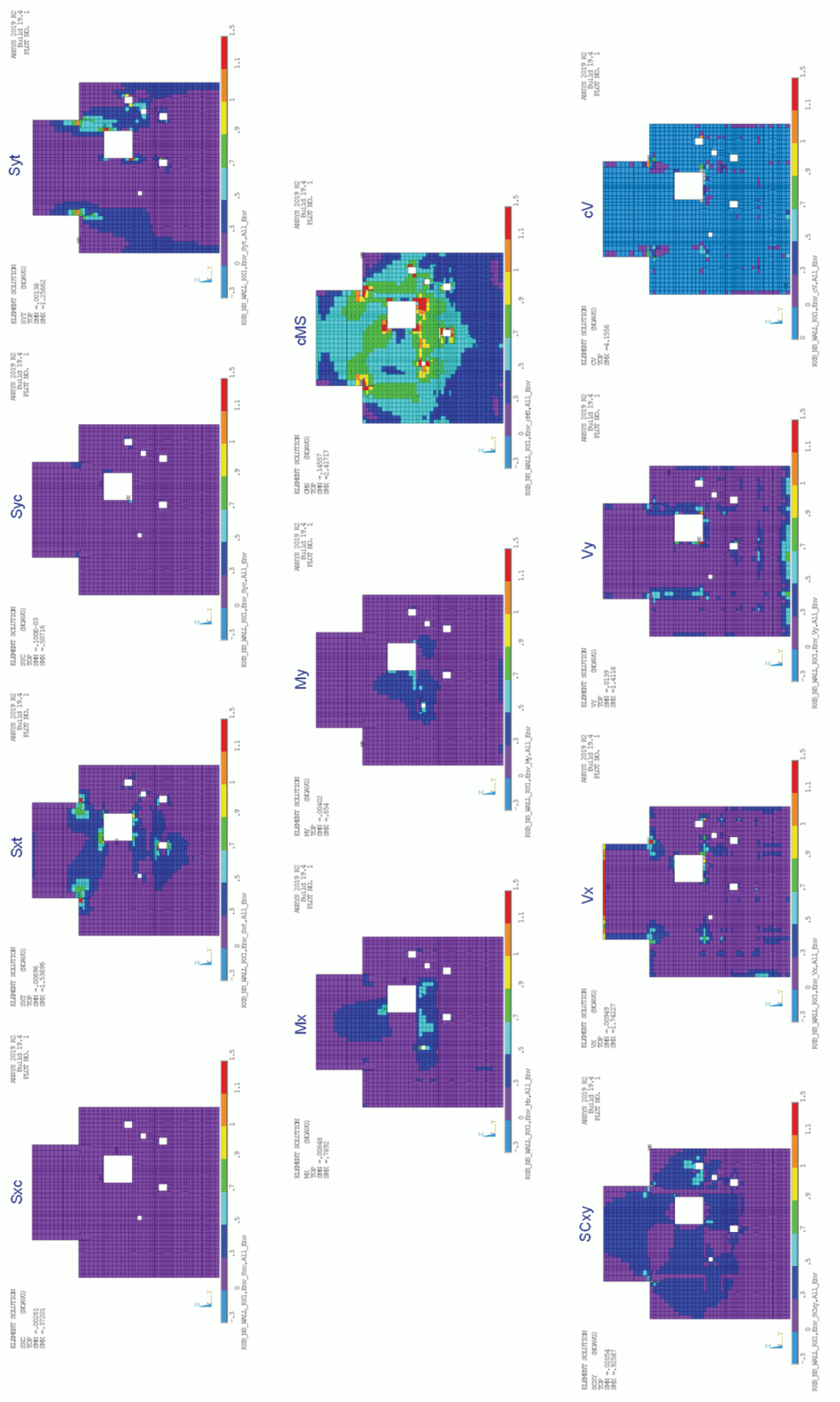
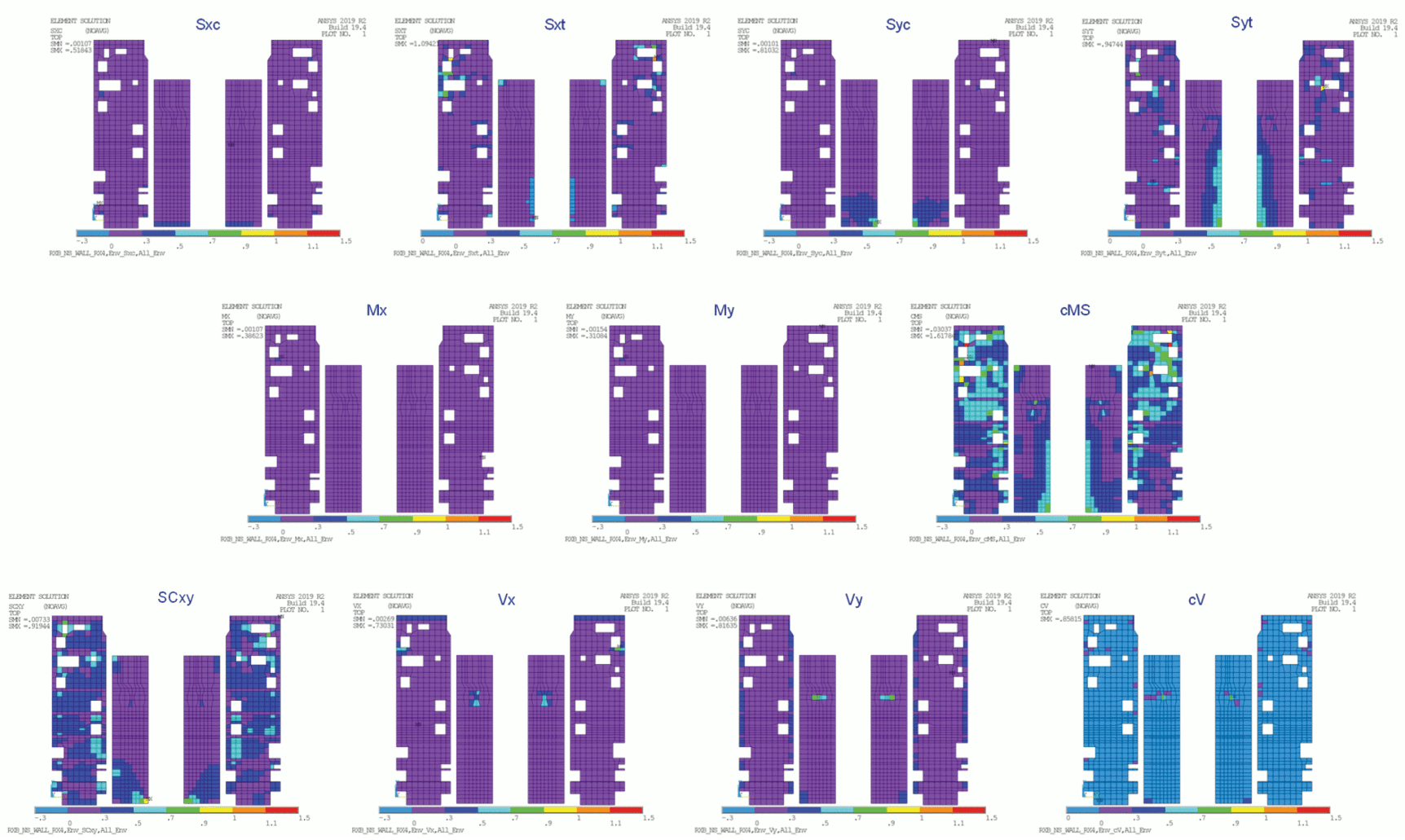


Figure 3B-17: Wall RX1 Enveloped DCR Contour Plots for All Design Conditions

Figure 3B-18: Wall RX4 Enveloped DCR Contour Plots for All Design Conditions



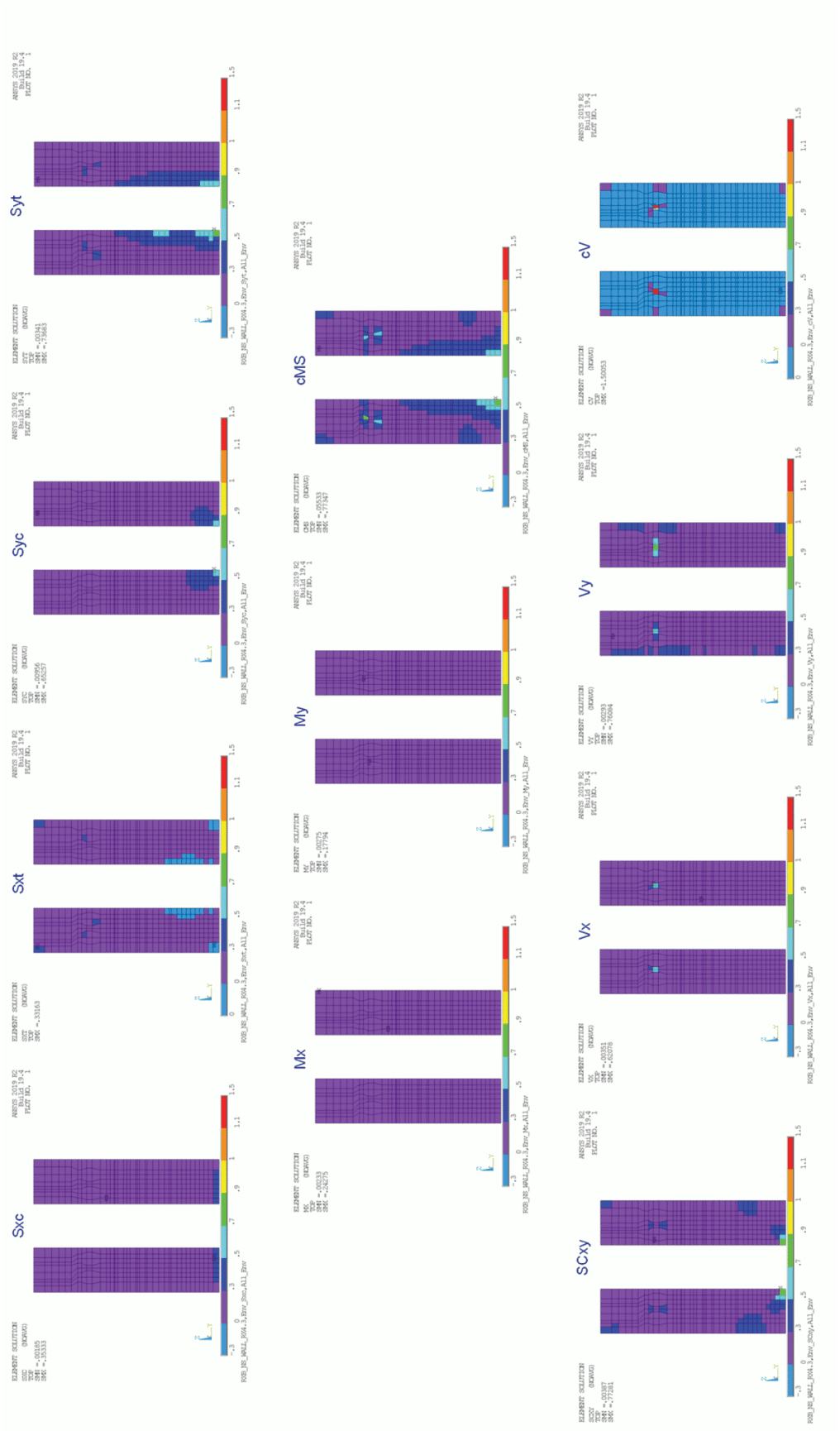


Figure 3B-19: Wall RX4.3 Enveloped DCR Contour Plots for All Design Conditions

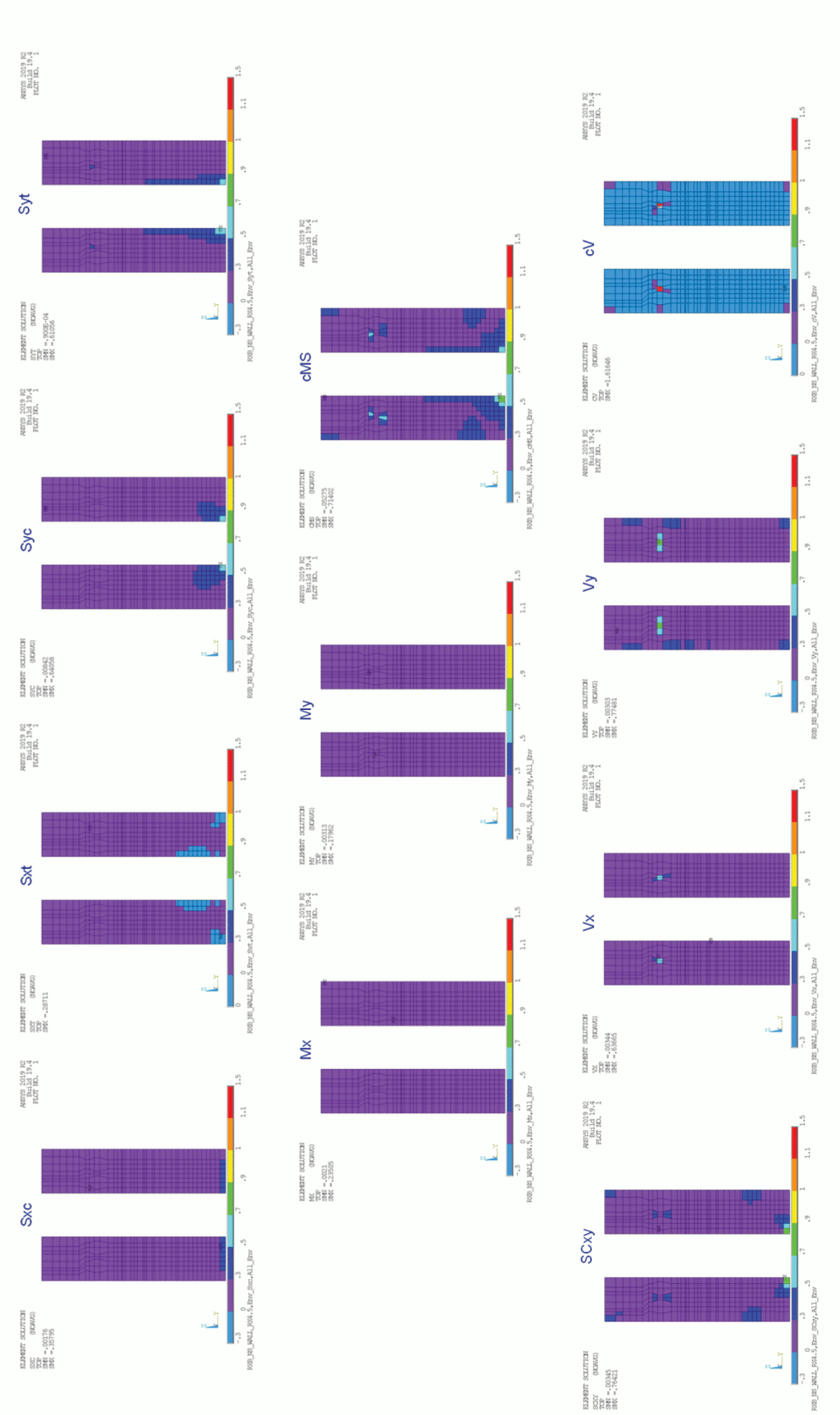


Figure 3B-20: Wall RX4.6 Enveloped DCR Contour Plots for All Design Conditions

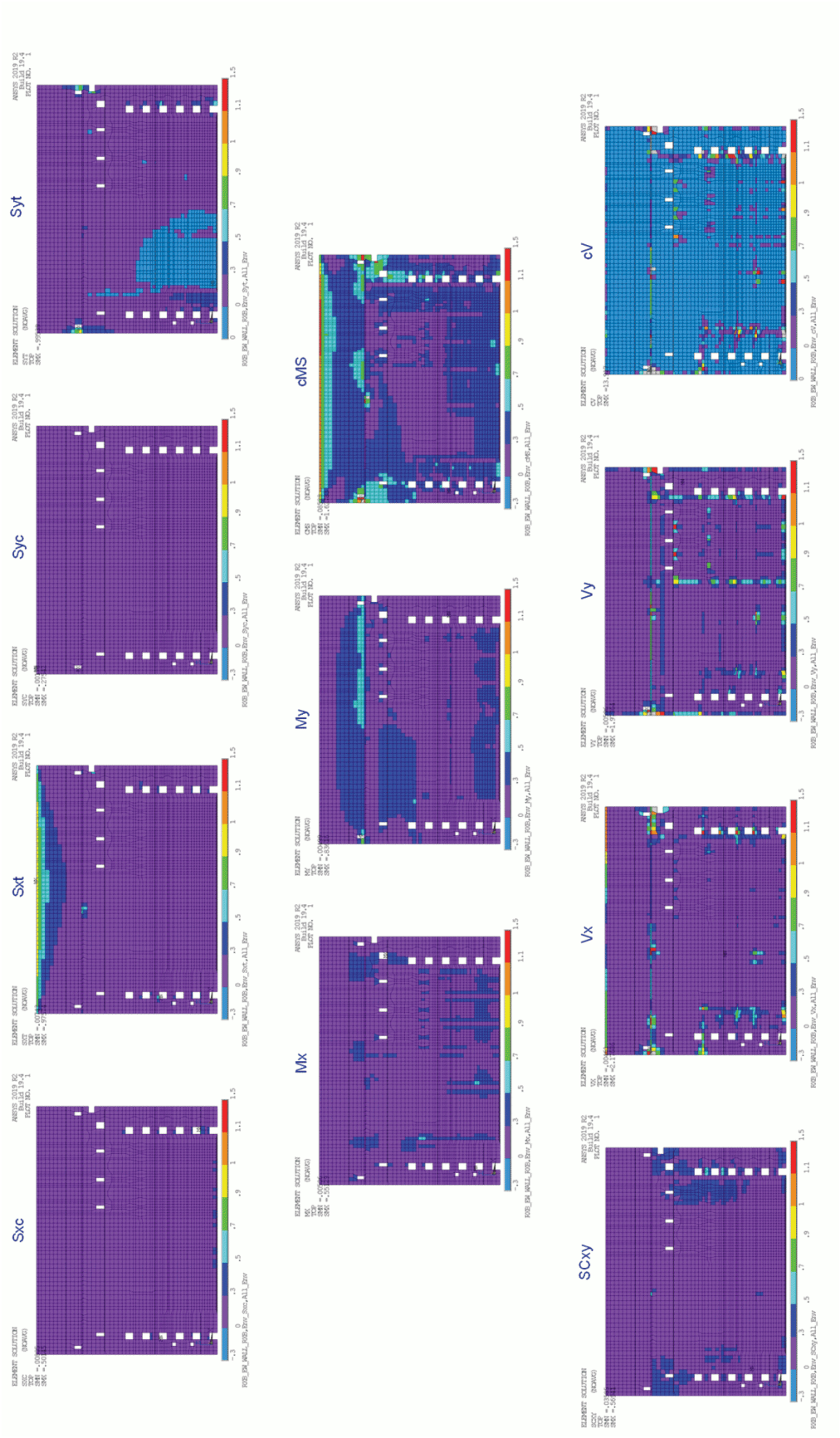


Figure 3B-21: Wall RXB Enveloped DCR Contour Plots for All Design Conditions

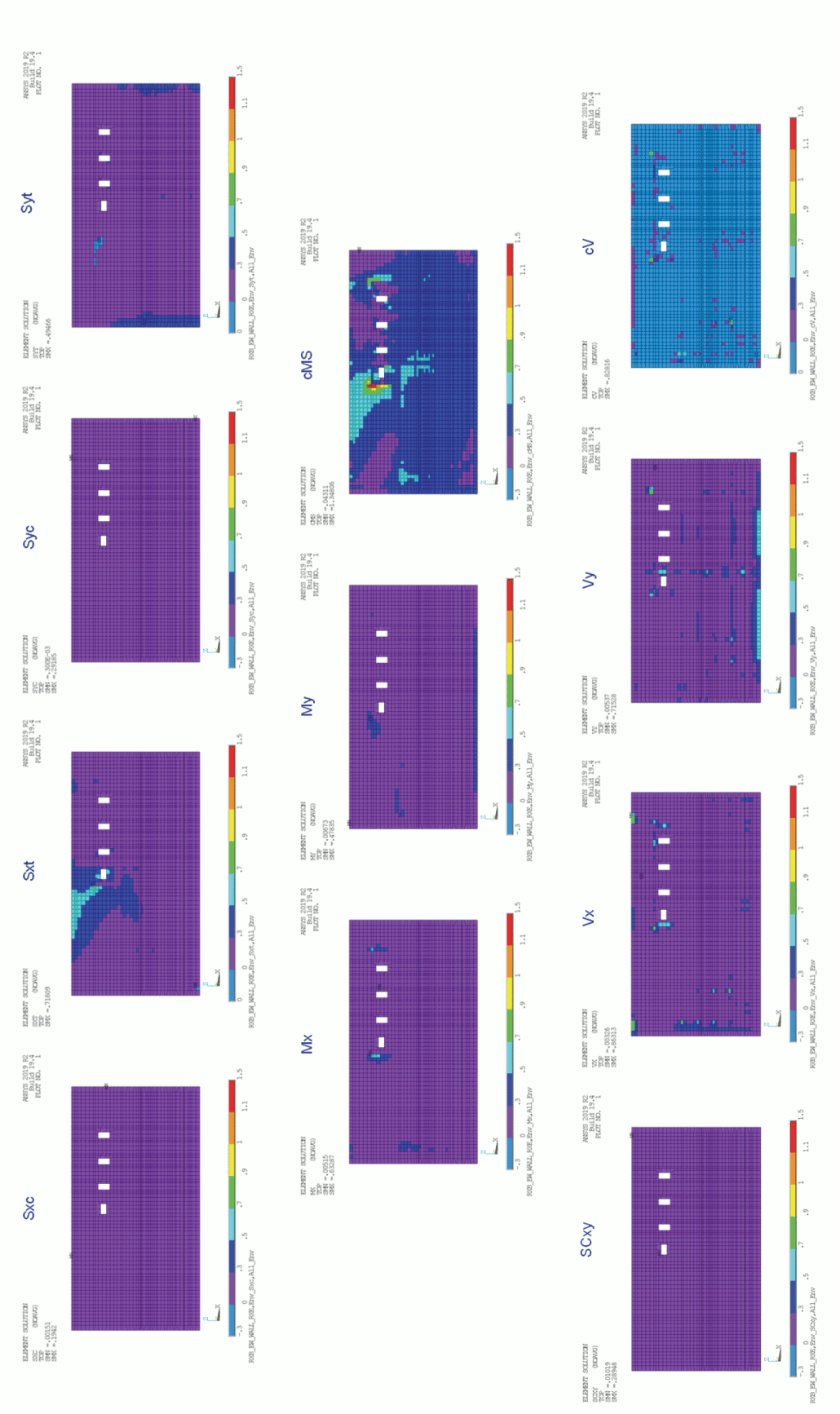


Figure 3B-22: Wall RXE Enveloped DCR Contour Plots for All Design Conditions

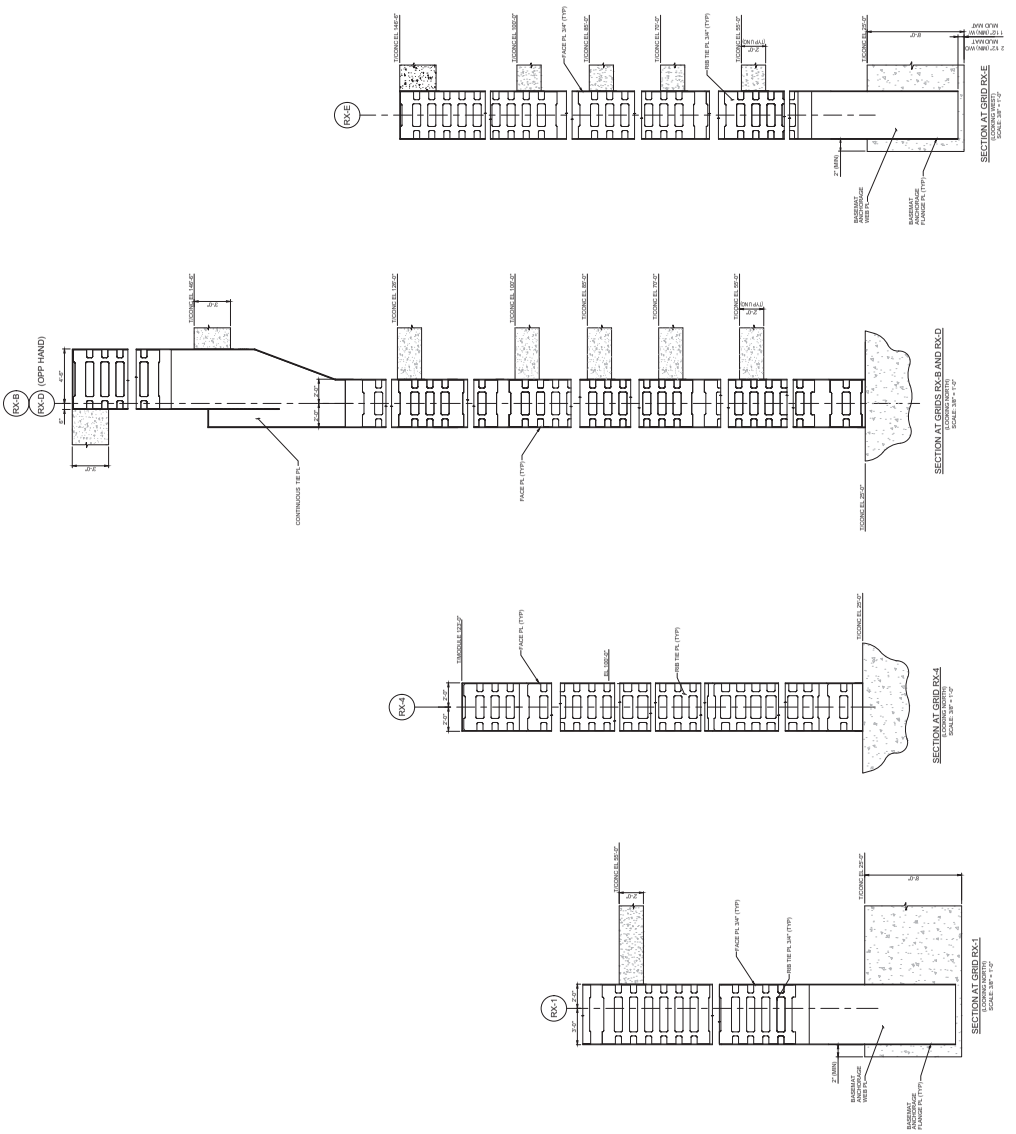


Figure 3B-23: Steel-Composite Wall Critical Sections

Figure 3B-24: Basemat Peak Contour Plots of Combined Demands for Load Combination LC6_p (force unit kip/ft and moment unit kip-in./ft)

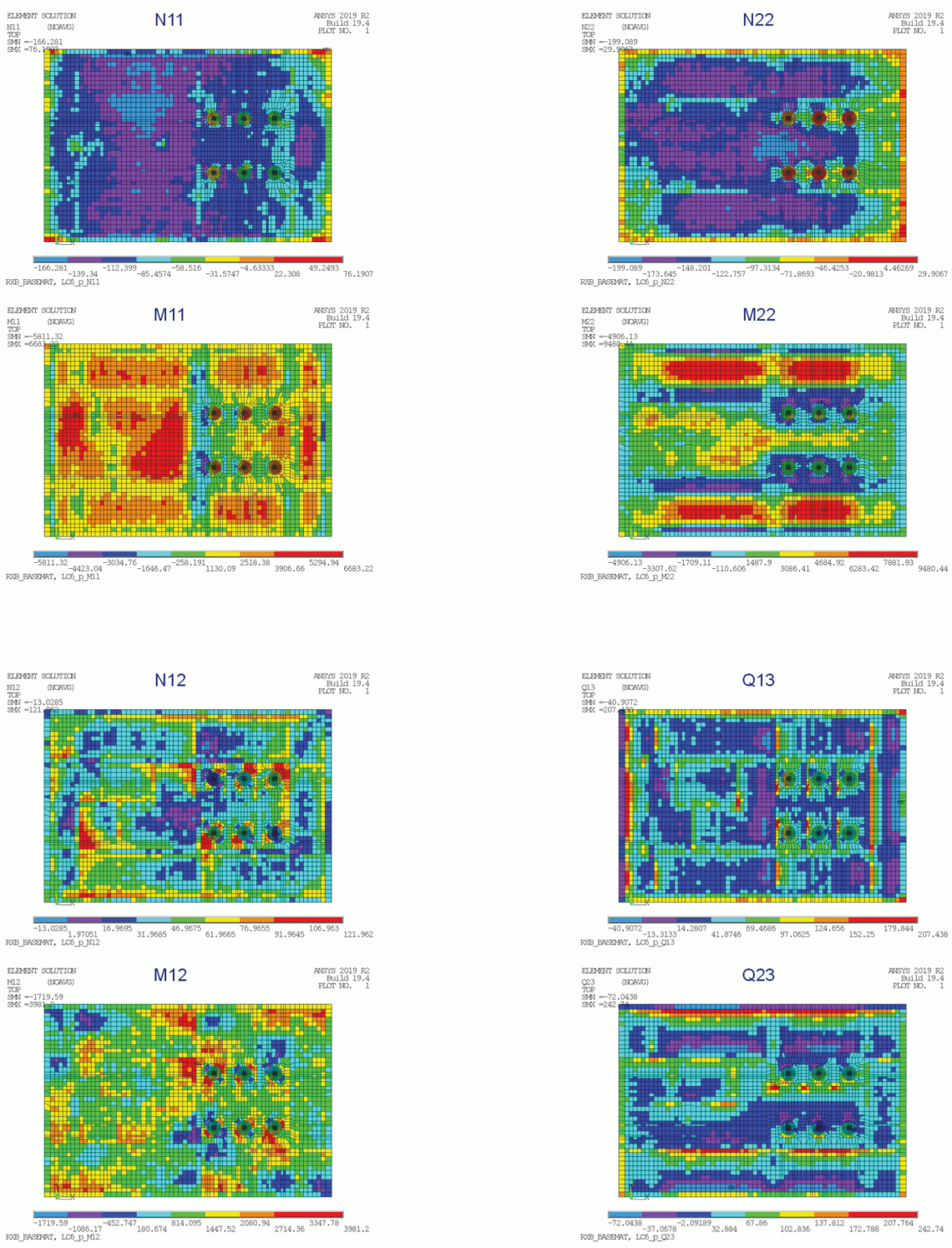


Figure 3B-25: Floor Slab at EL 100’ Peak Contour Plots of Combined Demands for Load Combination LC6_p (force unit kip/ft and moment unit kip-in./ft)

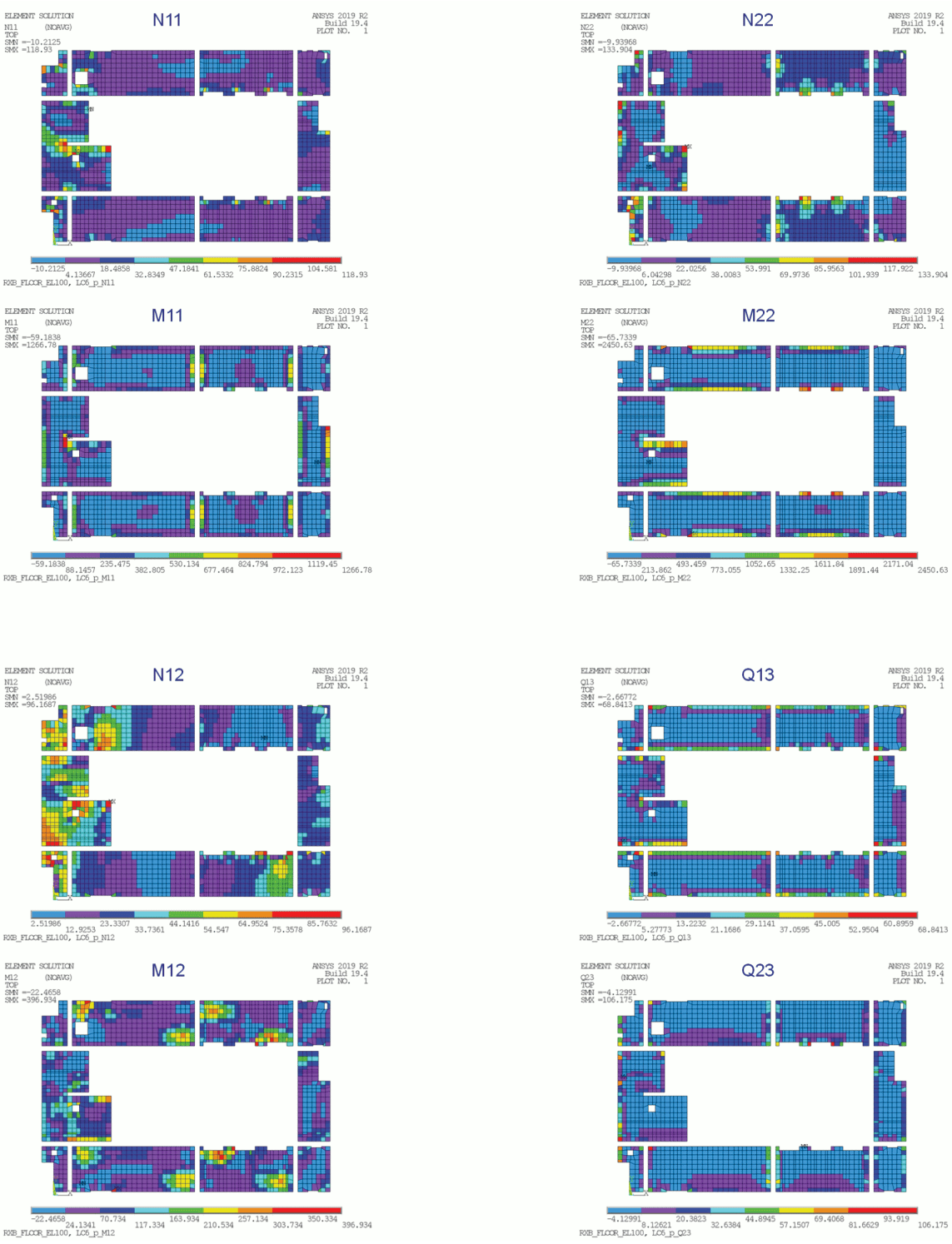
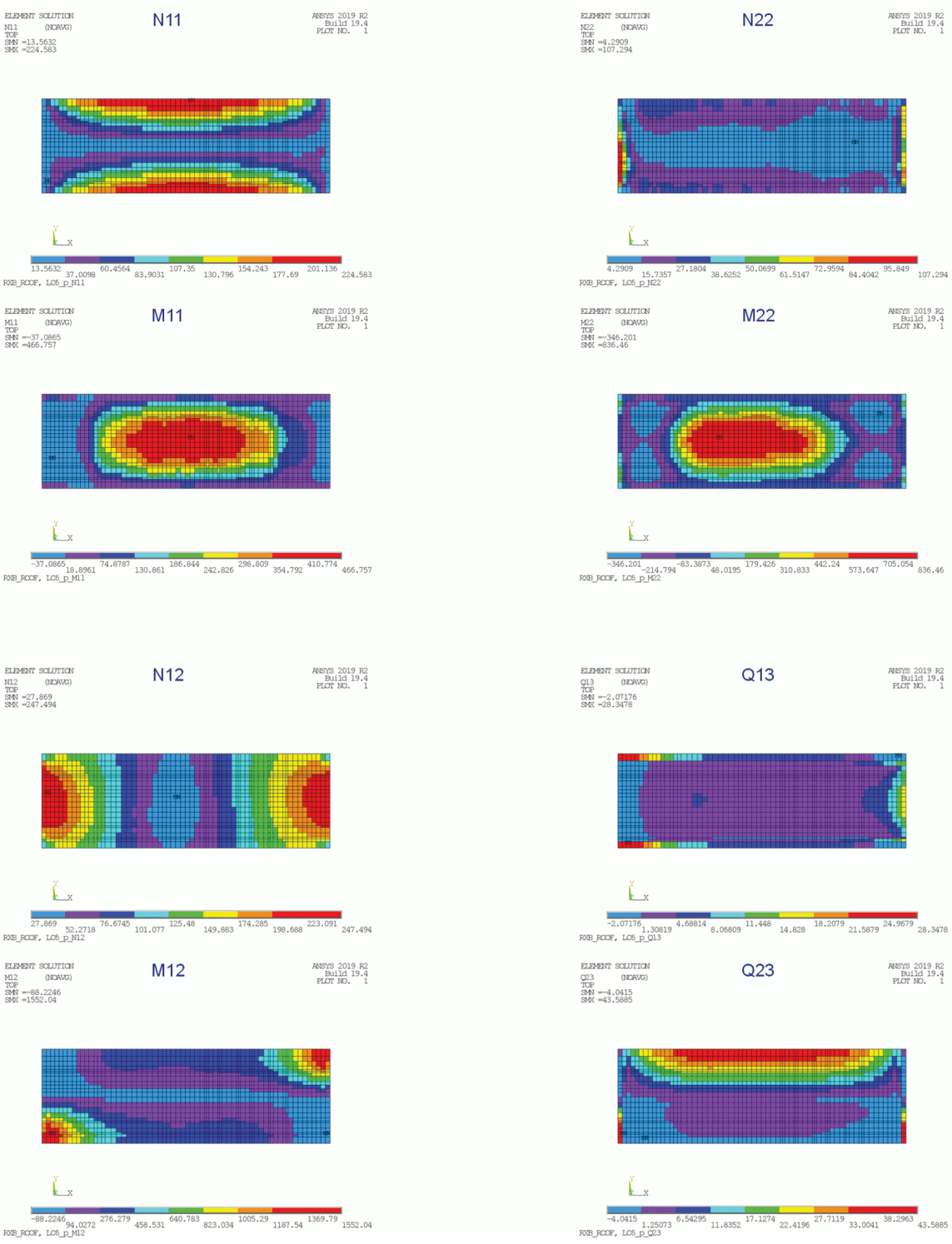
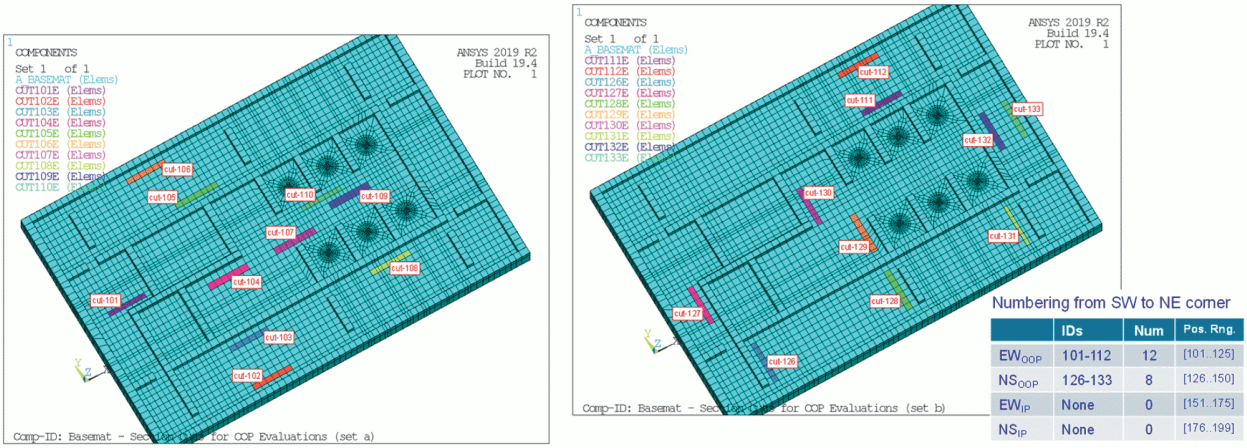


Figure 3B-26: Roof Peak Contour Plots of Combined Demands for Load Combination LC6_p (force unit kip/ft and moment unit kip-in./ft)



33

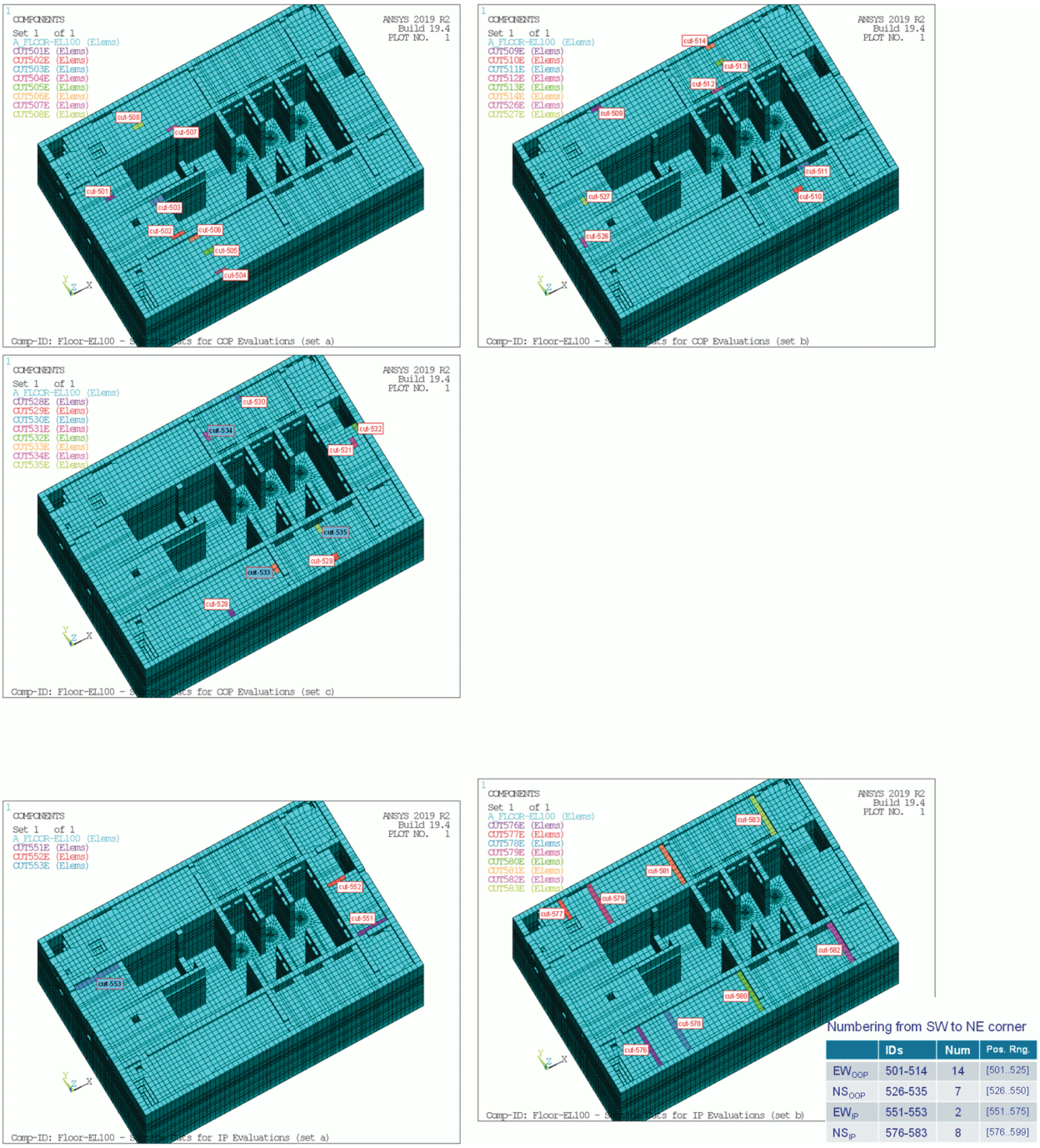
Figure 3B-27: Section Cuts used in Design Calculations of Basemat



Numbering from SW to NE corner

	IDs	Num	Pos. Rng.
EW_{OOP}	101-112	12	[101..125]
NS_{OOP}	126-133	8	[126..150]
EW_{IP}	None	0	[151..175]
NS_{IP}	None	0	[176..199]

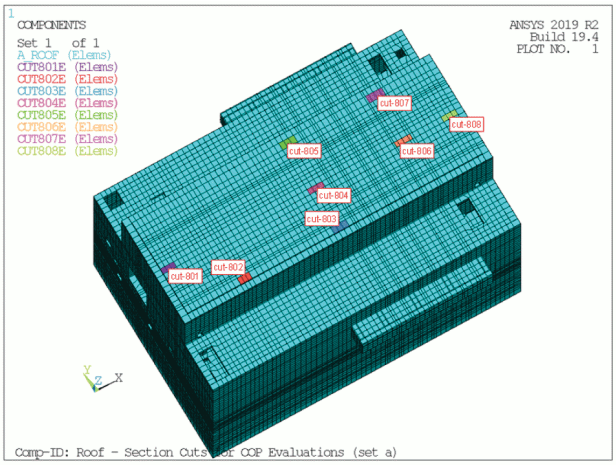
Figure 3B-28: Section Cuts used in Design Calculations of Floor Slab at EL 100 feet

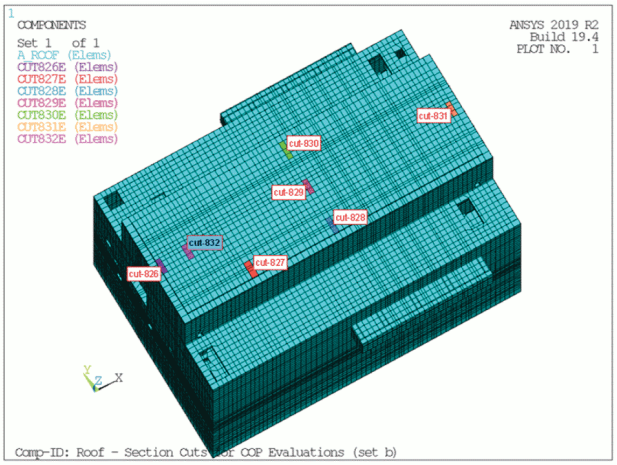


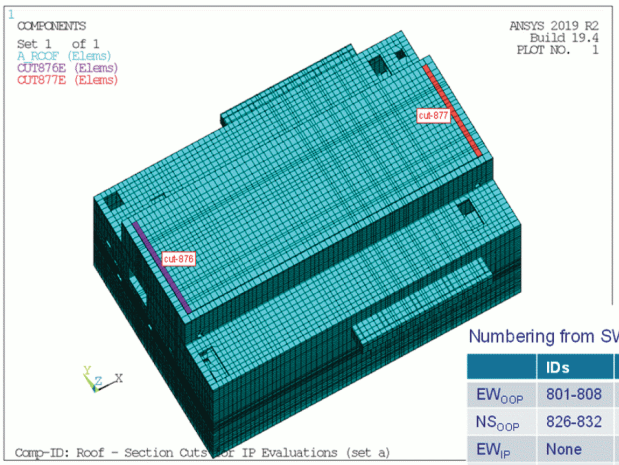
Numbering from SW to NE corner

	IDs	Num	Pos. Rng.
EW_{OOP}	501-514	14	[501..525]
NS_{OOP}	526-535	7	[526..550]
EW_{IP}	551-553	2	[551..575]
NS_{IP}	576-583	8	[576..599]

Figure 3B-29: Section Cuts used in Design Calculations of Roof Slab







Numbering from SW to NE corner

	IDs	Num	Pos. Rng.
EW_{OOP}	801-808	8	[801..825]
NS_{OOP}	826-832	6	[826..850]
EW_{IP}	None	0	[851..875]
NS_{IP}	876-877	2	[876..899]

Figure 3B-30: Reinforcement Layout of Reactor Building basemat. Any reinforcement that is different than the typical reinforcement is shown on the plan view.

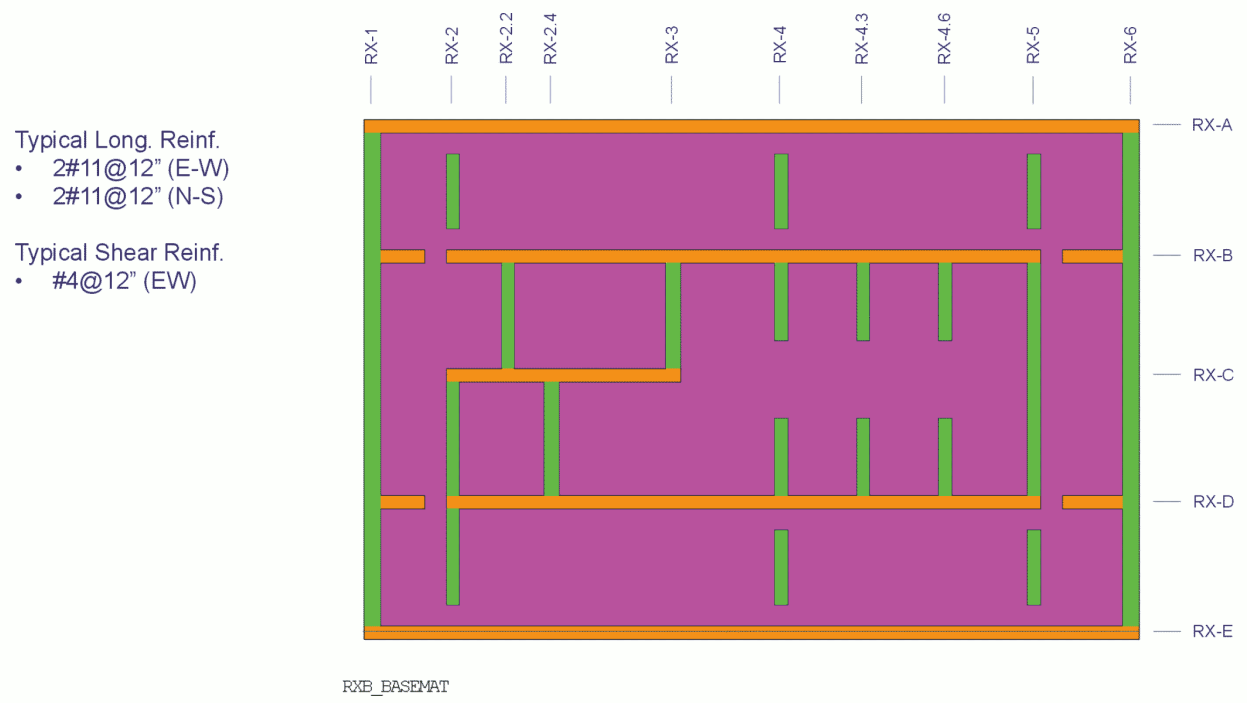


Figure 3B-31: Reinforcement Layout of Reactor Building Floor Slab at EL 100 feet. Any reinforcement that is different than the typical reinforcement is shown on the plan view.

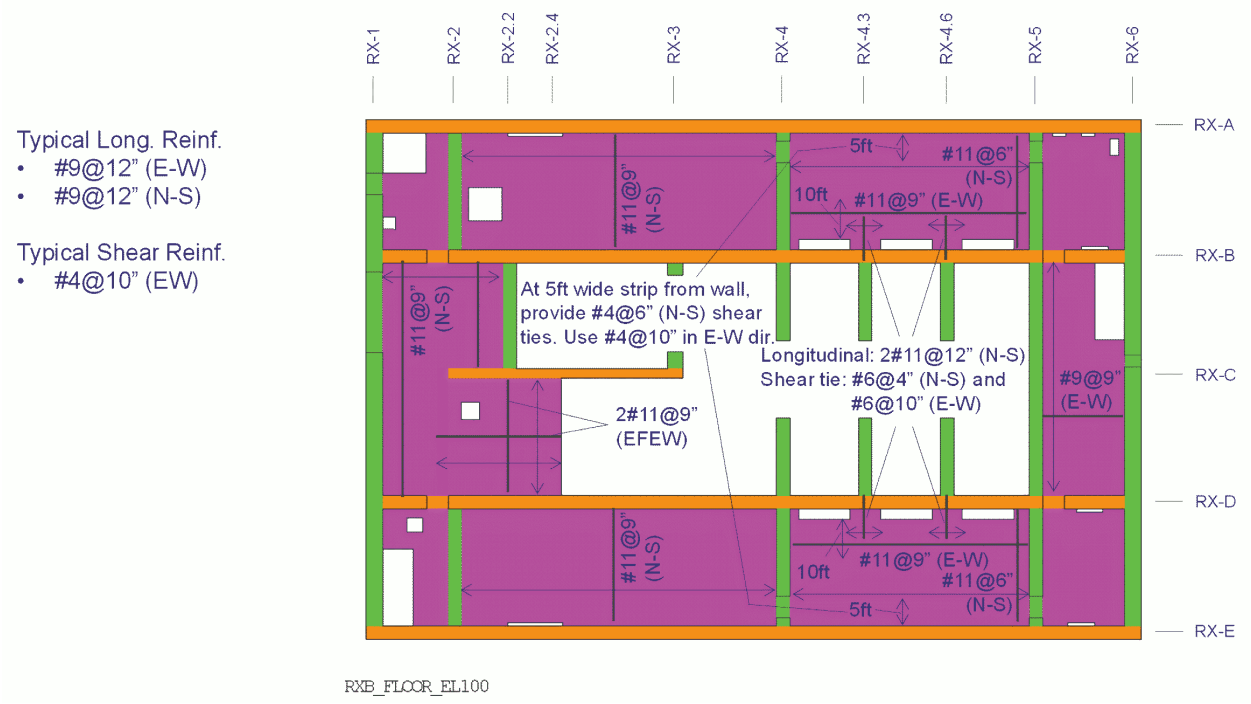


Figure 3B-31a: Reinforcement Layout of Reactor Building Roof Slab. Any reinforcement that is different than the typical reinforcement is shown on the plan view.

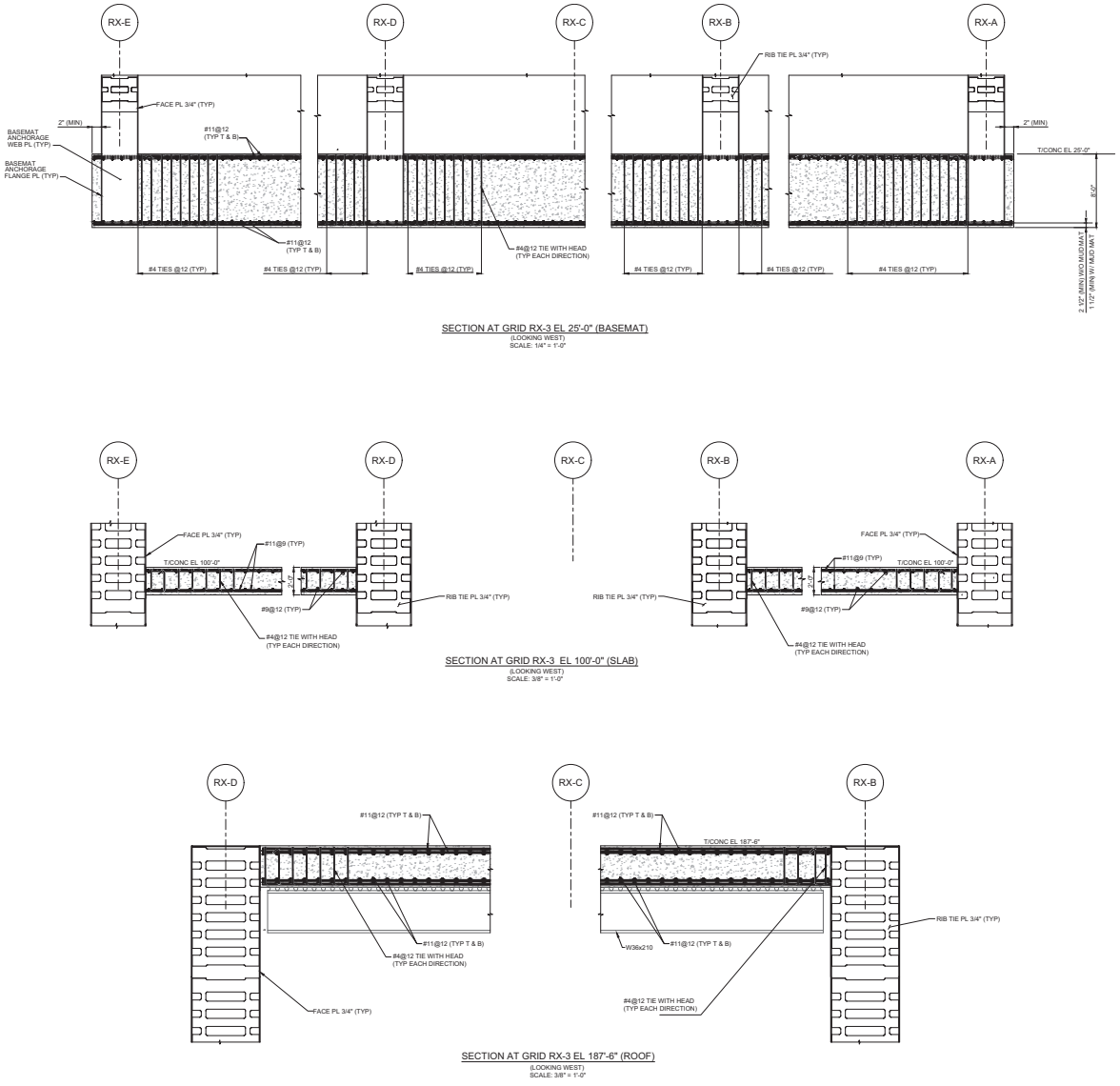
Typical Long. Reinf.
- 2#11@12” (E-W)
- 2#11@12” (N-S)

Typical Shear Reinf.
- #4@12” (EW)



RXB_ROOF

Figure 3B-32: Typical Reinforcement Layout of Reactor Building Basemat, Floor Slab at EL 100 ft, and the Roof Slab



Note: Any reinforcement that is different than the typical reinforcement is shown on the plan view.

Figure 3B-33: Plan View of Lower NPM Bay with Skirt Restraint

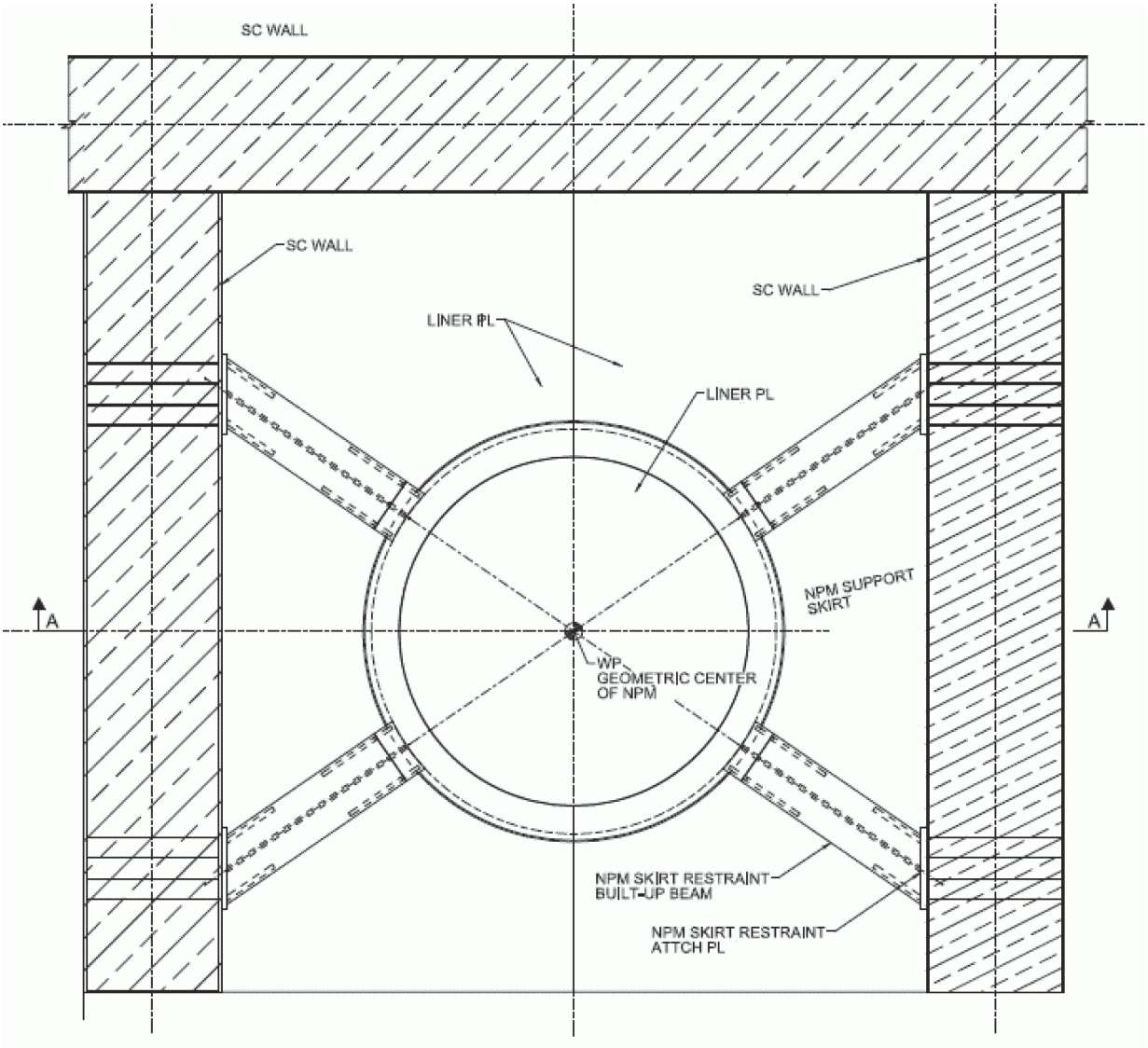
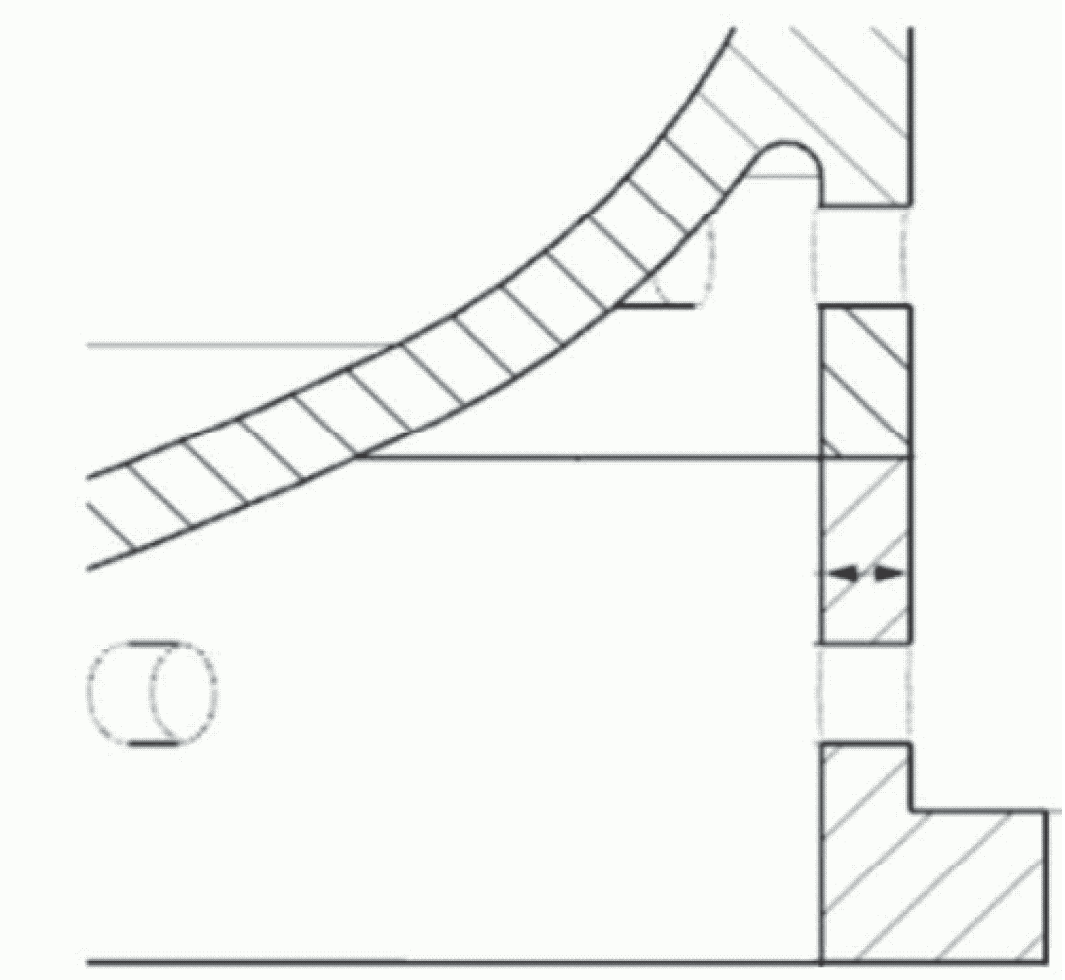


Figure 3B-34: NuScale Power Module Section View at Skirt Support



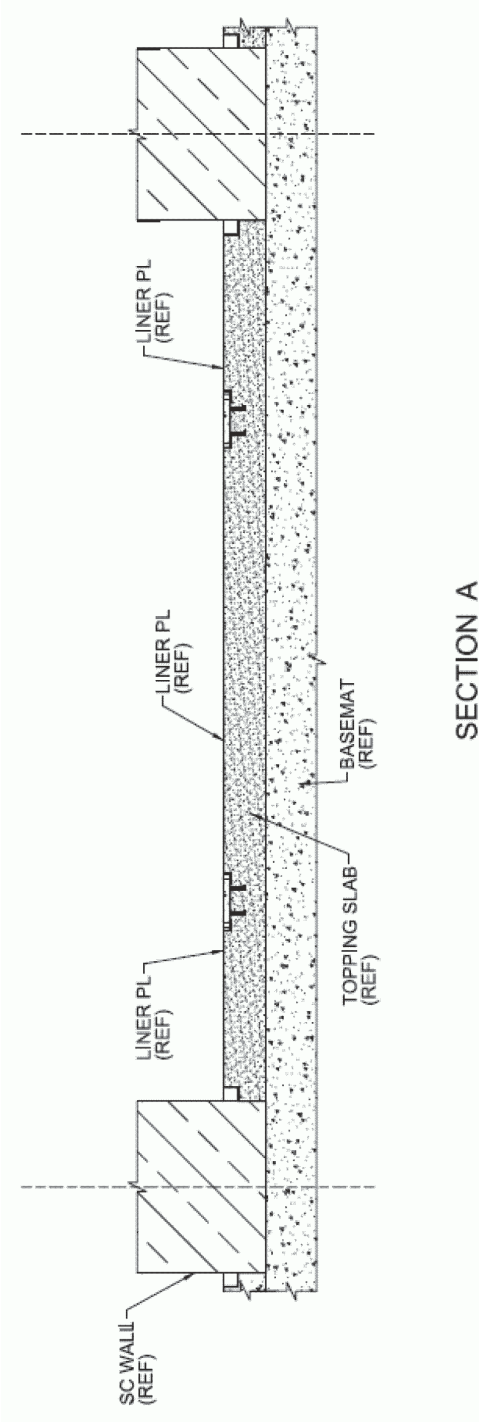


Figure 3B-35: Elevation View of Lower NPM Bay

Figure 3B-36: Plan View Layout of NPM Lug Restraint Configurations

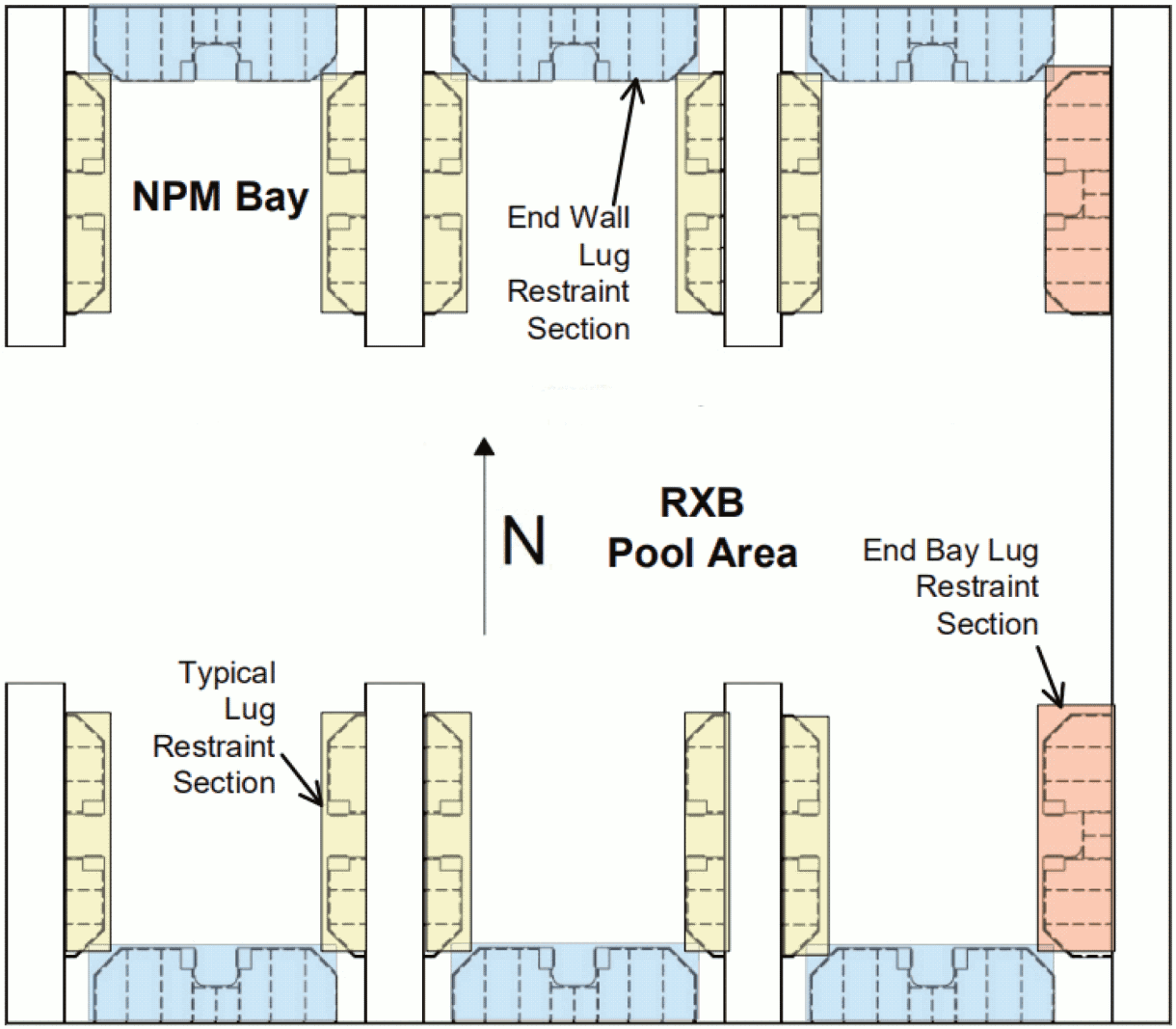


Figure 3B-37: Plan View of Typical NuScale Power Module Bay with Lug Restraints

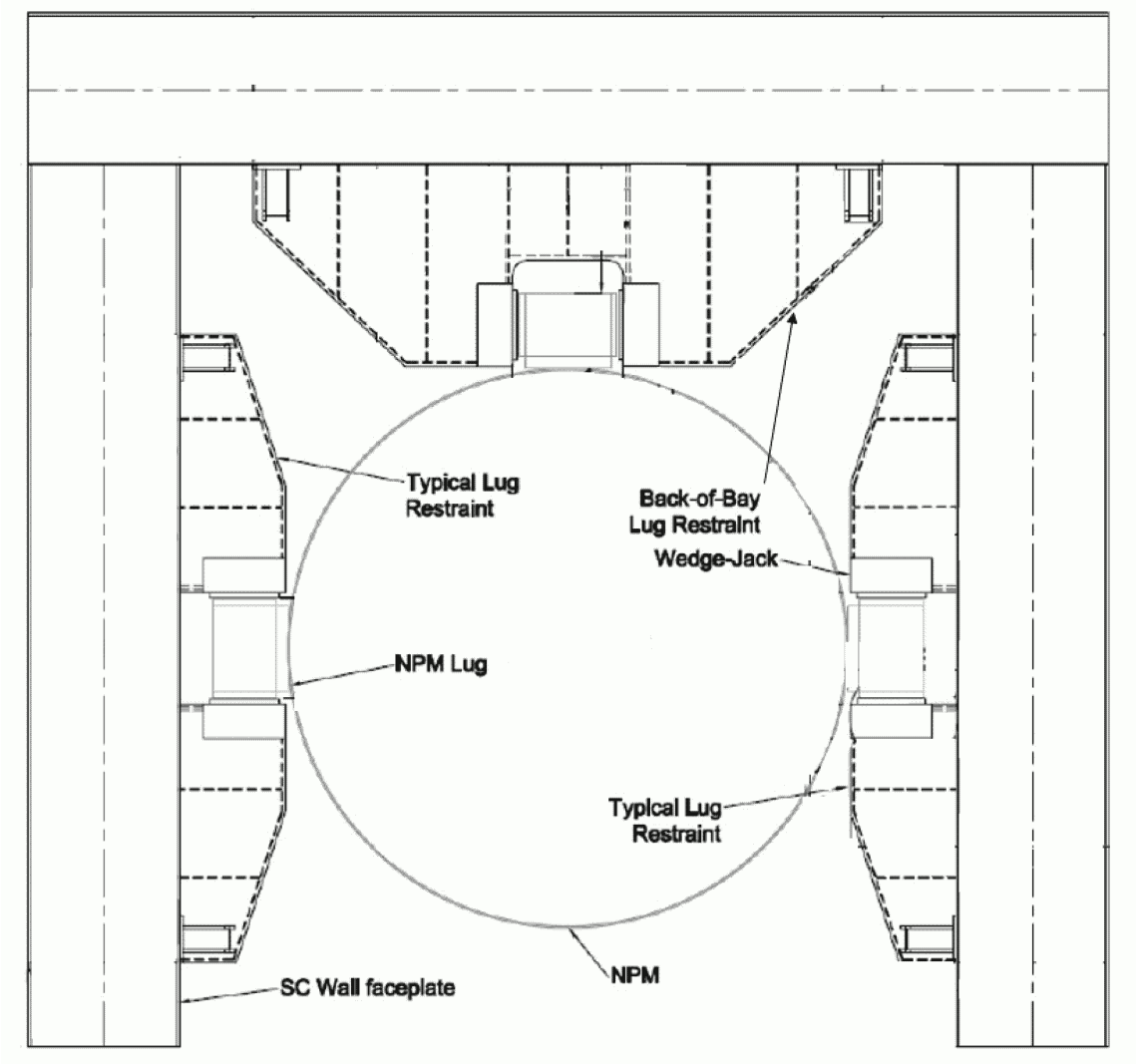
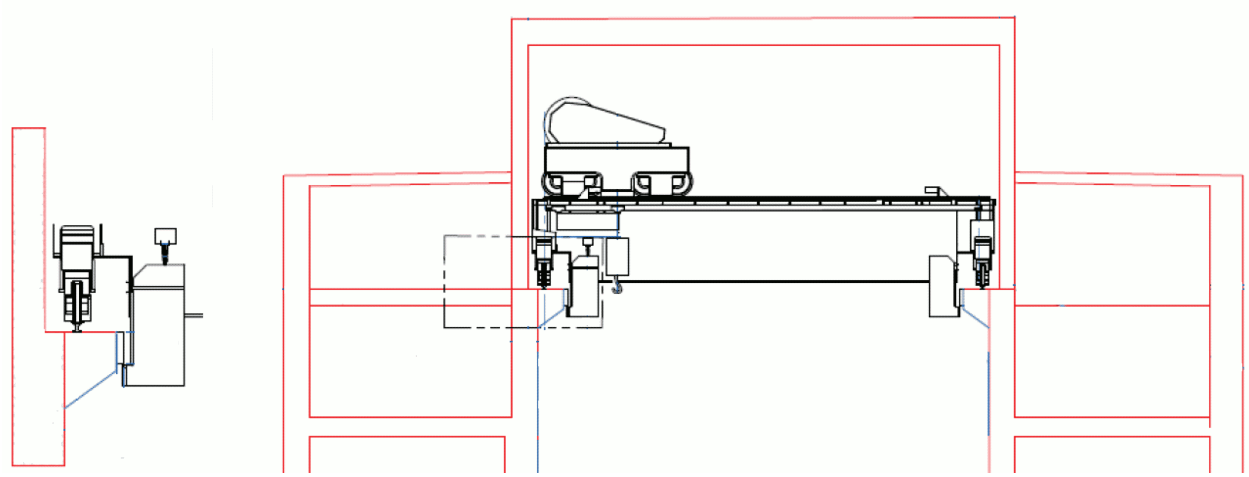
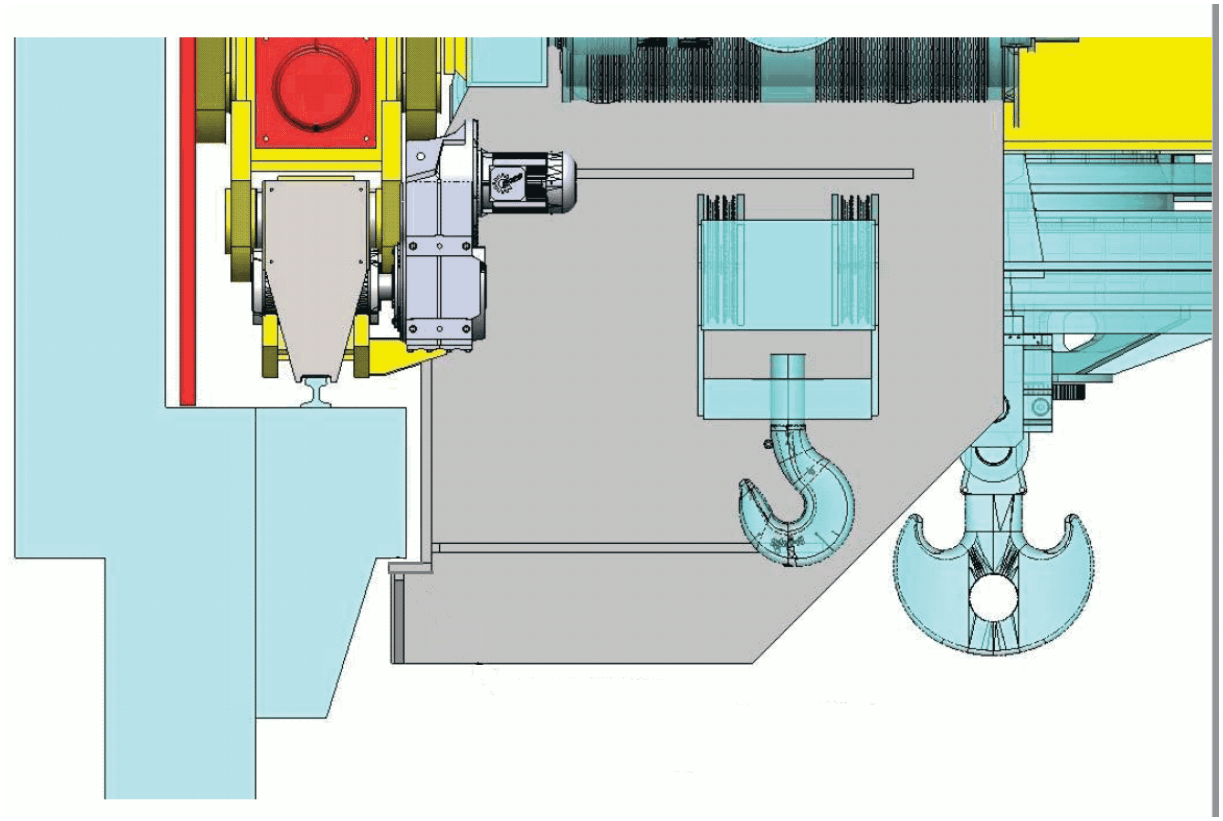


Figure 3B-38: Reactor Building Crane Corbel Location



The figure is to represent a concept of a stiffened beam that is to be attached to the SC structure. Components shown are not to scale.

Figure 3B-39: Reactor Building Crane Side View



The figure is to represent a concept of a stiffened beam that is to be attached to the SC structure. Components shown are not to scale.

Figure 3B-40: Coupler Dimensional Specifications

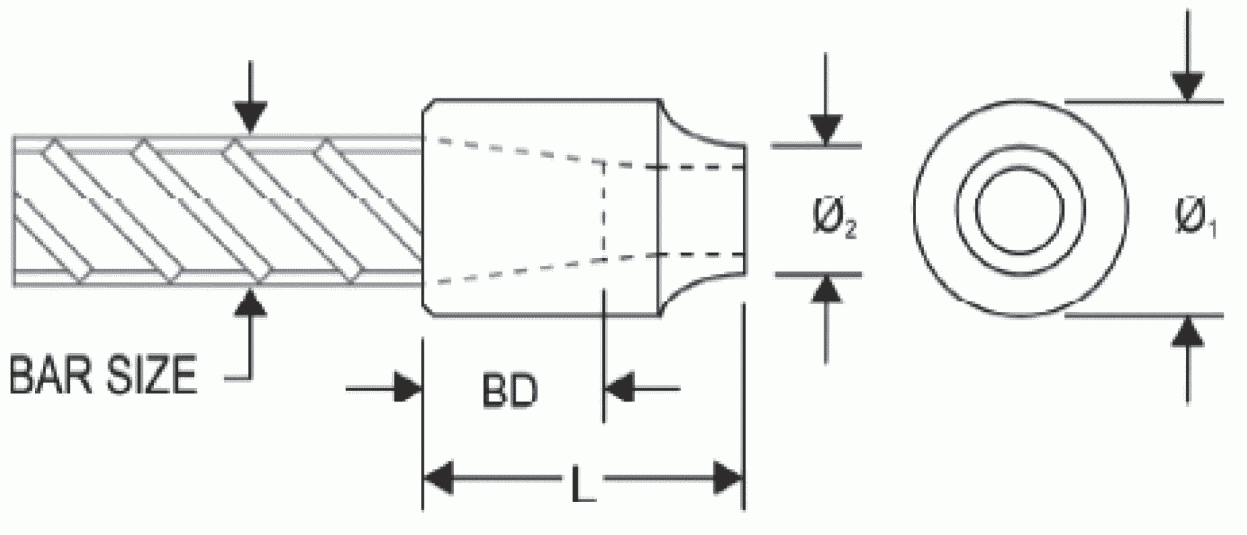


Figure 3B-41: Development Plate Dimensional Properties

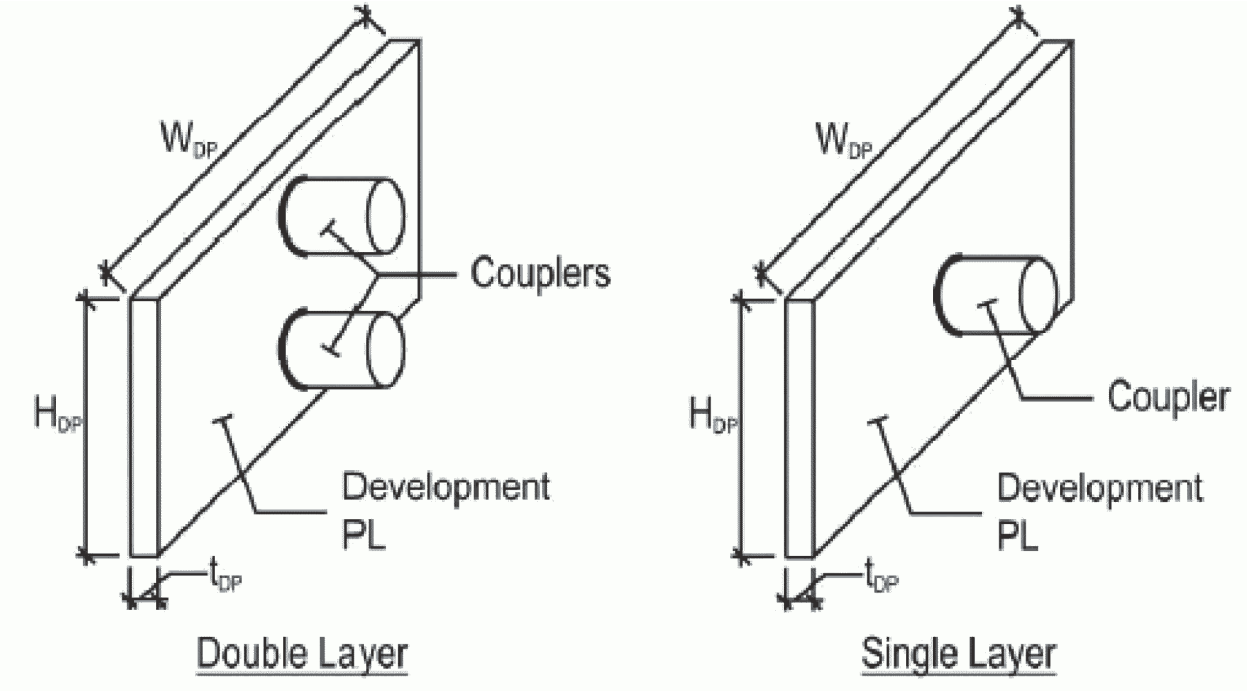


Figure 3B-41a: Control Building Wall 3 Contour Plot (Force unit kip/ft and Moment unit kip-in/ft)

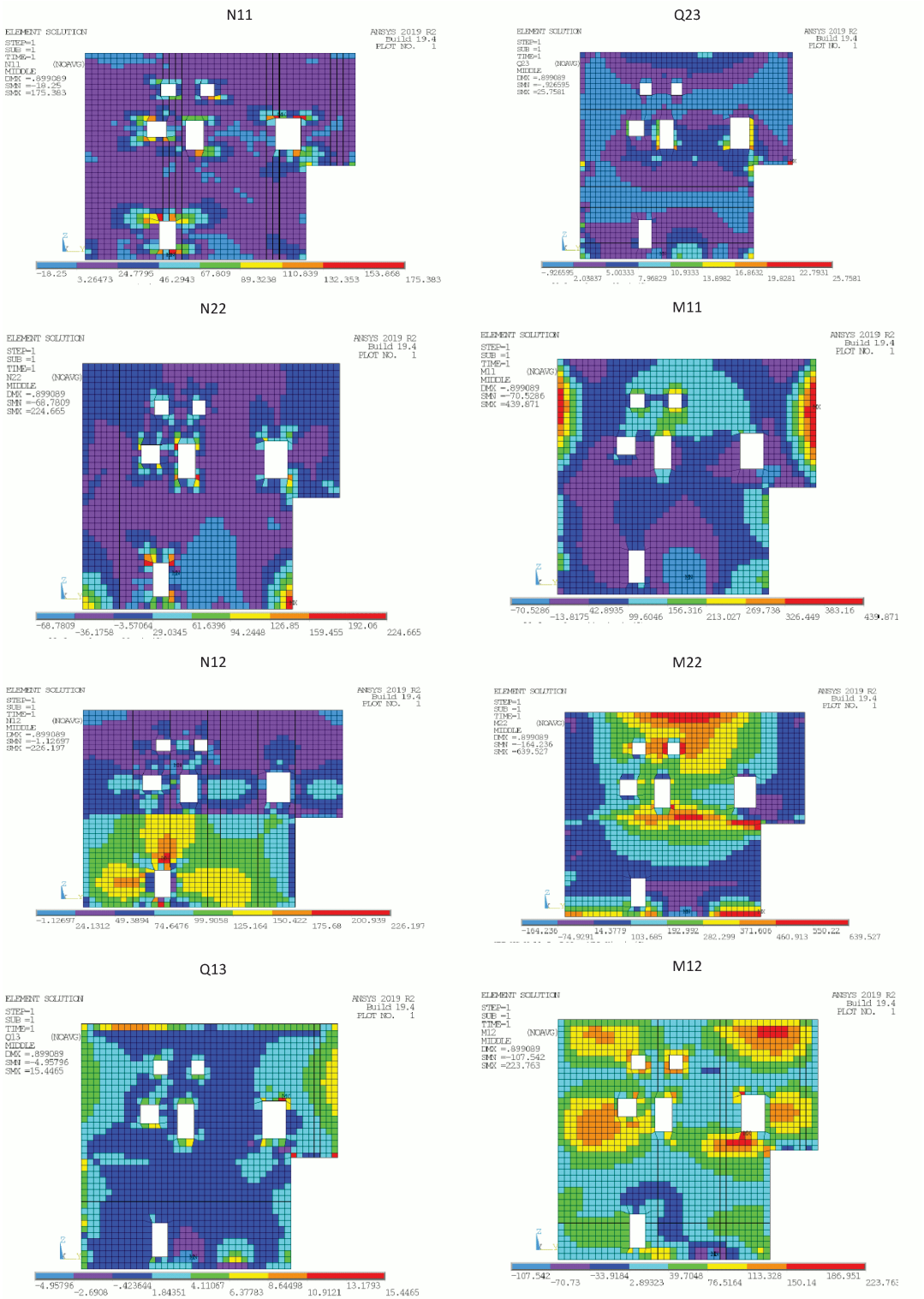


Figure 3B-41b: Control Building Wall 5 Contour Plot (Force unit kip/ft and Moment unit kip-in/ft)

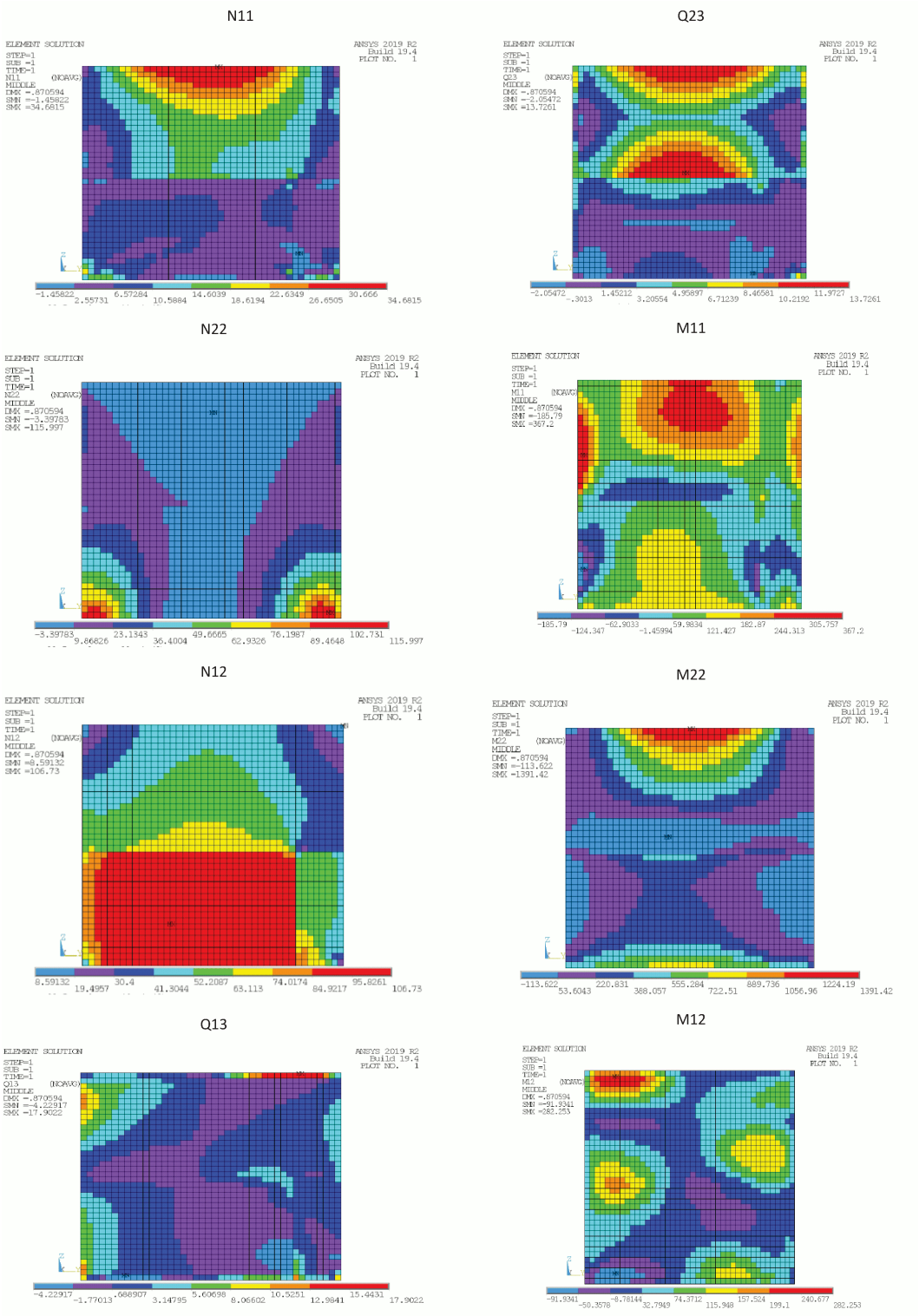


Figure 3B-41c: Control Building Wall H Contour Plot (Force unit kip/ft and Moment unit kip-in/ft)

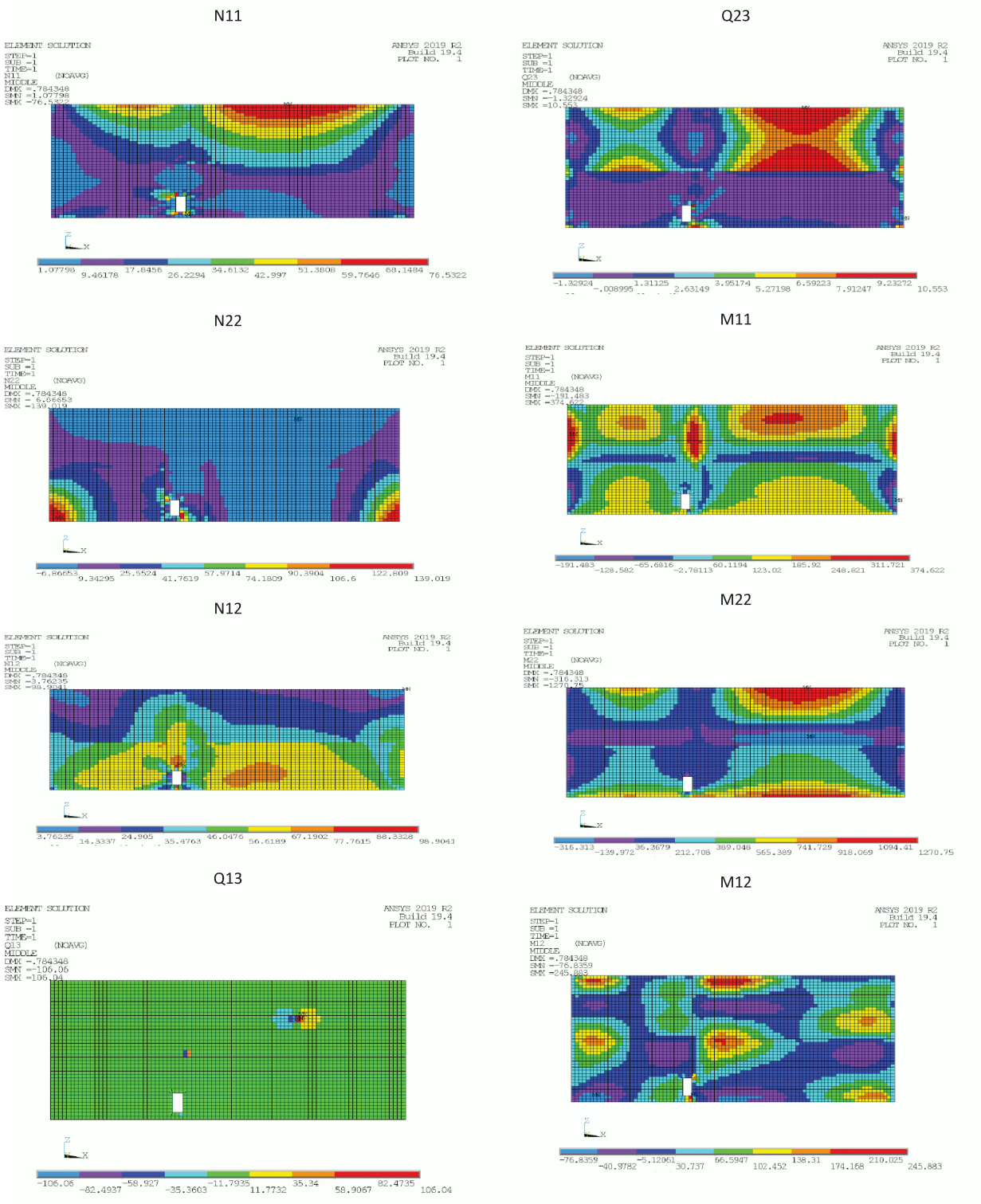


Figure 3B-41d: Control Building Slab at 123’ Contour Plot (Force unit kip/ft and Moment unit kip-in/ft)

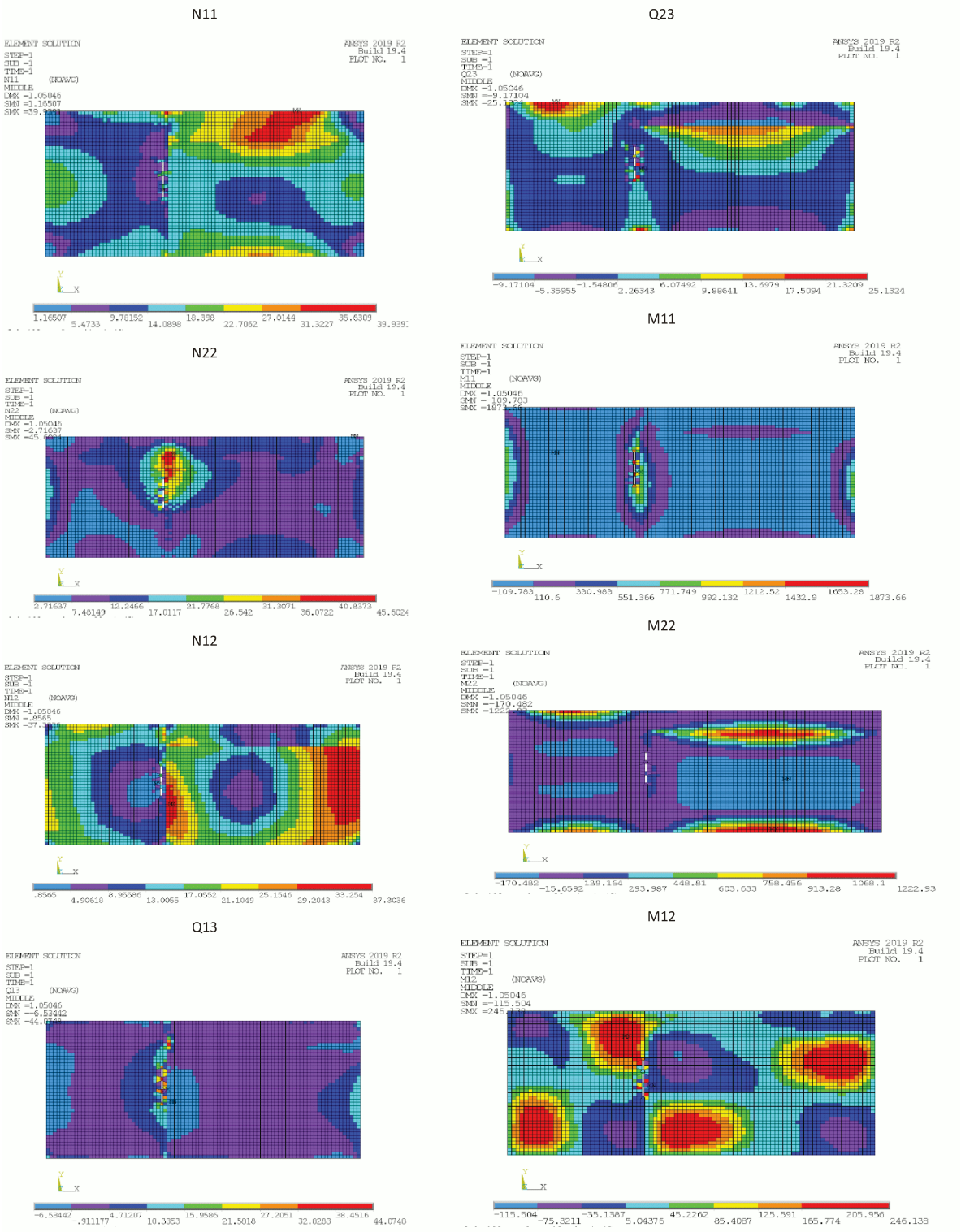


Figure 3B-41e: Control Building Slab at Basemat Contour Plot (Force unit kip/ft and Moment unit kip-in/ft)

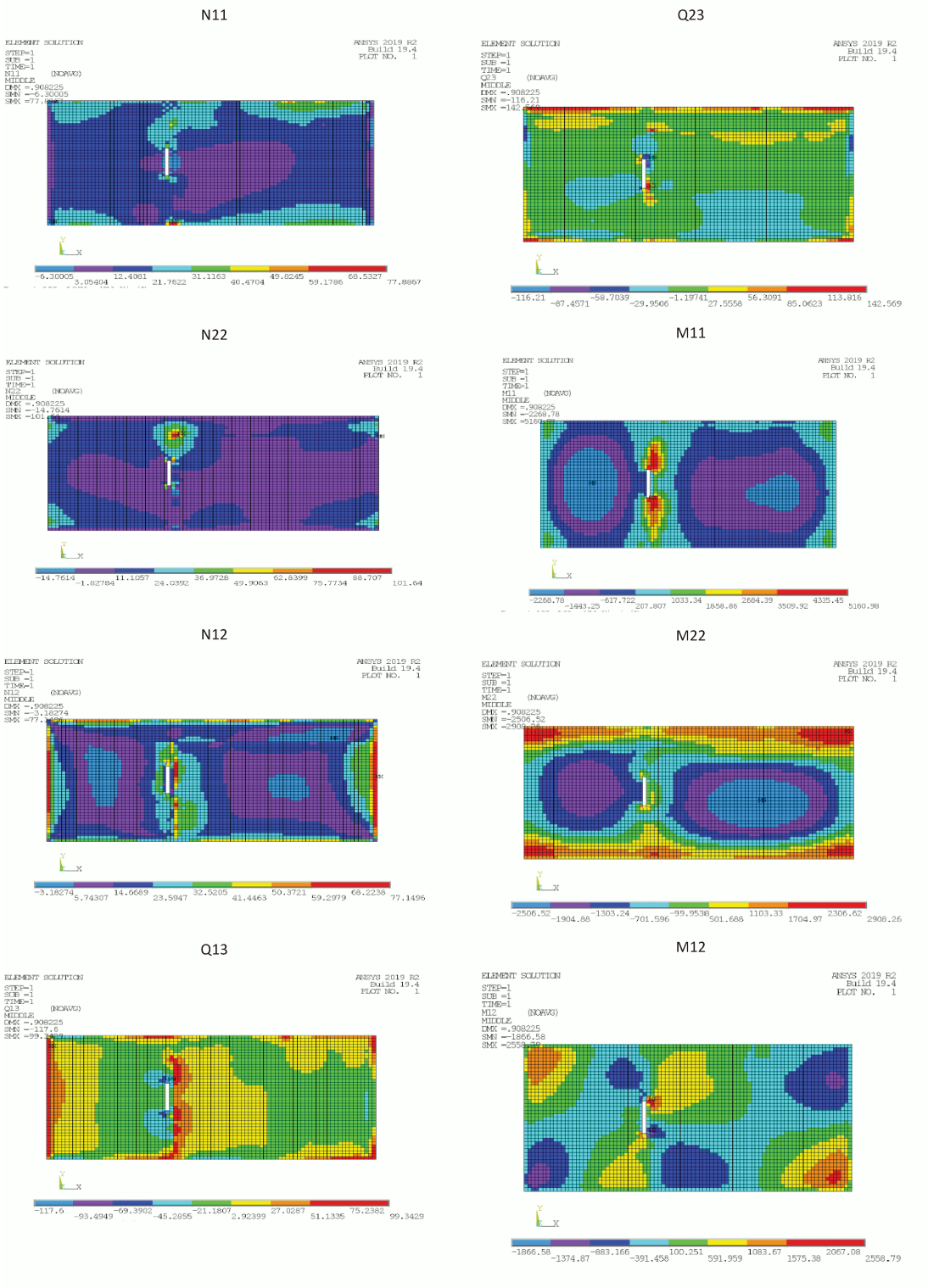


Figure 3B-41f: Section Cuts for Evaluating Control Building Wall at Gridline 3 between EL. 100 feet and EL. 123 ft (Region CRB_3_100)

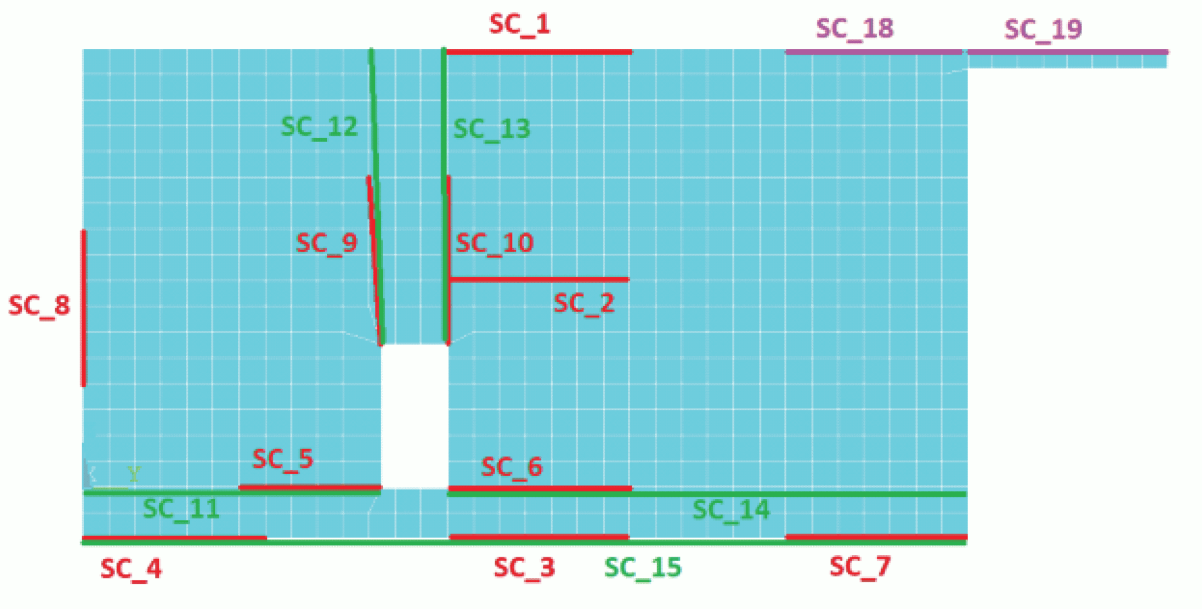


Figure 3B-42: Section Cuts for Evaluating Control Building Wall at Gridline 3 between EL 123 feet and Roof (Region CRB_3_123)

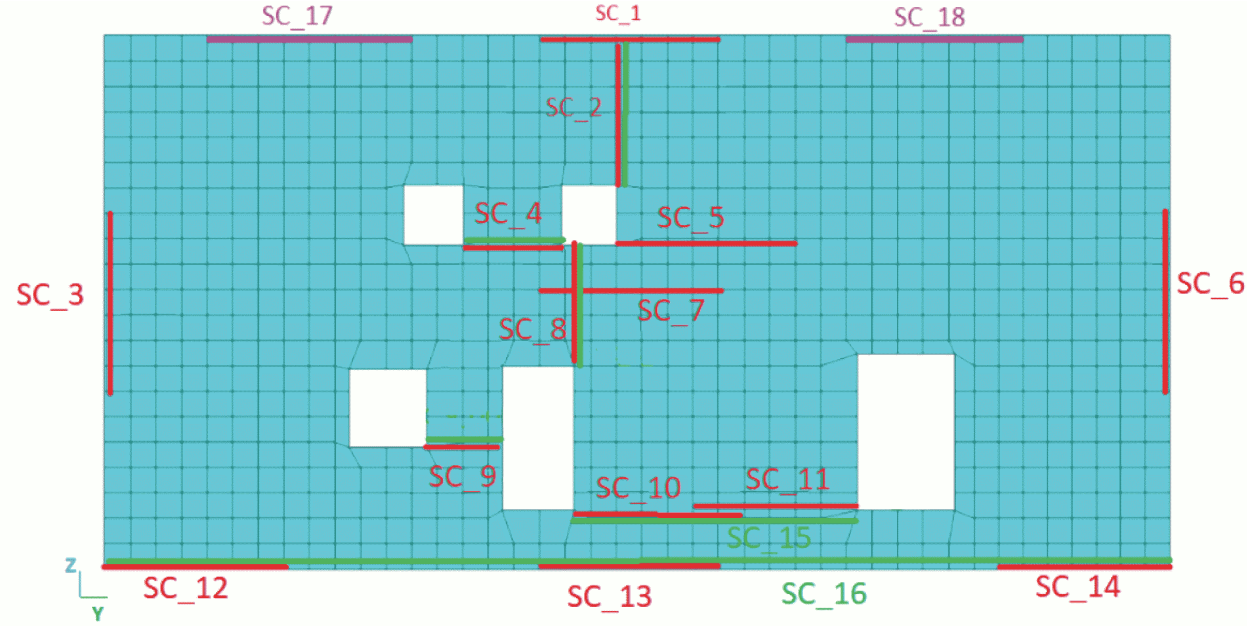


Figure 3B-43: Section Cuts for Evaluating Control Building Wall at Gridline 5 between EL. 100 feet and EL. 123 feet (Region CRB_5_100)

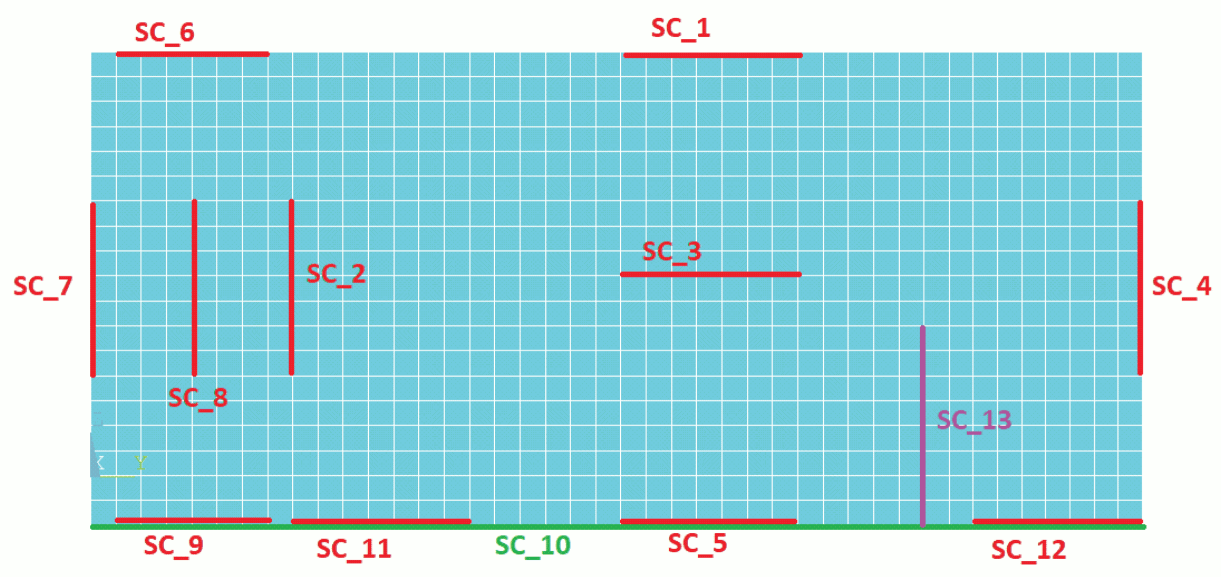


Figure 3B-44: Section Cuts for Evaluating CRB Wall at Gridline 5 between EL. 123'and roof (Region CRB_5_123)

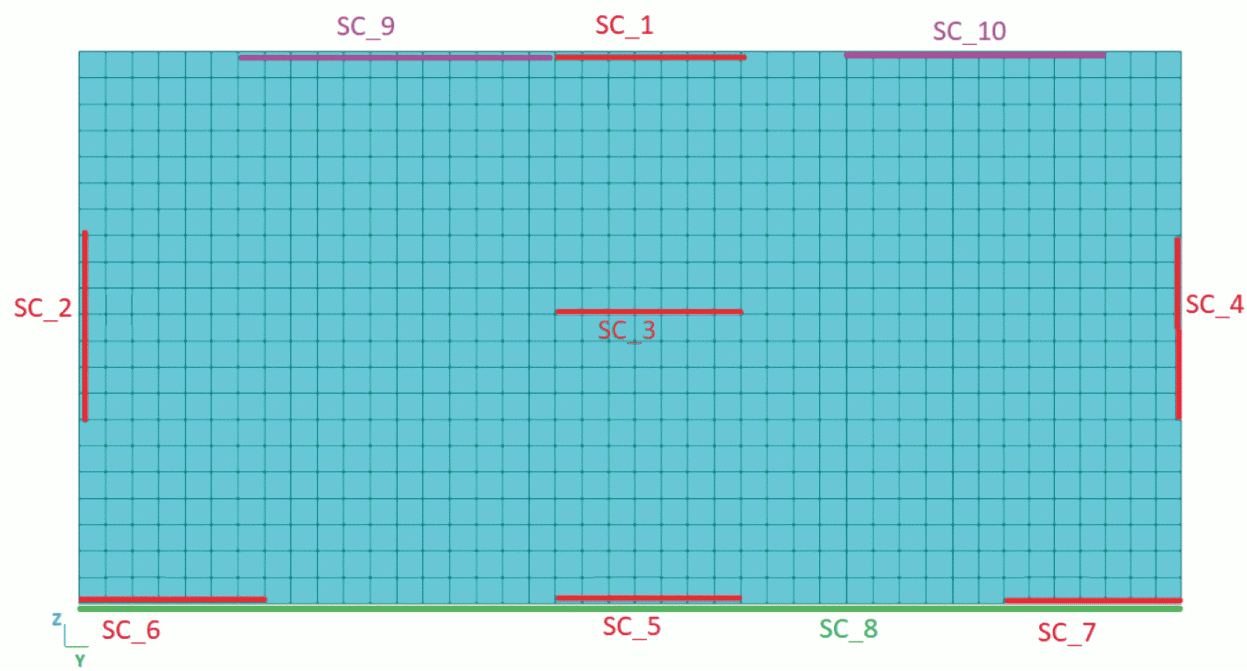


Figure 3B-45: Section Cuts for Evaluating Control Building Wall at Gridline H between EL. 100 feet and EL. 123 feet (Region CRB_H_100)

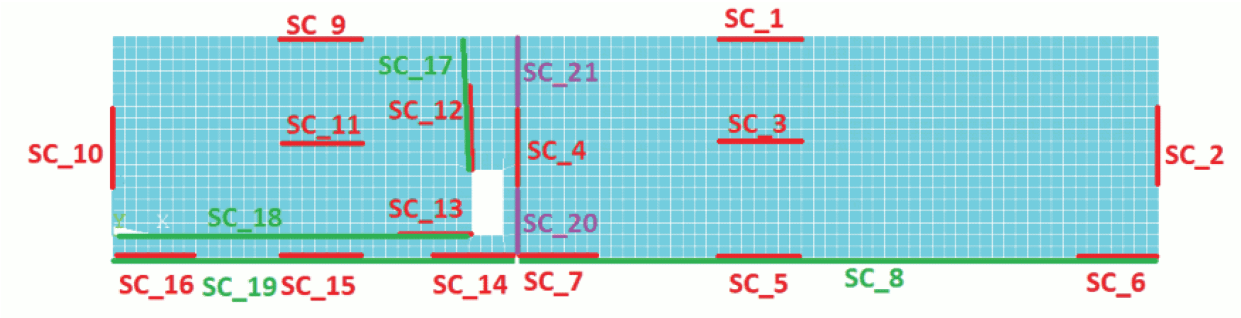


Figure 3B-46: Section Cuts for Evaluating Control Building Wall at Gridline H between EL. 123 feet and Roof (Region CRB_H_123)

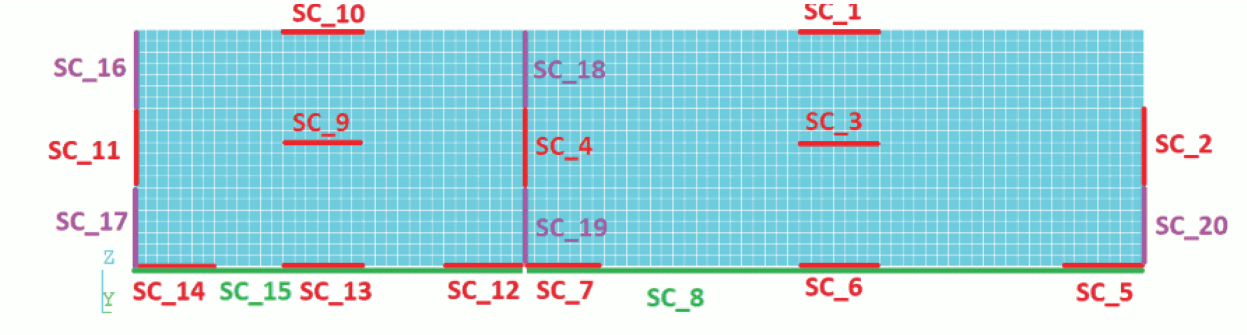


Figure 3B-47: Section Cuts for Evaluating the Control Building Basemat at EL. 100 feet (Region CRB_Basemat_100)

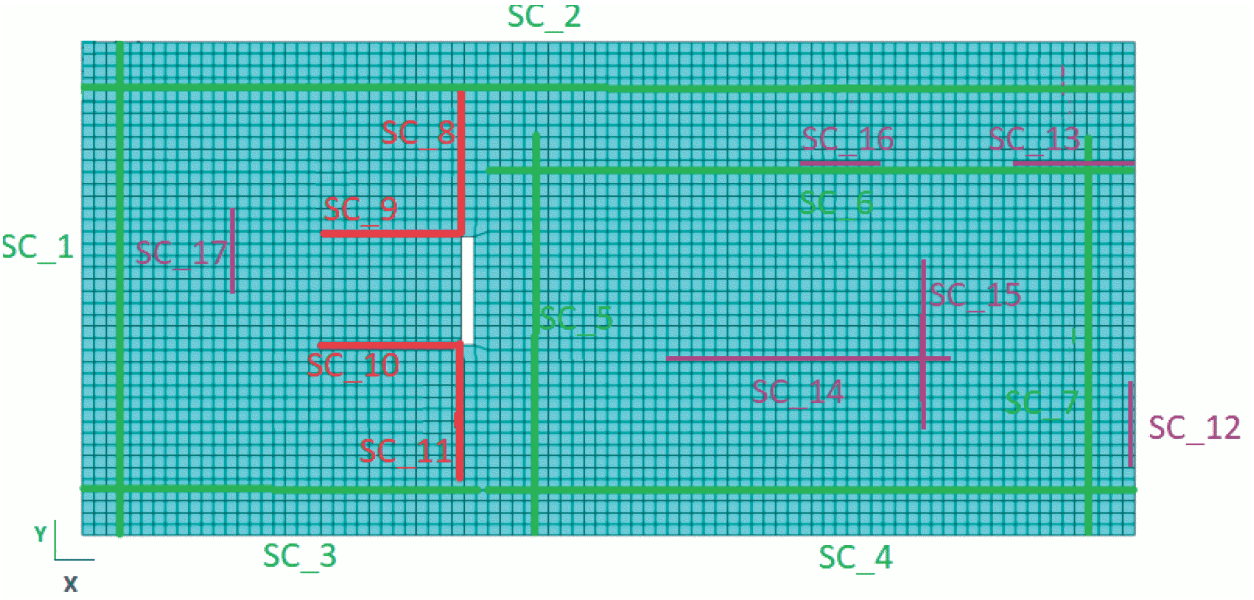


Figure 3B-48: Section Cuts for Evaluating Control Building Slab between Gridlines 1-3-F-H at EL. 123 feet (Region CRB_Slab_123_1)

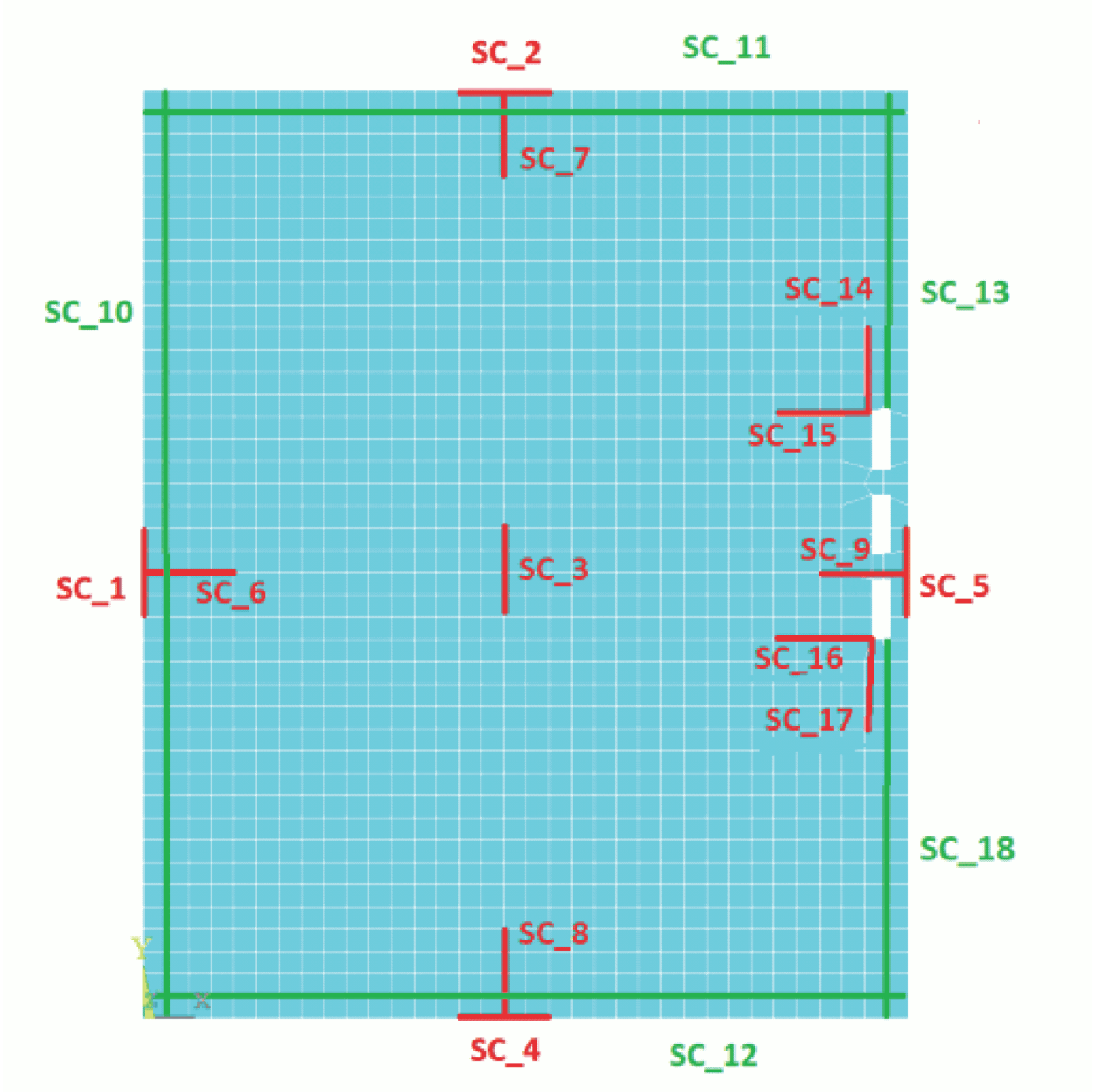


Figure 3B-49: Section Cuts for Evaluating Control Building Slab between Gridlines 3-5-G-H at EL. 123 feet (Region CRB_Slab_123_2)

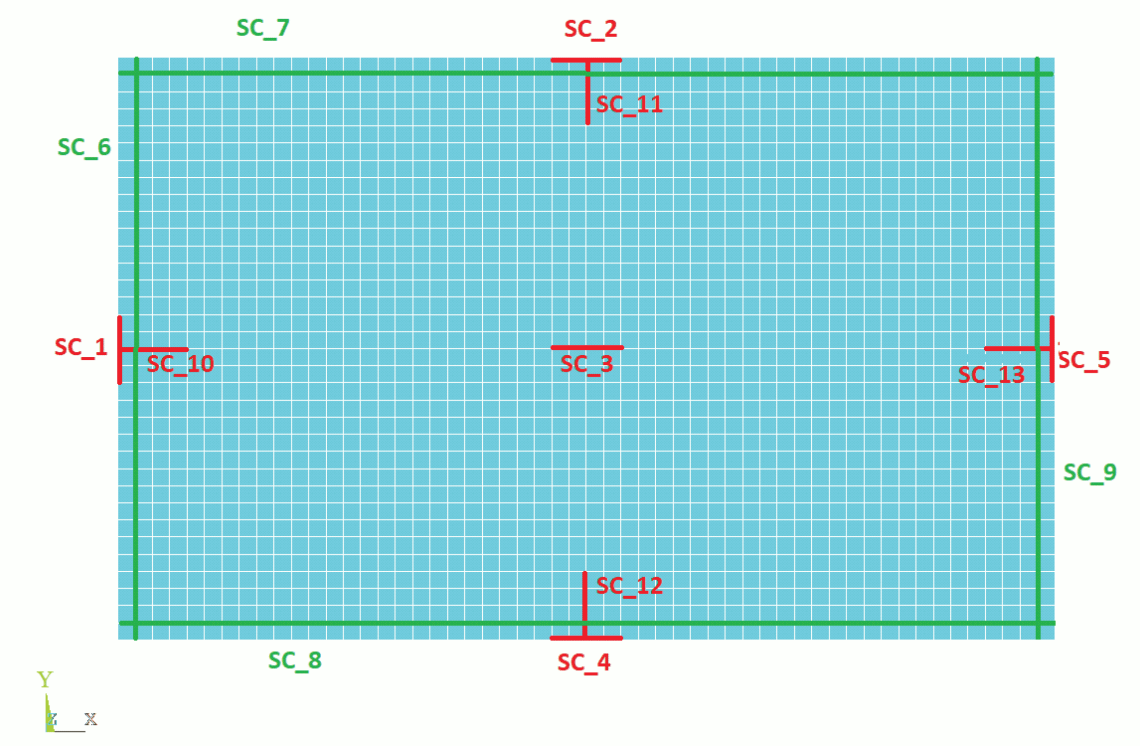
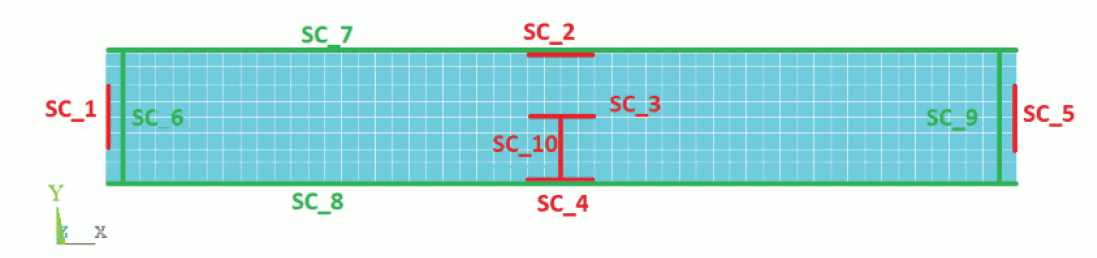


Figure 3B-50: Section Cuts for Evaluating Control Building Slab between Gridlines 3-5-F-G at EL 123 feet (Region CRB_Slab_123_3)



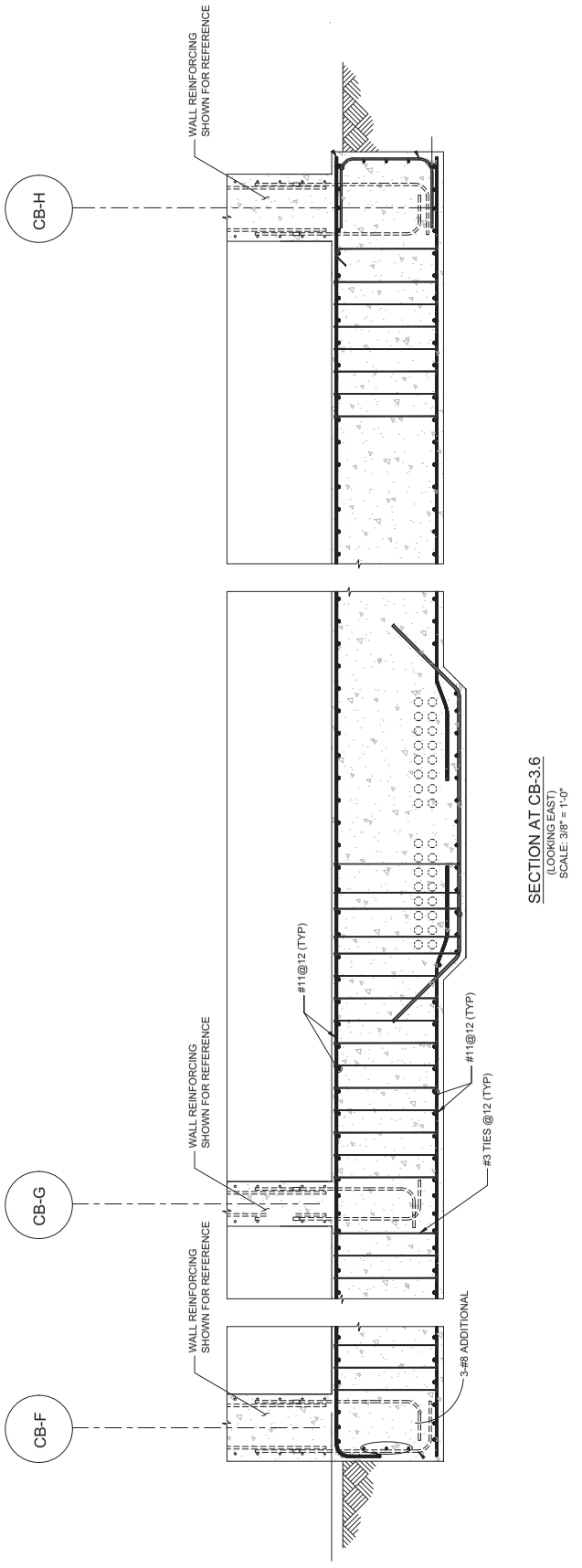


Figure 3B-51: Control Building Critical Sections Basemat Reinforcing

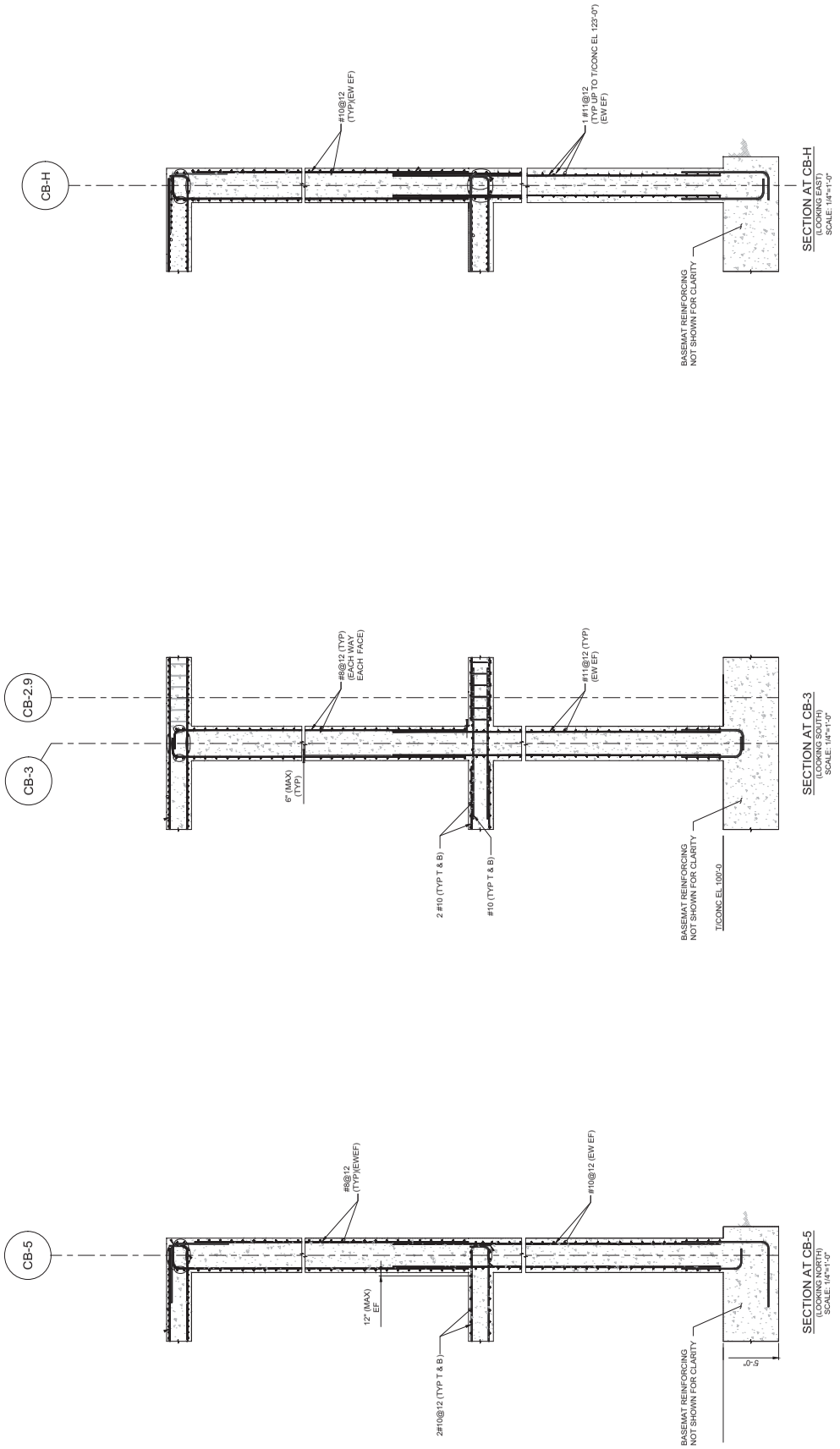


Figure 3B-52: Control building Critical Sections

Appendix 3C Methodology for Environmental Qualification of Electrical and Mechanical Equipment

3C.1 Purpose

This appendix describes the Environmental Qualification Program methodology for qualifying electrical equipment and mechanical equipment in accordance with the applicable requirements. Section 3.11 and Section 3.10 address the environmental qualification and seismic and dynamic qualification of electrical and mechanical equipment, respectively.

This appendix defines the qualification methods employed to ensure the functionality of mechanical and electrical equipment (including instrumentation and controls) required to perform a credited function during the full range of normal and accident loadings (including seismic), and under all normal environmental conditions, anticipated operational occurrences, and accident and post-accident environmental conditions.

3C.2 Scope

This appendix presents the methods and procedures for qualifying electrical and mechanical equipment to a range of environments that the equipment could be exposed during normal and abnormal conditions or design basis events (DBEs).

These methods and procedures apply to mechanical and electrical equipment associated with systems essential to reactivity control, decay heat removal, post-accident monitoring, containment isolation, maintenance of reactor coolant system (RCS) pressure boundary integrity, control room habitability, event severity mitigation, or system support functions.

This appendix specifies the plant environmental conditions to which equipment listed in Section 3.11 is designed and qualified. The environmental conditions are defined for plant conditions, including normal and abnormal operating conditions, and accident conditions including post-accident operations. The accident conditions considered are postulated events not reasonably expected to occur over the course of plant life and that could potentially result in creating adverse environmental conditions for qualified equipment to perform its credited function.

This appendix addresses the primary environmental parameters of pressure, temperature, relative humidity, radiation, chemical conditions, spray/wetting, and submergence. In accordance with 10 CFR 50.49, equipment subject to environmental qualification requirements is designed and qualified to the environmental conditions resulting from the most limiting design basis event (DBE). Natural phenomenon or external events are excluded from consideration in environmental qualification. The design and qualification parameters for the equipment meet the Environmental Qualification Program acceptance criteria.

3C.3 Regulatory and Industry Codes

General Design Criterion (GDC) 1, GDC 2, GDC 4, and GDC 23 of 10 CFR 50, Appendix A; Quality Assurance Criteria III, XI, and XVII of 10 CFR 50, Appendix B; and 10 CFR 50.49 establish the regulatory requirements for this program.

Active, passive, or fail safe electrical and mechanical equipment, including instrumentation, must be qualified to operate in environments associated with DBEs that they are required to operate. The primary objective of environmental qualification is to demonstrate with reasonable assurance that equipment for which there is an established qualified life or condition can perform its credited function without experiencing common-cause failures (CCFs) during and after applicable DBEs. Environmental design requirements apply to equipment located in mild or harsh environments. This appendix defines the environmental parameters for which equipment must be qualified. Electrical equipment located in a harsh environment is qualified in accordance with the requirements of 10 CFR 50.49. Active mechanical equipment located in a harsh environment is qualified to comply with the requirements of GDC 4 by incorporating the design basis environmental conditions into the design process. Mechanical equipment that performs a credited active function during or following exposure to harsh environmental conditions is qualified in accordance with American Society of Mechanical Engineers (ASME) QME-1, Appendix QR-B (Reference 3C-3) with the following exceptions.

- QR-B5200, Identification and Specification of Qualification Requirements, (g) material activation energy
- QR-B5300 Selection of Qualification Methods for determination and recording of shelf life of nonmetallics
- QR-B5500 Documentation, (h) shelf life preservation requirements

These exceptions are addressed with the following alternatives.

- *QR-B5200, Identification and Specification of Qualification Requirements, (g) material's activation energy (in conjunction with one of the above identification methods only and that is based on the material's critical failure mechanism in the intended service).*

  Alternative:

  In accordance with Appendix QR-B5200, nonmetallic material is qualified to perform its intended functions. Although activation energy might not be used for material identification purposes per QR-B5200, the activation energy is applied to the thermal energy equation for determining material degradation and qualification.

- *QR-B5300, Selection of Qualification Methods, last paragraph states, “The shelf life of all nonmetallics, and any applicable storage limitations, should be determined and recorded in the qualification documentation.”*

Alternative:

Shelf life and preservation requirements are documented in accordance with the NQA-1 2008, Requirement 13 and Subpart 2.2, in lieu of ASME QME-1 2017, Appendix QR-B5300. These requirements are not included in the equipment qualification record file (EQRF), but are documented separately.

- *QR-B5500, Documentation, (h) shelf life preservation requirements.*

  Alternative:

  Shelf life preservation requirements are documented in accordance with the NQA-1 2008, Requirement 13 and Subpart 2.2 in lieu of ASME QME-1 2017, Appendix QR-B5500, item (h). These requirements are not included in the EQRF, but are documented separately.

Mechanical and electrical equipment located in mild environments, and required to perform a credited function, is qualified in accordance with the provisions of GDC 4. For each piece of equipment selected for environmental qualification, the environmental parameters and the qualification process is listed in the associated EQRF.

3C.4 Qualification Process

3C.4.1 Specifications

The equipment specification identifies the applicable codes and standards, required operating times, performance requirements, credited functions, service conditions, accepted methods of qualification, and acceptance criteria. The equipment specification also provides the basis for establishing the environmental qualification of the specific equipment types.

3C.4.1.1 Design Life

Design life is the time period during which satisfactory performance can be expected for a specific set of service conditions.

3C.4.1.2 Qualified Life

Qualified life is the period of time, prior to the start of a DBE for which the equipment was demonstrated to meet the design requirements for the specified service conditions.

3C.4.1.3 Qualified Life Objective

The qualified life objective is based upon a specified set of service conditions. Age conditioning a test sample to simulate effects of significant aging mechanisms during a time equal to the qualified life objective can demonstrate qualified life. An adjunct to establishing a qualified life objective is establishing an end-condition objective of equipment condition indicators that correlate to

the ability of equipment to perform its credited function. In this case, the end condition is the basis of qualification, and the time to reach that end condition in service may be more or less than the qualified life established by age conditioning.

3C.4.1.4 Performance Criteria

The Qualification Test Program demonstrates the capability of the equipment to meet the credited function performance requirements defined in the equipment specification. The primary objective of qualification is to provide reasonable assurance that equipment, with a qualified life or condition established, can perform its credited functions without experiencing CCFs during and after applicable DBEs.

3C.4.2 Environmental Conditions

The environmental conditions considered in the qualification process are pressure, temperature, humidity, radiation, chemical effects, and submergence. The Environmental Qualification Program addresses the appropriate margins to be included during qualification. The plant environmental conditions are characterized as either harsh or mild.

3C.4.2.1 Harsh Environment

A harsh environment is an environment resulting from a DBE that would be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences. Equipment qualified to operate in a harsh environment must operate without a loss of capability to perform their credited function throughout their qualified life. Section 3.11 identifies harsh environment thresholds and the equipment requiring qualification for a harsh environment.

Instruments and devices requiring qualification include the associated sensors, and supporting loop components. The supporting components of a sensor, such as cables, connectors, terminals, junction boxes, preamplifiers, or other signal processing equipment, is qualified for the environmental conditions at the component's location. Electrical equipment in a harsh environment is qualified according to the requirements of Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974 (Reference 3C-2).

The design of mechanical equipment located in harsh environmental zones ensures performance under appropriate environmental conditions. The primary focus for mechanical equipment concerns materials sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms).

Table 3C-2 lists the harsh environmental zones within the Reactor Building (RXB).

3C.4.2.2 Mild Environment

A mild environment is never more severe than the normal plant environment, including during anticipated operational occurrences. To qualify equipment operating in a mild environment, the equipment specification provided to the vendor or supplier quantitatively describes the environmental conditions. Certification from the vendor or supplier that the equipment operates in the environment described in the specification is sufficient to qualify the equipment. Seismic and aging qualification may require additional analysis or testing.

Institute of Electrical and Electronics Engineers Standard 323-2003 (Reference 3C-1) addresses qualification of computer-based instrumentation and control systems to mild environments.

Table 3C-3 lists the mild environmental zones within the plant.

3C.4.2.3 Normal Operating Conditions

Table 3C-6 summarizes normal operating conditions.

3C.4.2.3.1 Normal Radiation Dose

The bases for the normal radiation integrated doses for equipment inside and outside containment are the maximum normal RCS radionuclide activities and system parameters as shown in Table 3C-6. These values are determined based on continuous operation and steady-state operating conditions for the life of the plant.

3C.4.2.3.2 Seismic

Section 3.7 and Section 3.10 address the methods, including applicable seismic loads, used for the seismic qualification of mechanical, electrical, and instrumentation and control equipment.

3C.4.2.3.3 Containment Test Environment

The containment hydrostatic test and subsequent containment leakage tests, as described in Section 6.2.6, are considered for equipment undergoing exposure to these pressure cycles. The qualification of equipment includes the containment pressure test environment when these test conditions are considered to be a significant aging mechanism.

3C.4.2.4 Design Basis Event Conditions

3C.4.2.4.1 Design Basis Events

Design basis events include infrequent events, anticipated operational occurrences, and design basis accidents as analyzed within the scope of Section 3.6 and Chapter 15.

3C.4.2.4.2 Design-Basis Accidents

The design basis accidents are reviewed and evaluated to determine which DBAs addressed in Chapter 15 have the potential to result in environmental conditions significantly more severe than normal operating conditions. Based on this review, evaluation of the following DBAs determines the mechanical and electrical equipment that requires environmental qualification.

Section 15.1.5 - steam system piping failure inside and outside of containment. That section covers main steam line breaks inside and outside of containment. For the purpose of environmental qualification, non-mechanistic pipe rupture is postulated inside the containment vessel (CNV) even though the main steam piping meets the break exclusion criteria of item 2.A(ii) of Branch Technical Position 3-4.

Section 15.2.8 - feedwater system pipe break inside and outside of containment. That section covers feedwater line breaks inside and outside of containment. For the purpose of environmental qualification, non-mechanistic pipe rupture is postulated inside the CNV even though the feedwater piping meets the break exclusion criteria of item 2.A(ii) of Branch Technical Position 3-4.

Section 15.4.8 - rod ejection accident. That section reflects a potential break in the RCS pressure boundary. The equipment relied upon to mitigate this accident is the same as that used for the spectrum of small break loss-of-coolant accidents (LOCAs) addressed by Section 15.6.5.

Section 15.6.5 - LOCAs from spectrum of postulated pipe breaks within the RCS pressure boundary inside and outside of containment. There are no large break LOCA events for the NuScale Power Plant US460 standard design. The small break LOCAs are the result of chemical and volume control system pipe rupture events postulated inside or outside of containment.

Note: The core damage event described in Section 15.10 is a special event outside of the scope of the Environmental Qualification Program.

Section 15.7.4 - radiological consequences of fuel handling accidents. That section covers the fuel handling accidents within the RXB pool area when an assembly is dropped in the reactor pool above the spent fuel racks or weir wall, dropped in the reactor core during refueling, or when an assembly impacts a spent fuel cask during loading.

Section 15.7.5 - spent fuel cask and NuScale Power Module drop accidents. The Reactor Building crane moves the spent fuel cask and NuScale Power Modules in the RXB refueling area. The Reactor Building crane design conforms to the single-failure-proof guidelines of NUREG-0612 so that a credible failure of a single component does not result in the loss of capability to stop and hold a critical load.

3C.4.2.4.3 Infrequent Events

Infrequent events (IEs) are reviewed and evaluated to determine which IEs addressed in Chapter 15 have the potential to result in environmental conditions significantly more severe than normal operating conditions. Based on this review, the following IE is evaluated to determine the mechanical and electrical equipment that requires environmental qualification in order to preclude environmentally induced CCF.

Section 15.6.2 - radiological consequences of failure of small lines carrying primary coolant outside of containment. Similar to Section 15.6.5, the section covers postulated chemical and volume control system pipe rupture events inside or outside of containment.

3C.4.2.4.4 Other Design Basis Events

Other design basis events are reviewed and evaluated to determine which of them have the potential to result in environmental conditions significantly more severe than normal operating conditions. Based on this review, the following events are evaluated to determine the mechanical and electrical equipment that requires environmental qualification in order to preclude environmentally induced CCF.

Section 3.6 - high-energy line breaks (HELB) outside containment. That section covers high-energy line breaks in the RXB not already addressed by Section 15.1.5, Section 15.2.8, or Section 15.6.5, such as the postulated rupture of the module heatup system piping in the heat exchanger room.

Section 3.6 - moderate energy line breaks (MELB) outside containment.

3C.4.2.4.5 Normal and Bounding Conditions

Table 3C-7 shows CNV and RXB temperature, pressure, and humidity experienced during the indicated DBE. Equipment that is required to perform a credited function and could potentially be subjected to the design basis environments, is qualified to these conditions for the required operating time.

3C.4.2.5 Design Basis Event Radiation Conditions

NuScale Topical Report TR-0915-17565-P-A (Reference 3C-4) provides the methodology for determining the source terms for equipment following design basis events. The limiting event and associated source terms from the design basis accidents discussed above are used to determine total integrated doses for environmental qualification.

The accident conditions integrated doses within the RXB are determined using the maximum normal core radionuclide inventory. The required dose used for environmental qualification considers the total integrated dose, consisting of

the normal dose plus the accident dose corresponding to the required post-accident operating time. The normal dose considers gamma and neutron effects, while the accident dose considers the gamma and beta dose expected at the equipment location.

The iodine spike design basis source term described in Section 15.0.3 is used in the Environmental Qualification Program as a bounding surrogate for the radiological consequences of DBEs that result in primary coolant entering the containment.

Based on the above, Table 3C-8 shows the integrated doses following a design basis event.

Section 3.11.4 discusses gamma and beta radiation effects.

3C.4.3 Margin

Margin used in the qualification program accounts for reasonable uncertainties in demonstrating satisfactory performance and normal variations in commercial production, thereby providing reasonable assurance that the equipment can perform under the most adverse service condition specified. Margins are also applied to tested conditions to account for the uncertainties associated with test measurement equipment. Margins, therefore, represent the conservatisms that exist when comparing the actual performance and environmental requirements established for plant equipment with those similar requirements demonstrated during test simulations. Margin application is in addition to conservatisms applied during the derivation of the DBE environmental conditions. Acceptable margin values (Table 3C-5) are developed using the guidelines in Reference 3C-2.

3C.4.4 Operating Time

Equipment required to be environmentally qualified has one or more of the following functions: reactivity control, decay heat removal, post-accident monitoring, containment isolation, maintenance of RCS pressure boundary integrity, control room habitability, event severity mitigation, or system support functions. For each function, a period of operability is assigned that ranges from one hour to a maximum of 720 hours. The assignment of these post-accident operating times is separated into the four distinct time frames related to plant status or system functional requirements. Table 3C-4 summarizes these operating time designations and durations.

Equipment that performs its credited function before significant changes in its environment may be qualified for shorter durations. Justification for establishment of shorter durations is in accordance with Regulatory Guide 1.89.

Table 3.11-1 specifies post-accident operating times for equipment to be qualified.

3C.4.5 Treatment of Failures

If a component fails during qualification testing, even in a so called fail safe mode, the test should be considered inconclusive with regard to demonstrating the ability of the component to function for the entire period before the failure.

The basis for this position is related to the fact environmental qualification testing is not statistically representative. There should be no credit for the entire time up to the point of failure because it would be inconclusive if another specimen would fail earlier than the initial failure. However, credit could be taken for some period of time well before the failure with proper justification.

3C.5 Qualification Degradation Mechanisms

3C.5.1 Aging

Equipment is qualified for aging by testing and analysis. The qualification process considers natural aging effects present during the installed service life of the equipment. The objective of the qualification program is to place the test specimen(s) in an end of life condition before exposure to simulated accident conditions. Significant types of degradation that can affect the ability of the equipment to perform its credited function during or following exposure to harsh environmental conditions must be considered in the qualification process. Typical aging mechanisms that are part of a qualification test program include

- thermal aging or thermal degradation.
- radiation aging.
- cyclic aging or wear related degradation.

Periodic inspection, testing, and calibration can monitor equipment for aging effects that are otherwise difficult to quantify or are not able to be fully simulated by the accelerated aging applied during a qualification test program.

The concept of condition based qualification may supplement the concept of qualified life. As the qualified life of the equipment approaches the end of its theoretical qualified life, periodic condition monitoring may be implemented to determine if actual aging is occurring at a slower rate such that further qualified service is possible based on the condition monitoring results. The use of condition monitoring is tied to the ability to monitor one or more condition indicators to determine whether equipment remains in a qualified condition. The trend of the condition indicator is determined during the performance of age conditioning of the test specimen during the qualification testing. The condition indicator must be measurable, linked to functional degradation of the qualified equipment, and have a consistent trend from unaged through the limit of the qualified pre-accident condition.

3C.5.1.1 Thermal Aging

As stated in NUREG-0588, the Arrhenius methodology is an acceptable method of addressing accelerated thermal aging. The development of the accelerated thermal aging parameters and activation energies consider or are based on the applicable guidance in IEEE Standard 1 (Reference 3C-6), IEEE Standard 98 (Reference 3C-7), IEEE Standard 99 (Reference 3C-8), and IEEE Standard 101 (Reference 3C-9). The selection of activation energies is based on material properties representative of the credited function of the item. Justification is provided for use of thermogrametric analysis to establish an activation energy that demonstrates that the resulting qualified life is conservative or representative of actual degradation under normal service conditions.

The minimum acceptable accelerated aging time is established in accordance with Reference 3C-2 or associated daughter standards. For thermal aging of materials where diffusion limited oxidation effects have the potential to not fully simulate actual thermal aging degradation effects, the thermal acceleration rates are adjusted to minimize or otherwise account for these effects.

3C.5.1.2 Radiation Aging

Radiation aging may be performed separately from the accident radiation exposure or the accident radiation exposure may be performed as part of the radiation aging. Radiation aging is performed using a radiation source as specified in the equipment specification. The maximum acceptable dose rate is established in accordance with Reference 3C-2. For radiation aging of materials where diffusion limited oxidation effects have the potential to not fully simulate actual aging degradation effects from irradiation, the dose rates are adjusted to minimize or otherwise account for these effects.

3C.5.1.3 Cyclic Wear Aging

Cyclic wear aging simulates electrical or mechanical degradation of the equipment due to normal operation of the equipment. This aging simulates wear related degradation as well as fatigue effects. The definition of the required number of cycles to be simulated during the qualification test program considers expected service conditions, and is based on a conservative estimation of equipment cycles during power operation, module startup, module shutdown, outages, maintenance activities and surveillance activities.

3C.5.2 Significant Aging Mechanisms

Review of equipment is performed and considers the design, function, materials, and environment for its specified application to identify potentially significant aging mechanisms. As specified in IEEE Standard 627-2010 (Reference 3C-5), the assessment of equipment aging effects is an essential part of the qualification process to determine if aging has a significant effect on operability. The

assessment includes an analysis of the equipment to determine significant aging mechanisms. When these mechanisms are identified, a suitable aging program is developed. This program, in conjunction with other parts of the qualification program, provides assurance that significant aging mechanisms are unlikely to contribute to common-mode failures adverse to the credited function of the equipment. When natural aging is utilized in the qualification program, it may not be necessary to conduct a detailed analysis to determine significant aging mechanisms.

3C.5.3 Synergistic Effects

Environmental qualification in accordance 10 CFR 50.49 requires synergistic effects be considered. Regulatory Guide 1.89, Section C.5.a provides further guidance for addressing synergisms.

The synergistic relationship among multiple stresses usually cannot be deduced from physical principles; rather, an experimental approach must be employed. Synergistic stresses usually require extensive testing to reveal their magnitudes, because most interaction effects are minute by comparison to the primary effects, and thus require significantly more experimental evidence to identify. Current research, as referenced below, indicates that synergistic effects can typically be categorized under two main headings.

- Test sequence effects - The sequence in which radiation and thermal aging exposures occur is an important consideration. Sequential and simultaneous tests can produce variances in degradation. Radiation combined with elevated temperatures, or radiation followed by elevated temperatures may produce more material degradation than when thermal aging precedes radiation exposure (NUREG/CR-3629). The possibility that significant synergistic effects may exist is addressed by using the worst-case aging sequence, conservative accelerated aging parameters, and conservative DBE test levels to provide confidence that synergistic effects are enveloped.
- Radiation dose rate effects - The need for qualification because of radiation exposure is evaluated for each piece of equipment. The radiation environment is based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional. For many materials, it has been observed that lower dose rates produce more degradation than higher dose rates for the same total applied dose (NUREG/CR-2157).

3C.5.4 Loss of Ventilation

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

The normal and abnormal environmental conditions shown in Appendix 3C, Table 3C-6 and Table 3C-7, reflect anticipated normal and maximum conditions. The heating ventilation and air conditioning systems are nonsafety-related and are assumed to not be functional during DBEs.

3C.6 Qualification Methodology

A qualification plan defines tests, inspections, performance evaluation, acceptance criteria, and required analysis to demonstrate that, when called upon, the qualified equipment can perform its credited function for the required post-accident operating time.

Before equipment is qualified, its performance requirements, the environmental and operational service conditions to which it must be qualified, and the significance of aging degradation are identified. The equipment is then qualified using one or a combination of the following methods: testing, analysis, or operating experience. Although type testing is the preferred method of qualification, a qualification program may involve some combination of these methods.

3C.6.1 Type Testing

Type testing is conducted to verify equipment operates with no loss of ability to perform its credited function during DBEs. Type test conditions simulate specified service conditions. Appropriate margin must be added to DBE parameters if not otherwise included in the specified service conditions.

The type test program is designed to demonstrate equipment can perform its credited function within the acceptance criteria applicable for normal, abnormal, and DBE service conditions. The type test consists of a demonstration of credited functions under a planned sequence of environmental tests both before and after age conditioning (Reference 3C-2). Regulatory Guide 1.180 specifies electromagnetic compatibility design requirements for electromagnetic and radio-frequency interference and power surges for equipment and is independent of the Environmental Qualification Program.

In accordance with Reference 3C-2, a test plan is prepared at the beginning of the test program.

Similarity

Analysis may be employed to demonstrate that the test results obtained for one piece of equipment apply to a similar piece of equipment.

3C.6.2 Analysis

Analytical techniques are used in qualification in a variety of ways, including evaluating aging effects, demonstrating qualification for particular DBE conditions, and evaluating differences between installed and tested equipment. Qualification by analysis requires a logical assessment or a valid mathematical model of the equipment to be qualified. When quantitative analysis is used for qualification, it

needs to be supported by test data, operating experience, or physical laws of nature to demonstrate that the equipment can perform its credited function under specified conditions.

3C.6.3 Operating Experience

Operating experience can serve as a basis for the qualification of equipment.

In accordance with Reference 3C-2, auditable data are maintained for environmental qualification of equipment qualified on the basis of operating experience that addresses the following criteria.

- The equipment cited for operating experience is identical or justifiably similar to the equipment to be qualified.
- The equipment cited for operating experience has operated under service conditions that equal or exceed in severity service conditions for which the equipment is to be qualified, and has performed its design function related to safety under these conditions.
- The normal and abnormal service condition requirements were satisfied before the occurrence of the DBE conditions.
- Margin is considered in determining the accident service conditions for the equipment to be qualified.

In accordance with Reference 3C-3, operating experience can be used to address the qualification of mechanical equipment principally because of the severe process conditions experienced by mechanical equipment during normal service applications.

Operating experience is used on an infrequent basis to qualify electrical equipment to harsh environments, principally because LOCA-type pipe break accidents rarely occur.

3C.6.4 On-going Qualification

The Environmental Qualification Program may employ on-going qualification, though this method is not acceptable as a sole means for qualifying equipment for DBE conditions. Its use is generally limited to areas subjected to mild environment conditions. Supplemental test, analysis, or experience data are also required to address equipment operability and performance during and after a seismic DBE.

3C.6.5 Combination of Methods

Equipment may be qualified by test, analysis, previous operating experience, or any combination of these three methods. Using a combination of methods may be appropriate under a variety of circumstances, such as:

- equipment is too complex for analysis alone or too large for testing alone
- test data are available on samples of similar design and materials of different sizes, so extrapolation may be possible

- verification of a mathematical model using partial type test to determine mode shapes and resonant frequencies
- operating experience provides the basis for developing simulated aging techniques
- analysis of an assembly to determine the environment to which components are to be tested
- two subassemblies that have been tested and qualified separately are combined into a complete assembly, and analysis of certain parameters (e.g., individual subassemblies' error rates and response times) demonstrates that the combination is also qualified

The combined qualification demonstrates that the equipment can perform its credited function under DBE service conditions throughout its qualified life. Combined qualification provides auditable data by which the various primary qualification methods may be brought together to satisfy the qualification program requirements.

3C.7 Documentation

The documentation required for environmental qualification of equipment consists of an equipment specification and qualification report as described in the applicable codes and standards identified in Section 3.11.2. The EQRF documents the summaries and results of qualification tests for electrical and mechanical equipment and components. The EQRF provides the auditable documentation that demonstrates that the equipment is environmentally qualified for its application and can accomplish its credited function. The elements of the EQRF include: equipment identification, interfaces, qualified life, credited design functions, and service conditions following the guidance of IEEE Std. 323-1974 (Reference 3C-2) for harsh environment applications and IEEE Std. 323-2003 (Reference 3C-1) for mild environment applications.

3C.7.1 Mild Environment Documentation

The documents that demonstrate the qualification of equipment located in a mild environment, include design and purchase specifications, seismic test reports (if applicable), and an evaluation or certificate of conformance. The specifications contain a description of the functional requirements for a specific environmental zone during normal environmental conditions and anticipated operational occurrences.

3C.7.2 Harsh Environment Documentation

The qualification documentation for equipment located in a harsh environment demonstrates the equipment is qualified for its application, meets its specification requirements, and has its qualified life and periodic surveillance, maintenance or condition monitoring interval established. Data used to demonstrate the qualification of the equipment are pertinent to the application and is organized in a

understandable and traceable manner that permits independent auditing of the conclusions presented.

3C.8 References

3C-1 Institute of Electrical and Electronics Engineers, "Qualifying Class 1E Equipment for Nuclear Generating Stations," IEEE Standard 323-2003, Piscataway, NJ.

3C-2 Institute of Electrical and Electronics Engineers, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations," IEEE Standard 323-1974, Piscataway, NJ.

3C-3 American Society of Mechanical Engineers, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," ASME QME-1-2017, New York, NY.

3C-4 NuScale Power, LLC, “Accident Source Term Methodology,” TR-0915-17565-P-A, Revision 4.

3C-5 Institute of Electrical and Electronics Engineers, “IEEE Standard for Qualification of Equipment Used in Nuclear Facilities,” IEEE Standard 627-2010, Piscataway, NJ.

3C-6 Institute of Electrical and Electronics Engineers, "General Principles for Temperature Limits in the Rating of Electrical Equipment and for the Evaluation of Electrical Insulation," IEEE Standard 1-2000, Reaffirmed 2005, Piscataway, NJ.

3C-7 Institute of Electrical and Electronics Engineers, "The Preparation of Test Procedures for the Thermal Evaluation of Solid Electric Insulating Materials," IEEE Standard 98-2016, Piscataway, NJ.

3C-8 Institute of Electrical and Electronics Engineers, "Recommended Practice for the Preparation of Test Procedures for the Thermal Evaluation of Insulation Systems for Electric Equipment," IEEE Standard 99-2007, Piscataway, NJ.

3C-9 Institute of Electrical and Electronics Engineers, "IEEE Guide for the Statistical Analysis of Thermal Life Test Data," IEEE Standard 101-2004, Reaffirmed 2010, Piscataway, NJ.

Table 3C-1: Environmental Qualification Zones - Reactor Building

Zone	Description	Environment
Reactor Building Pool Area Zones		
RXBP-1	NP06-2-A-Room-042, Module pool bay vapor space - outside containment and under the bioshield (top of module) (Figure 1.2-8 Reactor Building 25 foot Elevation)	Harsh
RXBP-2	NP06-2-A-Room-040, 041, and 042, pool level to ceiling (RXB pool room vapor space external to top of module) (Figure 1.2-8 Reactor Building 25 foot elevation) and NP06-2-A-Room-435 (RXB pool room vapor space) (Figure 1.2-12 Reactor Building 85 foot elevation) and NP06-2-A-Room-516 (RXB pool room vapor space) (Figure 1.2-13 Reactor Building 100 foot elevation) and NP06-2-A-Room-607, 609, and 612 (RXB pool room vapor space) (Figure 1.2-16 Reactor Building 126 foot elevation)	Harsh
RXBP-3	NP06-2-A-Room-042 from top of liner plate up to top of pool level (surrounding each module under the bioshields) (Figure 1.2-8 Reactor Building 25 foot elevation)	Harsh
RXBP-4	NP06-2-A-Room-040, 041, and 042 from top of liner plate up to top of pool level (RXB pool room liquid space) (Figure 1.2-8 Reactor Building 25 foot elevation)	Harsh
Reactor Building Gallery Area Zones		
RXBG-1	NP06-2-A-Room-001 through 039 (Figure 1.2-8 Reactor Building 25 foot elevation) and NP06-2-A-Room-104, 105, and 131 (Figure 1.2-9 Reactor Building 40 foot elevation)	Harsh
RXBG-2	NP06-2-A-Room-101, 102, 103, 106 through 130, 132 and 133 (Figure 1.2-9 Reactor Building 40 foot elevation)	Harsh
RXBG-3	NP06-2-A-Room-201 through 223 (Figure 1.2-10 Reactor Building 55 foot elevation)	Harsh
RXBG-4	NP06-2-A-Room-301 through 305, 316, 317, 318, and 329 through 333 (Figure 1.2-11 Reactor Building 70 foot elevation)	Harsh
RXBG-5	NP06-2-A-Room-306 through 315 and 319 through 328 (Figure 1.2-11 Reactor Building 70 foot elevation)	Mild
RXBG-6	NP06-2-A-Room-401 through 407, 418, 419, 420, 431 through 434, and 436 (Figure 1.2-12 Reactor Building 85 foot Elevation)	Harsh
RXBG-7	NP06-2-A-Room-408 through 417 and 421 through 430 (Figure 1.2-12 Reactor Building 85 foot elevation)	Mild
RXBG-8	NP06-2-A-Room-501 through 515 (Figure 1.2-13 Reactor Building 100 foot elevation)	Harsh
RXBG-9	NP06-2-A-Room-601 through 606, 608, 610, 611, 613, 614, and 615 (Figure 1.2-14 Reactor Building 126 foot elevation)	Harsh
RXBG-10	NP06-2-A-Room-701, 702, 703 and 704 (Figure 1.2-15 Reactor Building 146 foot 6 inch elevation)	Harsh
Containment Vessel Zones		
CNV-1	NP06-2-A-Room 042, Containment Vessel - bottom of containment to bottom of upper core plate	Harsh
CNV-2	NP06-2-A-Room 042, Containment Vessel - bottom of upper core plate to lower riser top	Harsh
CNV-3	NP06-2-A-Room 042, Containment Vessel - lower riser top to upper riser top	Harsh

Table 3C-1: Environmental Qualification Zones - Reactor Building (Continued)

Zone	Description	Environment
CNV-4	NP06-2-A-Room 042, Containment Vessel - upper riser top to top of RPV head	Harsh
CNV-5	NP06-2-A-Room 042, Containment Vessel - top of RPV head to bottom of torispherical head	Harsh
CNV-6	NP06-2-A-Room 042, Containment Vessel - bottom of torispherical head to top of containment	Harsh

Table 3C-2: Designated Harsh Environment Areas

Area	Basis	Comment/Remarks
RXBP-1	Steam Exposure Harsh environment as a result of primary and secondary HELBs that occur in this area Total integrated dose (60 yrs. + accident) > 1.0E4 rad	None
RXBP-2	Steam Exposure Harsh environment as a result of primary and secondary HELBs that occur in the top of module ≥ 120°F and > 18°F increase above normal operating conditions with RH ≥85% Total integrated dose (60 yrs. + accident) > 1.0E4 rad	None
RXBP-3	Harsh environment as a result of primary and secondary HELBs that occur at the top of module ≥ 120°F and > 18°F increase above normal operating conditions due to DBA inside the CNV Total integrated dose (60 yrs. + accident) > 1.0E4 rad	None
RXBP-4	Harsh environment as a result of primary and secondary HELBs that occur at the top of module ≥ 120°F and > 18°F increase above normal operating conditions due to DBA inside the CNV	None
RXBG-1 through RXBG-4, RXBG-6, and RXBG-8 through RXBG-10	These areas contain high and moderate energy piping.	The note in Table 3C-3 clarifies classification of zones RXBG-5 and RXBG-7.

Note: The Radioactive Waste Building and Turbine Generator Building are considered Harsh Non-Environmental Qualification Zones.

Table 3C-3: Designated Mild Environment Areas

Area	Basis	Comment/Remarks
CRB-1 through CRB-4	No harsh environment DBA or IE are postulated to occur in this building. Total integrated dose (60 years + accident) ≤1.0E3 rad Control Building (CRB) does not contain high energy piping systems (>200°F or > 275 psig) and flooding analysis demonstrates that no equipment required to perform a credited function is submerged Maximum temperature is 120°F with humidity 85%	Satisfies mild environment criteria with the exception of exceeding the 1.0 E+03 rad threshold for electronics located in the elevation 100' mechanical equipment room (Room 003).
RXBG-5 and RXBG-7	No harsh environment DBA or IE are postulated to occur in these rooms. Total integrated dose (60 years + accident ≤1.0E4 rad) for electrical equipment. Maximum temperature is 120°F with humidity 85%	Satisfies mild environment criteria with the exception of exceeding 1.0 E+03 rad threshold for electronics located in the corridors (Rooms 306, 319, 408, and 421).

Table 3C-4: Equipment Post-Accident Operating Times

Description	Time Frame (hours)	Actions Accomplished	Basis
Short Term (ST)	≤ 1	• Event detection • Initiation of trip and engineered safety features actuation • Achievement of hot shutdown	Note 1
Intermediate Term (IT)	ST ≤ IT ≤ 36	• Achievement of safe shutdown • RCS depressurization and cooldown • Maintain fission product barrier integrity	Note 2
Long Term (LT)	IT ≤ LT ≤ 72	• Maintaining safe shutdown • Maintain fission product barrier integrity	Note 3
Extended Term (ET)	LT ≤ ET ≤ 720	• Maintaining safe shutdown • Maintain fission product barrier integrity • Monitoring of fission product barrier integrity	Note 4

Notes:

1. The short term post-accident operating time is assigned to components associated with event detection, reactor trip initiation, or engineered safety features actuation that occur very early in the accident sequence. This includes the module protection system initiation of
   - reactor trip.
   - containment isolation.
   - decay heat removal system actuation,
   - emergency core cooling system actuation.
   - de-energizing the pressurizer heaters.
   - isolation of demineralized water.

   Short term actions are also associated with the achievement of hot shutdown.
2. Intermediate term actions are associated with the achievement of safe shutdown using decay heat removal system. The intermediate term time frame extends to 36 hours and is used to qualify equipment that is relied upon to support the emergency core cooling system hold for up to 24 hours.
   Examples of equipment assigned an intermediate term post-accident operating time includes:
   - reactor vent valves
   - reactor recirculation valves
3. The long term time frame extends to 72 hours. This category is considered the maximum post-accident operating time for HELB and MELB events outside containment in areas that are readily accessible after break termination or isolation.
   Examples of equipment assigned to this category includes the following.
   - Equipment that is relied upon to mitigate a HELB or MELB outside containment that are located in the top of module area (outside containment and under bioshield).
   - Augmented DC power system batteries for separation groups B and C which are sized to support an extended loss of AC power for up to 72 hours.
4. The extended time frame of 720 hours represents the maximum post-accident operating time used to qualify equipment that is relied upon to maintain a safe shutdown condition. Equipment assigned to this post-accident operating time category are typically located inside the CNV or in an inaccessible area outside of containment, such as under the bioshield.

Table 3C-5: Environmental Qualification Program Margin Requirements

Parameter	Required Margin	Notes
Peak Temperature	+15°F	Applied to accident profile peaks only
Peak Pressure	+10% of gauge, but not more than 10 psig for equipment external to containment +10% of gauge, but not more than 100 psig for equipment internal to containment.	Peak pressure is modified for containment as the containment operates at much higher conditions compared to legacy pressurized water reactors.
Radiation	+10%	On accident dose only
Power Supply Voltage	±10%	Of rated value, not to exceed equipment design limits.
Equipment Operating Time	+10%	For the period of time the equipment is required to operate following the start of a DBE. Section 3C.4.4 and Table 3C-4.
Seismic Vibration	+10%	Margin added to acceleration requirements at the mounting point of equipment.
Line Frequency	N/A	Line frequency margin is not applicable because the relied upon electrical power is from augmented DC power.
Time	+10%	In addition to the period of time the equipment is required to be operational following the DBE.
Environmental Transients	2 or more	The initial transient and the dwell peak temperature shall be applied at least twice

Table 3C-6: Normal Operating Environmental Conditions

Zone	Temperature (°F)	Pressure (psig) (Nominal)	Maximum Relative Humidity (%)	60 Years Integrated N Dose (Rads)	60 Years Integrated γ Dose (Rads) (Includes fission γ, N-γ, coolant)	Water Level (feet above RXB pool floor)
CNV-1	Figure 3C-2 (1)	-14.684	0	9.00E+11	1.75E+11	Note 3
CNV-2	Figure 3C-2 (1)	-14.684	0	3.25E+11	5.50E+10	Note 3
CNV-3	Figure 3C-4 (2)	-14.684	0	4.75E+09	2.00E+08	Note 3
CNV-4	Figure 3C-4 (2)	-14.684	0	1.50E+08	1.75E+07	Note 3
CNV-5	Figure 3C-4 (2)	-14.684	0	6.50E+07	8.50E+06	Note 3
CNV-6	Figure 3C-4 (2)	-14.684	0	3.90E+07	1.75E+06	Note 3
RXBP-1	140	0	≤ 85	2.00E+07	3.50+E06	0
RXBP-2	140	0	≤ 85	2.50E+04	4.00E+04	0
RXBP-3	120	0 + submergence head	-	6.50E+10	2.50E+10	53 ± 1
RXBP-4	120	0 + submergence head	-	1.50E+06	1.75E+08	53 ± 1
RXBG-1	105	0	≤ 85	-	1.00E+08	-
RXBG-2	105	0	≤ 85	-	5.10E+04	-
RXBG-3	105	0	≤ 85	-	2.50E+05	-
RXBG-4	105	0	≤ 85	-	6.10E+02	-
RXBG-5	85	0	≤ 85	-	2.00E+03	-
RXBG-6	105	0	≤ 85	-	1.75E+02	-
RXBG-7	85	0	≤ 85	-	3.00E+03	-
RXBG-8	105	0	≤ 85	-	3.00E+04	-
RXBG-9	105	0	≤ 85	-	2.50E+02	-
RXBG-10	105	0	≤ 85	-	0.00E+00	-
CRB-1	68	4.51E-03 (4)	≤ 85	-	-	-
CRB-2	68	4.51E-03 (4)	≤ 85	-	-	-
CRB-3	68	4.51E-03 (4)	≤ 85	-	-	-
CRB-4	68	4.51E-03 (4)	≤ 85	-	-	-

Notes: 1. Reactor pressure vessel (RPV) wall temperatures are shown as pressurizer (PZR) wall temperatures on Figure 3C-2 as the PZR wall temperature bounds the RPV temperatures at saturation conditions. CNV wall temperatures are shown Figure 3C-2.

2. Reactor pressure vessel wall temperatures are shown as PZR wall temperatures on Figure 3C-3 as the PZR wall temperature bounds the RPV temperatures at saturation conditions. CNV wall temperatures are shown Figure 3C-4.
3. Containment water level is 0 feet for normal power operation and is a maximum of 55 feet for all other conditions. For anticipated operational occurrence conditions the water level is 19 feet.
4. The CRB is maintained at a slightly positive air pressure to control ingress of airborne reactivity from the RXB or atmosphere outside of the CRB.

Table 3C-7: Design Basis Event Environmental Conditions

Zone	DBE	Temperature (°F)[1][2][3]	DBE	Pressure (psig)	DBE	Relative Humidity (%)	Water Level (feet above RXB pool floor)[4]	Pipe Rupture (water spray)
CNV-1	All Events	Figure 3C-1 and Figure 3C-2	All Events	1005	All Events	100	55	-
CNV-2	All Events	Figure 3C-1 and Figure 3C-2	All Events	1005	All Events	100	55	-
CNV-3	All Events	Figure 3C-3 and Figure 3C-4	All Events	1005	All Events	100	55	Yes
CNV-4	All Events	Figure 3C-3 and Figure 3C-4	All Events	1005	All Events	100	55	Yes
CNV-5	All Events	Figure 3C-3 and Figure 3C-4	All Events	1005	All Events	100	55	Yes
CNV-6	All Events	Figure 3C-3 and Figure 3C-4	All Events	1005	All Events	100	55	Yes
RXBP-1	HELB	Figure 3C-5	HELB	≤ 6	All Events	100	55	Yes
RXBP-2	HELB	Figure 3C-6	HELB	≤ 6	All Events	100	55	-
RXBP-3	HELB	≤ 212	HELB	≤ 6 + submergence head	All Events	-	55	-
RXBP-4	HELB	≤ 212	HELB	≤ 6 + submergence head	All Events	-	55	-
RXBG-1	HELB	199	HELB	≤ 6	All Events	100	15	Yes
RXBG-2	HELB	180	HELB	≤ 6	All Events	100	12.75	Yes
RXBG-3	HELB	223	HELB	≤ 6	All Events	100	12.75	Yes
RXBG-4	HELB	163	HELB	≤ 6	All Events	100	7.8	Yes
RXBG-5	HELB	≤ 120	HELB	No appreciable change from normal	All Events	≤ 85	0	-
RXBG-6	HELB	170	HELB	≤ 6	All Events	100	12.8	Yes
RXBG-7	HELB	≤ 120	HELB	No appreciable change from normal	All Events	≤ 85	0	-
RXBG-8	HELB	393	HELB	≤ 6	All Events	100	5.9	Yes
RXBG-9	HELB	175	HELB	≤ 6	All Events	100	4.2	-
RXBG-10	HELB	134	HELB	≤ 6	All Events	100	5.4	-

Table 3C-7: Design Basis Event Environmental Conditions (Continued)

Zone	DBE	Temperature (°F)[1][2][3]	DBE	Pressure (psig)	DBE	Relative Humidity (%)	Water Level (feet above RXB pool floor)[4]	Pipe Rupture (water spray)
CRB-1	Passive Cooling	≤ 120	All Events	No appreciable change from normal	All Events	≤ 85	0.5	-
CRB-2	Passive Cooling	≤ 120	All Events	No appreciable change from normal	All Events	≤ 85	0.6	-
CRB-3	Passive Cooling	≤ 120	All Events	No appreciable change from normal	All Events	≤ 85	1	-
CRB-4	Passive Cooling	≤ 120	All Events	No appreciable change from normal	All Events	≤ 85	1.9	-

Notes: 1. The temperature listed represents the highest temperature of any room (outside of containment) in a given zone.
2. Reactor pressure vessel wall temperatures are shown as PZR wall temperatures on Figure 3C-2 as the PZR wall temperature bounds the RPV temperatures at saturation conditions. CNV wall temperatures are shown Figure 3C-2.
3. Reactor pressure vessel wall temperatures are shown as PZR wall temperatures on Figure 3C-3 as the PZR wall temperature bounds the RPV temperatures at saturation conditions. CNV wall temperatures are shown Figure 3C-4.
4. The water level listed represents the highest water level of any room (outside of containment) in the zone.
5. The CNV post-accident pH for any postulated accident that results in core damage is 6.9 at 1000 ppm boron concentration and 7.3 at 200 ppm boron concentration. These values remain essentially unchanged between 25 degree C and 200 degree C.

Table 3C-8: Limiting Design Basis Accident EQ Radiation Dose

Zone	Dose	Accident Integrated Dose (rads)			
		1 hour	36 hours	72 hours	720 hours
CNV-1	Integrated ß	1.75E+02	2.00E+03	2.75E+03	5.00E+03
	Integrated γ	6.00E+02	4.75E+03	6.00E+03	1.25E+04
CNV-2	Integrated ß	6.75E+04	8.50E+05	1.25E+06	3.50E+06
	Integrated γ	2.25E+05	2.00E+06	2.50E+06	4.25E+06
CNV-3	Integrated ß	6.75E+04	8.50E+05	1.25E+06	3.50E+06
	Integrated γ	2.25E+05	2.00E+06	2.50E+06	4.25E+06
CNV-4	Integrated ß	6.75E+04	8.50E+05	1.25E+06	3.50E+06
	Integrated γ	2.25E+05	2.00E+06	2.50E+06	4.25E+06
CNV-5	Integrated ß	6.75E+04	8.50E+05	1.25E+06	3.50E+06
	Integrated γ	2.25E+05	2.00E+06	2.50E+06	4.25E+06
CNV-6	Integrated ß	6.75E+04	8.50E+05	1.25E+06	3.50E+06
	Integrated γ	2.25E+05	2.00E+06	2.50E+06	4.25E+06
RXBP-1	Integrated ß	8.25E+00	3.00E+02	6.00E+02	8.25E+03
	Integrated γ	2.75E+03	2.25E+04	3.00E+04	5.75E+04
RXBP-2	Integrated ß	1.25E-01	4.00E+00	7.75E+00	1.25E+02
	Integrated γ	2.75E+03	2.25E+04	2.75E+04	5.00E+04
RXBP-3	Integrated ß	1.25E-01	4.00E+00	7.75E+00	1.25E+02
	Integrated γ	2.75E+03	2.25E+04	2.75E+04	5.00E+04
RXBP-4	Integrated ß	1.25E-01	4.00E+00	8.00E+00	1.25E+02
	Integrated γ	2.75E+03	2.25E+04	2.75E+04	5.00E+04
RXBG-1	Integrated ß	3.00E+03	1.25E+05	1.50E+05	3.25E+05
	Integrated γ	8.75E-01	2.50E+01	3.00E+01	6.00E+01
RXBG-2	Integrated ß	3.00E+03	1.25E+05	1.50E+05	3.25E+05
	Integrated γ	8.75E-01	2.50E+01	3.00E+01	6.00E+01
RXBG-3	Integrated ß	3.00E+03	1.25E+05	1.50E+05	3.25E+05
	Integrated γ	1.00E+02	2.75E+03	3.25E+03	6.25E+03
RXBG-4	Integrated ß	4.50E+02	2.00E+04	2.25E+04	4.75E+04
	Integrated γ	8.75E-01	2.50E+01	3.25E+01	6.25E+01
RXBG-5	Integrated ß	-	-	-	-
	Integrated γ	≤ 1.00E+04	≤ 1.00E+04	≤ 1.00E+04	≤ 1.00E+04
RXBG-6	Integrated ß	3.50E+02	1.50E+04	1.75E+04	3.75E+04
	Integrated γ	1.25E+00	3.00E+01	4.00E+01	7.75E+01
RXBG-7	Integrated ß	-	-	-	-
	Integrated γ	≤ 1.00E+04	≤ 1.00E+04	≤ 1.00E+04	≤ 1.00E+04
RXBG-8	Integrated ß	9.00E+02	3.75E+04	4.50E+04	9.50E+04
	Integrated γ	8.75E+01	2.25E+03	3.00E+03	5.50E+03
RXBG-9	Integrated ß	9.00E+02	3.75E+04	4.50E+04	9.50E+04
	Integrated γ	3.00E+00	7.75E+01	1.00E+02	2.00E+02
RXBG-10	Integrated ß	9.00E+02	3.75E+04	4.50E+04	9.50E+04
	Integrated γ	3.00E+00	7.75E+01	1.00E+02	2.00E+02

Table 3C-8: Limiting Design Basis Accident EQ Radiation Dose (Continued)

Zone	Dose	Accident Integrated Dose (rads)			
		1 hour	36 hours	72 hours	720 hours
CRB-1	Integrated ß	5.81E+00	5.81E+00	5.81E+00	5.81E+00
	Integrated γ	5.42E+00	8.53E+00	1.05E+01	2.23E+01
CRB-2	Integrated ß	5.81E+00	5.81E+00	5.81E+00	5.81E+00
	Integrated γ	5.42E+00	8.53E+00	1.05E+01	2.23E+01
CRB-3	Integrated ß	2.07E+01	7.20E+02	1.17E+03	3.79E+03
	Integrated γ	5.42E+00	8.53E+00	1.05E+01	2.23E+01
CRB-4	Integrated ß	5.81E+00	5.81E+00	5.81E+00	5.81E+00
	Integrated γ	5.42E+00	8.53E+00	1.05E+01	2.23E+01

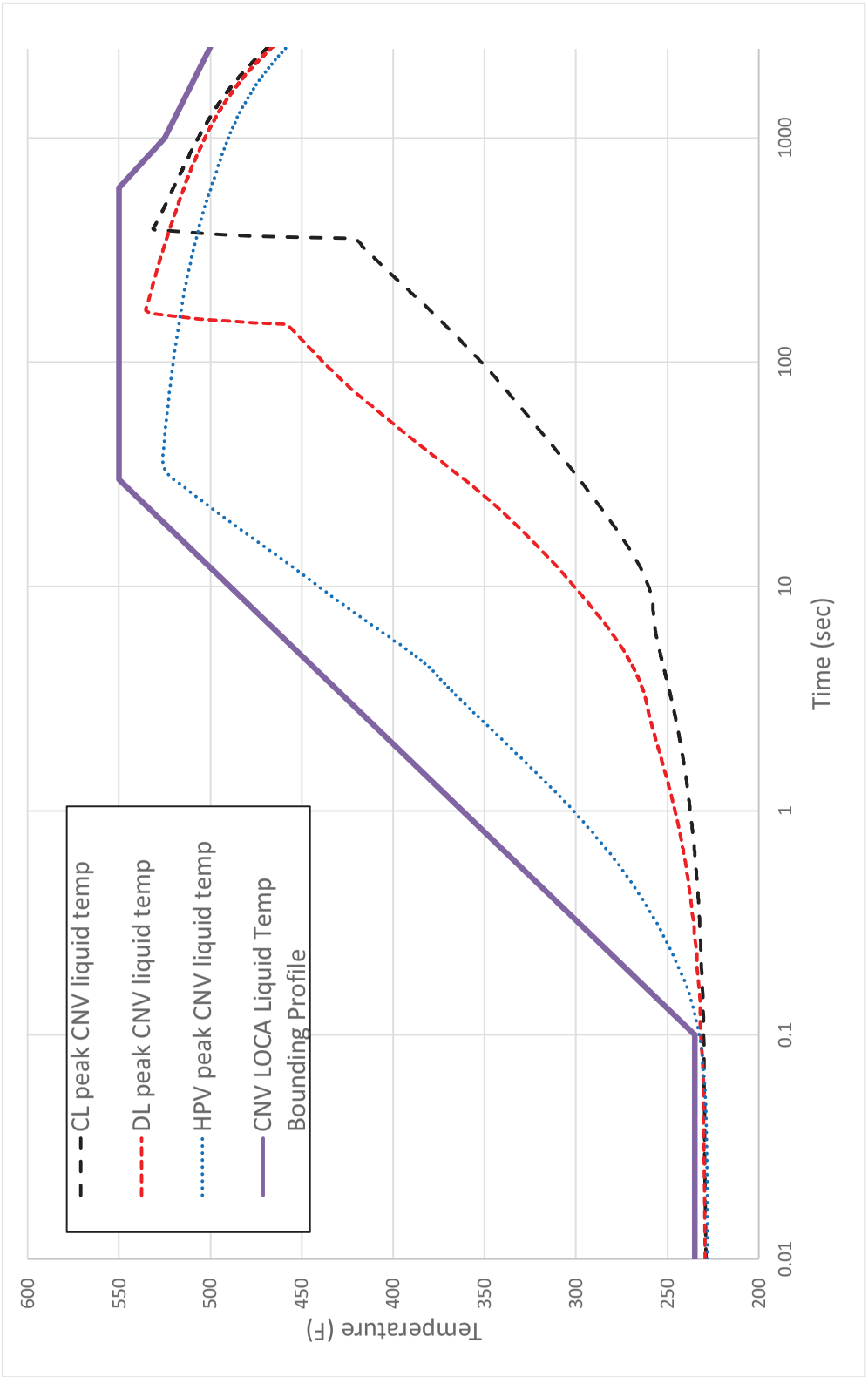


Figure 3C-1: Containment Liquid Space Liquid Temperatures with Bounding Curve (Zones CNV1 and CNV2)

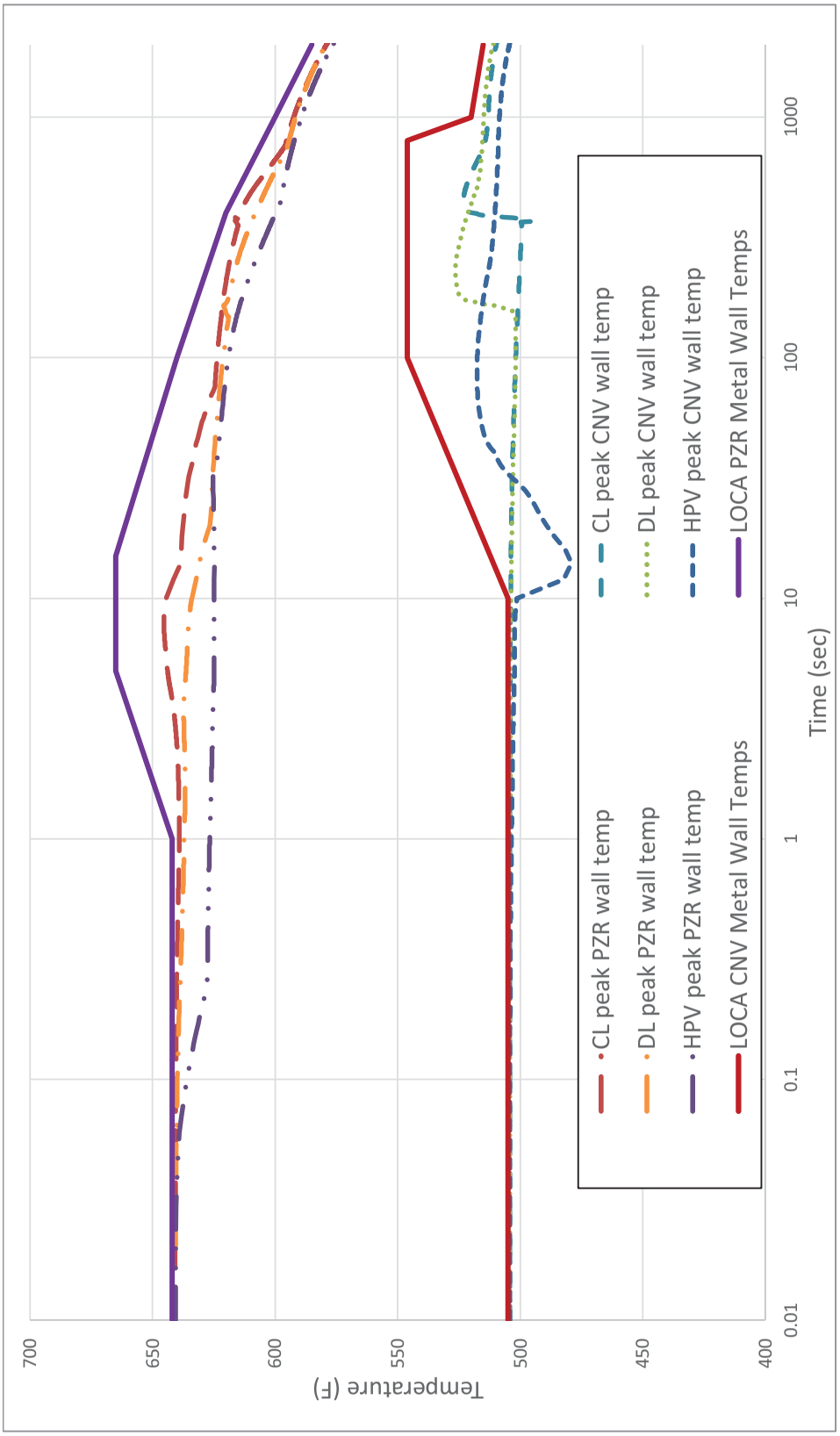


Figure 3C-2: Containment Liquid Space Wall Temperatures with Bounding Curve (Zones CNV1 and CNV2)

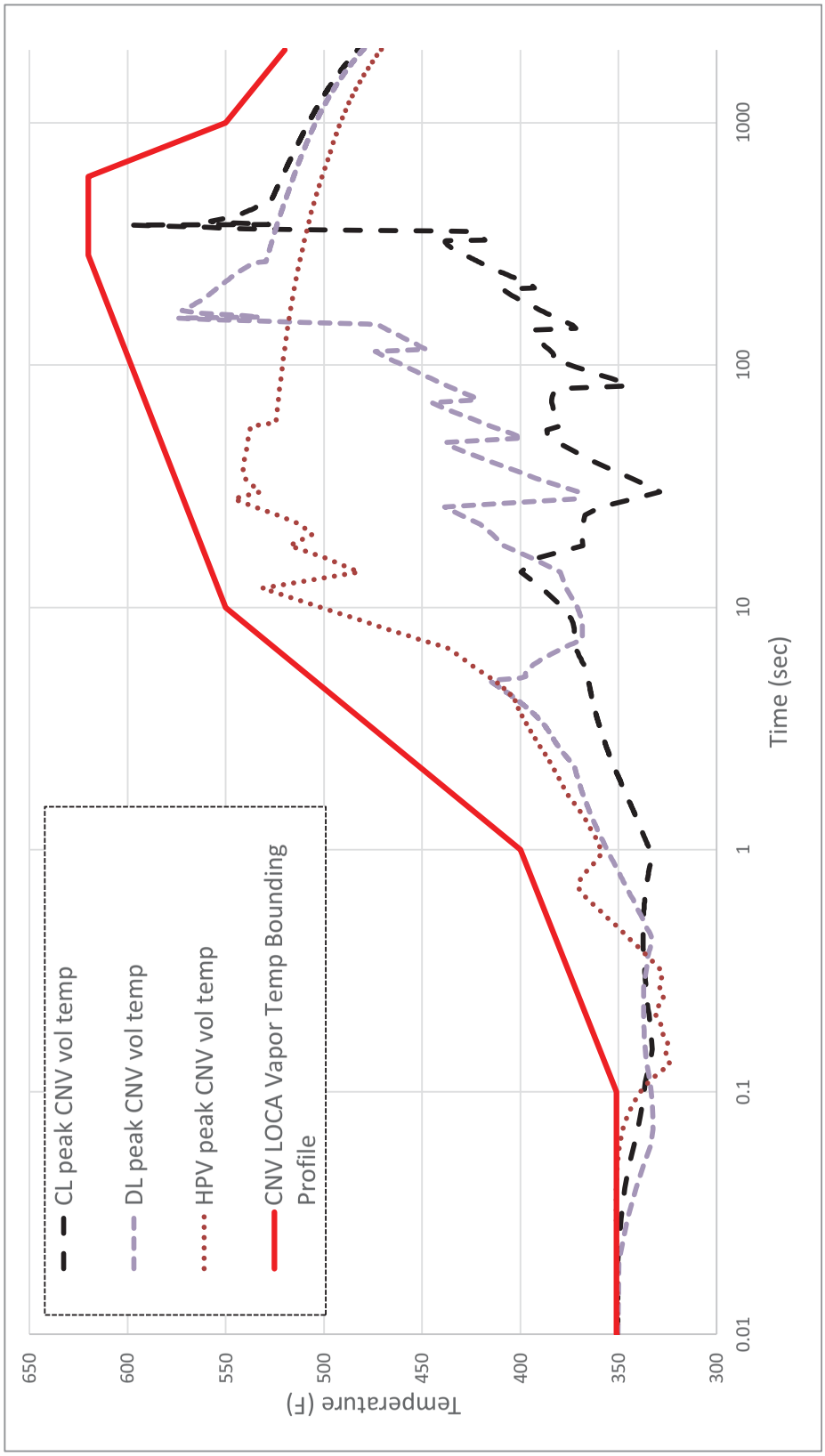


Figure 3C-3: Containment Vapor Space Gas Temperatures with Bounding Curve (Zones CNV-3 through CNV-6)

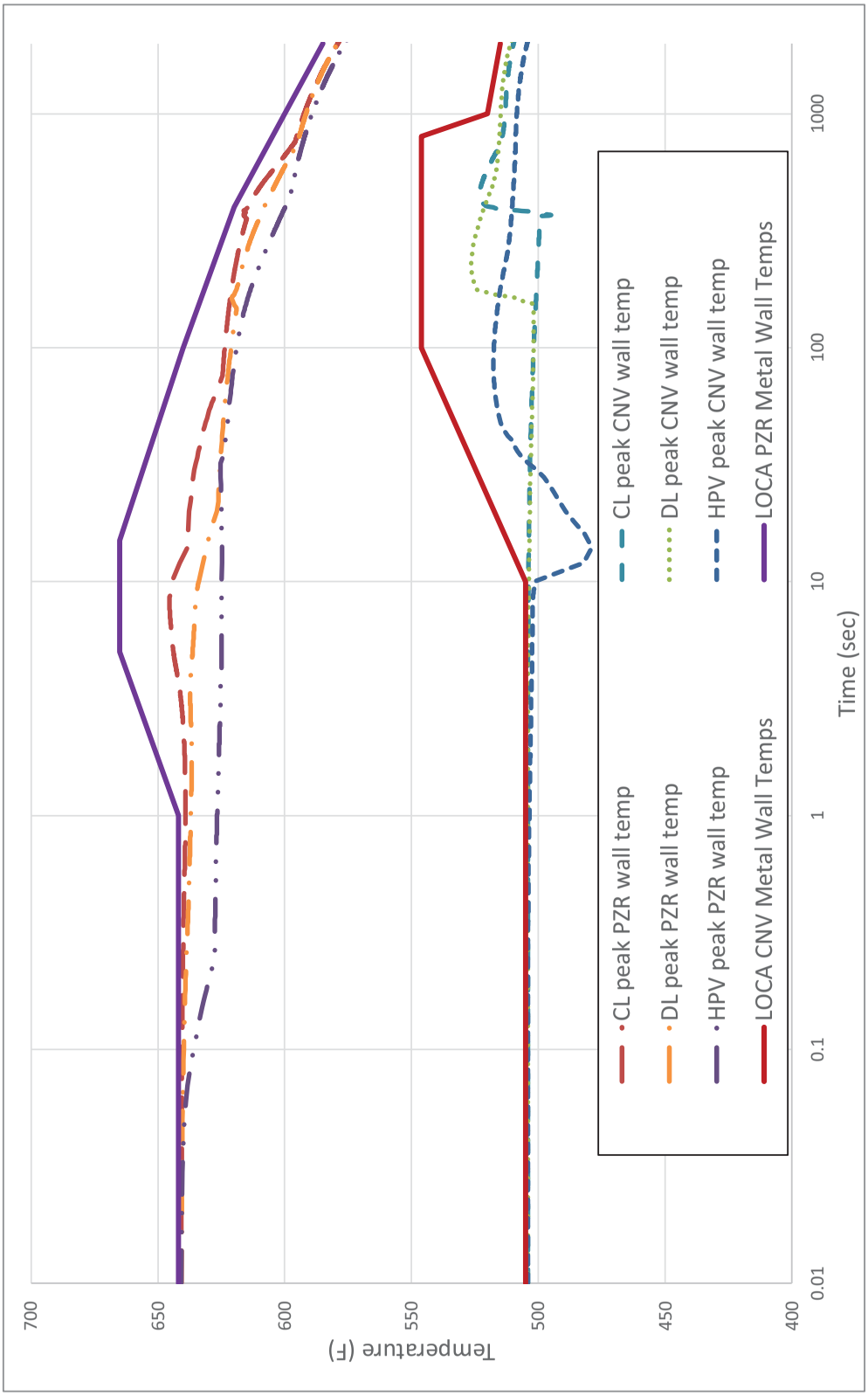


Figure 3C-4: Containment Vapor Space Wall Temperatures with Bounding Curve (Zones CNV-3 through CNV-6)

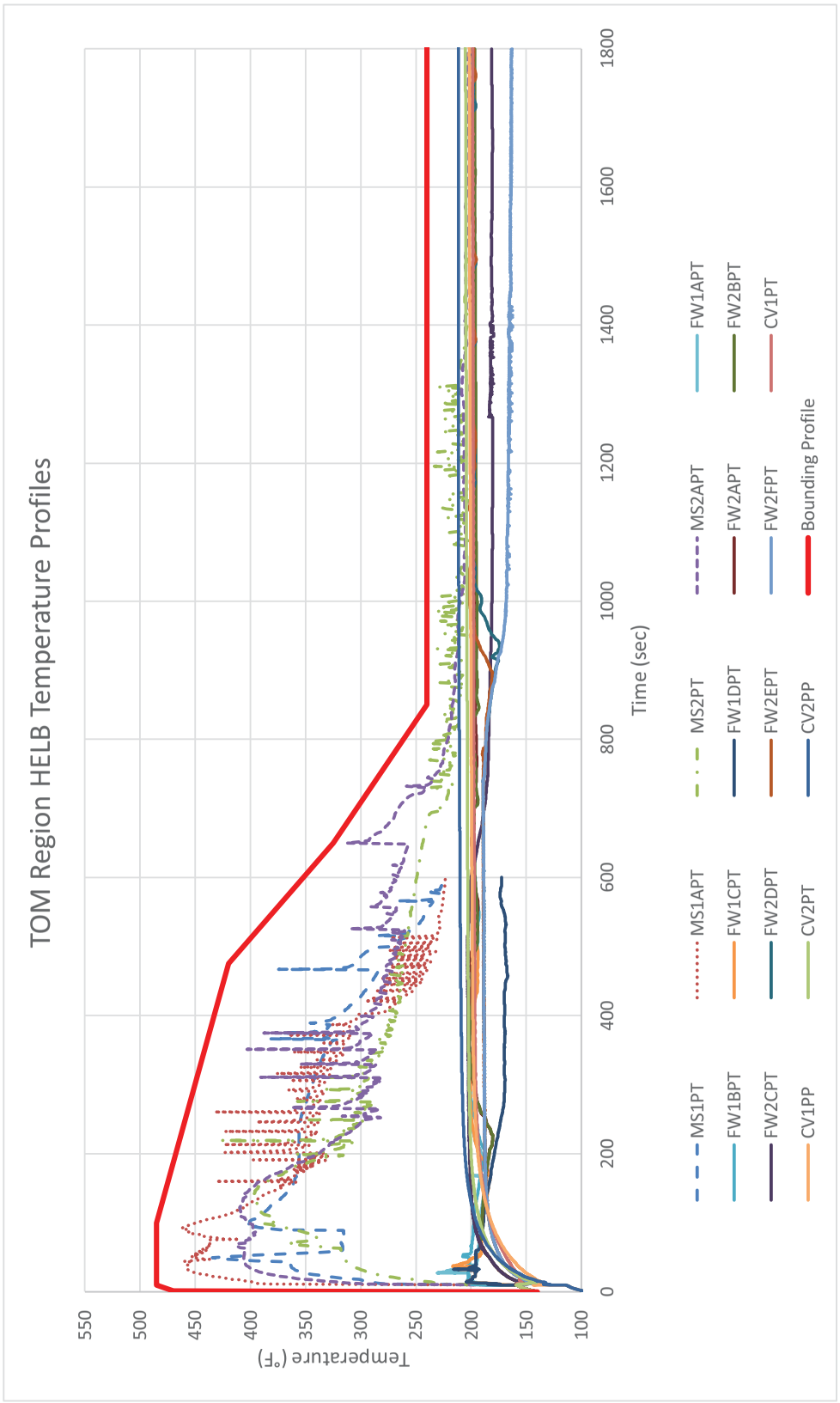


Figure 3C-5: Top of Module HELB Composite and Bounding Temperature Profile (Zone RXBP-1)

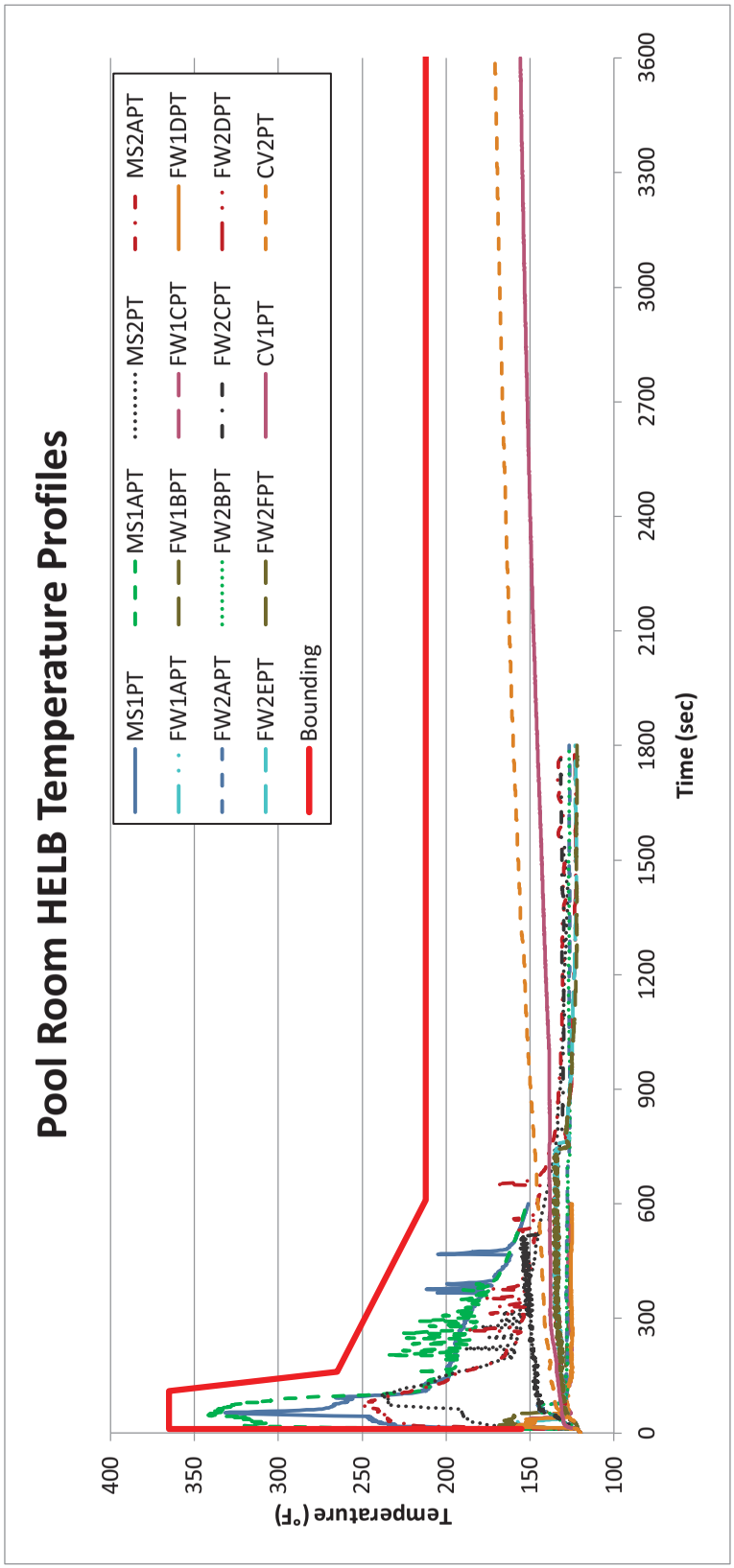


Figure 3C-6: Pool Room HELB Composite and Bounding Temperature Profile (Zone RXBP-2)

Enclosure 2:

Readiness Assessment Review responses for Chapter 3

The table below provides the NuScale responses to each of the Nuclear Regulatory Commission readiness assessment observations on draft Chapter 3, “Design of Structures, Systems, Components and Equipment,” of the Standard Design Approval Application.

Section	Observation	Response
3.2.1	The submittal should explain what the new design will entail given that the following paragraphs were removed from Section 3.2.1.4: *There are no Seismic Category I Structures, Systems and Components (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.*	The listed sections were removed because they repeated information that is provided in more technically relevant chapters of the Final Safety Analysis Report (FSAR). The sections were replaced with references to the appropriate chapters where the information can be found. A reference was added to Section 3.2.1.4 pointing to Chapter 11 for information regarding structures, systems, and components (SSC) used for radioactive waste.
3.2	Per Standard Review Plan (SRP) Section 3.2.1, “Seismic Classification,” Revision 3, staff reviews the seismic classification design criteria of those structure, system, and components (SSCs) that are important to safety and are specified as seismic Category I by the applicant’s safety analysis report (SAR) and designed to withstand, without loss of function, the effects of a safe shutdown earthquake (SSE). The review also covers identification of SSCs that are not required to remain functional following a seismic event, but whose failure could reduce the functioning of any seismic Category I SSCs (seismic Category II). For a SSC, the delineations between seismic Category I and Category II need to be identified. As an example, SDA Table 3.2-1 Seismic Classification of Building Structures lists the reactor building (RXB) as seismic Category I (as expected). However, the document in the Electronic Reading Room titled, “Design Changes Roadmap-7.15.pdf” reclassifies some floor and roof slabs above grade as seismic Category II. If the RXB contains	Table 3.2-1 was updated to reflect the correct seismic classifications of SC-I and SC-II, consistent with the information presented in FSAR Section 1.2.

Section	Observation	Response
	seismic Category II structures, the entire RXB should not be identified in Table 3.2-1 as seismic Category I without identifying the exceptions or providing justification (e.g., Seismic II/I interactions). The RXB discussion above is one example identified by the staff and it is recommended that the extent of condition is investigated because the staff do not have all the SAR chapters. The staff also factors this seismic information into the Chapter 19 evaluation of the expected probabilistic risk assessment (PRA) based seismic margins assessment.	
3.2.2	The submittal should explain what criteria will be used for the instrument sensing lines given that the following paragraph was removed from Section 3.2.2: *Safety-related instrument sensing lines are designed and constructed in accordance with ANSI/ISA-67.02.01-1999 (Reference 3.2-2) as described in RG 1.151. The standard ANSI/ISA-67.02.01-1999 establishes the applicable code requirements and code boundaries for the design and installation of instrument sensing lines interconnecting safety-related piping and vessels with both safety-related and nonsafety-related instrumentation. This is further discussed in Section 7.2.2.*	The listed sections were removed because they repeated information that is provided in more technically relevant chapters of the FSAR. The sections were replaced with pointers to the chapters where the information can be found. A reference was added to Section 3.2.2 pointing to Section 7.2 for information on instrument sensing lines.
3.2.2	The submittal should explain what design criteria will be used given that the following paragraph was removed from Section 3.2.2: *The reactor vessel internals (see Section 3.9.5) and steam generator supports, and tube supports (see Section 5.4.1.5) comply with the design and construction requirements of Subsection NG of Section III, Division 1 of the ASME BPVC (Reference 3.2-1).*	The listed sections were removed because they repeated information that is provided in more technically relevant chapters of the FSAR. The sections were replaced with references to the appropriate chapters where the information can be found. A reference was added to Section 3.2.2 pointing to Section 5.4 for information on reactor vessel internals, steam generator supports, and tube supports.
3.3.1	Clarify the maximum wind speed. The staff noted the wind speed 190 mph in Section 3.3.1.1 and the wind speed 145 mph in Section 3.3.1.2.	The correct operating wind speed is 190 mph. Section 3.3.1.1 has been revised to the correct value. Additional clarification was added to specify that the 270 mph value applies to the maximum wind speed for tornadoes and that the 290 mph value applies to the maximum wind speed for hurricanes.
3.4.1	Provide flood analysis methodology, assumptions, and SSC subject to flood protection for RXB Flood Analysis and control building (CRB) Flood Analysis.	The flood analyses for the Reactor Building (RXB) and the Control Building (CRB) were completed and added to Section 3.4.1. The information added to Section 3.4.1 includes the methodology, considerations, and assumptions used in the analyses. As described in COL Item 3.4-1, an applicant that references the NuScale Power Plant US460 standard design will confirm

Section	Observation	Response
		the final location of structures, systems, and components subject to flood protection.
3.5.1	Describe CRB as related to its missile protection barrier function or explain the reason for the elimination of this function being taken credit of in the design certification application (DCA).	For the Design Certification Application site layout, the CRB was located within the low trajectory zone. For the US460 standard design, the CRB is located outside the low trajectory zone and is not assessed for low trajectory turbine missiles. A sentence with this statement was added to Section 3.5.1.3.
3.5.1	Identify all the SSC subject to missile protection according to Regulatory Guide (RG) 1.115.	The SSC subject to missile protection are provided in Section 17.4. Section 3.5.1.3 was revised to include this statement. Since the CRB is located outside the low-trajectory turbine missile zone, a list of essential SSC located inside the building is not needed. A list of essential SSC located inside the RXB is not needed. Because essential SSC located inside the RXB are protected by barriers, the design does not need to credit physical separation of redundant safety-related equipment to meet General Design Criterion 4.
3.5.1	Understood that this section is still in development (Sections 3.5.1.3.3.1 and 3.5.1.3.3.2). However, the NRC staff would note that the information in Section 3.5.1.3 should be consistent with design control document, including using the same methodology for calculating missile, weight, speed, acceptance criteria, etc. In addition, the NRC staff notes that specific information has been deleted in this version that would be needed to come to a reasonable assurance finding, including the following: Section 3.5.1.3 needs to include a description on the barriers approach NuScale will use as in DCA. Specific wording quoted from RG 1.115 need not be included, but a description of how barriers will be used and acceptance criteria to be met should be included. Section 3.5.1.3.2 does not specify bounding missile speed. Combined License (COL) Item 3.5-2 was deleted and should be included in 3.5.1.3.4 to address turbine missiles from nearby or co-located facilities. Section 3.5.1.3.5 does not demonstrate that all essential SSCs are protected from postulated low-trajectory missiles as was previously done. Information needs to be included.	The design incorporates the use of steel-composite (SC) walls. The associated calculations and analyses are now complete. Section 3.5.1.3 includes bounding turbine missile properties, methodology, acceptance criteria, and a description of how barriers are used and essential SSC are protected from postulated low-trajectory missiles. The content of COL Item 3.5-2 is combined with, and included in, COL Item 3.5-3 to align it with related content. Figures 3.5-1 and 3.5-2 are revised to reflect the site layout, and location of barriers used for protection against low-trajectory turbine missiles. Information in Table 3.5-2 is now located within Section 3.5.1.3 or is considered unnecessary. Section 3.5, as revised, contains no tables.

Section	Observation	Response
	No information provided in Figures 3.5-1 and 3.5-2 for missile trajectory essential SSCs, etc. Table 3.5-2 does not have any information on NuScale turbine missile.	
3.8.2	In Subsection 3.8.2.3.6, identified “aircraft hazard,” and “explosion pressure waves” as the external environmental loadings. It is not clear why “aircraft hazard,” and “explosion pressure waves” were considered under the external environmental loadings.	For the US460 design the RXB is not a containment building. The containment for the design is a pressure vessel that surrounds the reactor pressure vessel. An “aircraft hazard” or “explosion pressure waves” that would result inside of the RXB would result in external loadings on the containment vessel (CNV). The US460 design considered aircraft impact and explosion loads but they are resisted by the RXB and therefore not applicable to the CNV.
3.8.2	In Subsections 3.8.2.3.9 provides discussion on the load combinations for the containment vessel (CNV). It is not clear why combinations of classified stresses (primary, secondary, peak) for comparison to specified code limits were not considered as provided in Figure XIII-2100-1 of American Society of Mechanical Engineers (ASME) Section III, Appendix XIII.	The load combinations discussed in Section 3.8.2 are to report the applicable combination of loadings that shall be considered in the corresponding ASME Code qualifications of individual components for various Service Levels. The classification of the stress (i.e., primary, secondary, or peak) resulting from the loading is dependent on the geometry and the location. Table XIII-2600-1 and Table XIII-2600-2 of Mandatory Appendix XIII of Section III of the ASME Code report how to appropriately classify stresses for various locations. In addition, Paragraph XIII-1220 of Mandatory Appendix XIII of Section III of the ASME Code provides the basis for determining stresses and how those stresses should be considered. The combination of Paragraph XIII-1220 and Tables XIII-2600-1 and XIII-2600-2 are utilized to classify the stresses for each of the load combinations.
3.8.2	In Subsections 3.8.2.4 and 3.8.2.4.1, the last sentence in second and third paragraphs, respectively, states, “Alternatively, limit analyses to determine lower bound limit buckling loads may be employed in lieu of Code Case N-759-2.” It is not clear which provision(s) in ASME Section III, Division 1, Subsection NB provides the design requirements for determining; the lower bound limit buckling loads using limit analyses.	Subparagraphs XIII-1220(k) and (l) of Mandatory Appendix XIII of Section III of the ASME Code define limit analysis and collapse load analysis, respectively. The evaluation of buckling, and elastic instability that results in collapse, is considered in a limit load analysis. FSAR Section 3.8.2 has been revised to reflect this clarification.
3.8.4	In Subsection 3.8.4.1.4.2, middle of the first paragraph states, “The liner is 304L, or equivalent, stainless steel that is 0.25 in. thick in most locations and covers the pool floor.” The requirements for designing SC walls are summarized in TR-0929-71621 with carbon-steel (CS) plates only and is described in Subsection 3.8.4.6.1.3. The current design standard/guidance may not support the construction of the RXB pool SC	The 0.25 in.-thick pool liner is only used over the floor surface and it is considered to be a non-structural component. In this regard, the current design standard and guidance remains applicable for the structural components. Structural

Section	Observation	Response
	walls with 0.25-inch-thick stainless steel (SS) liner plate (0.25-inch-thick SS liner is for maintaining the pool water inventory only, not providing structural integrity of the SC wall).	integrity of the SC walls is not impacted. FSAR Section 3.8.4 has been revised to support this response.
3.8.4	In Subsection 3.8.4.6.1.3, first sentence states, “The requirements for designing SC walls are summarized in TR-0929-71621.” The TR provides a design methodology of SC walls constructed using carbon-steel (CS) plates. The SC walls may also be constructed with SS pool liner and the CS plates and the tie-bars. This will create a detrimental degrading condition where the SS material will be the "noble-material," and the CS material will be the "sacrificial material." Furthermore, it is not clear, what would be the percent (%) of critical damping value for this SC configuration and there might be an issue related to the thermal expansions of SS vs. CS under accident conditions. Finally, the current design standard/guidance may not support the application of dissimilar materials to construct the SC walls.	The design of SC walls precludes accelerated corrosion. For accelerated corrosion to occur, both the carbon steel and the stainless steel need to be contacting the same electrolyte (water). The carbon steel and stainless steel do not contact the same electrolyte. Damping resulting from the microstructure of steel materials is a small fraction of the damping in the structure. In this context, differences in the microstructure between carbon steel and stainless steel alloys are insignificant. The same percent critical damping can be used in an SC wall system with carbon steel and stainless steel alloy faceplates.
3.8.5	In Subsection 3.8.5.6.4, the allowable limits for vertical displacements, differential settlements (tilting settlements) of the RXB and CRB foundations were provided. However, the allowable limits for foundation settlements between structures (e.g., RXB and CRB, RXB and RWB, etc.) were not provided. It is also not clear whether any comparative analytical assessments of compounded settlements between structures were performed to ensure the structural integrity of tunnels between buildings. Furthermore, the limitations and assessments for angular-distortions in foundations of buildings may also be provided and considered (justification(s) may be provided for not determining the angular-distortions in building foundations).	Section 3.8.5 contains more information than was available for the Readiness Assessment. Section 3.8.5.9 addresses settlement approach and results. Table 2.0-2 shows maximum allowable settlement for the Reactor Building, Control Building, and Radioactive Waste Building, including maximum allowable tilt, maximum allowable total settlement, and maximum allowable differential settlement.
3.8.5	In Subsection 3.8.5.6.5, “Thermal Loads,” no information was provided.	Section 3.8.5 contains more information than was available for the Readiness Assessment. The Thermal Loads section has been updated with additional details provided in Appendix 3B.
3.9.2	Provide design of steam generator tube supports and demonstrate how this design will avoid damage to the tubes. Also provide the final design of the steam generator tube inlet flow restrictors, and any testing plans to ensure leakage flow mechanisms will not damage the restrictors in both normal and reverse flow conditions.	The steam generator tube supports are described in SDAA Section 5.4.1.2 (System Design). The steam generator inlet flow restrictor design is also included in Section 5.4.1.2 and discussed in the Comprehensive Vibration Assessment Program Analysis (CVAP) Measurement and Inspection Plan Technical Report (TR-121354).
3.9.6	Section 3.9.6 and Table 3.9-18 should be reviewed to update the discussion and relief/alternative requests as necessary to reflect that the ASME Operation and Maintenance (OM) Code of record for the NuScale SDA is the 2017 Edition of the ASME OM Code as incorporated by	Alternate authorizations for active hydraulic-operated valves (HOVs) and emergency core cooling system (ECCS) valves have been added to FSAR Section 3.9.6.4 (Relief Requests and Alternative Authorizations to the Code). Note 16 in Table

Section	Observation	Response
	reference in 10 CFR 50.55a. The ASME OM Code of record for the NuScale DCA was the 2012 Edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a. Therefore, the relief/alternative requests in Section 3.9.6 for the NuScale SDA might need to be revised or supplemented to address the implementation of the 2017 Edition of the ASME OM Code as incorporated by reference in Title 10 of the Code of Federal Regulations (10 CFR) 50.55a. The justification for the alternative request for inadvertent actuation block (IAB) valve testing in Section 3.9.6 should be reviewed considering the revised emergency core cooling system (ECCS) design with IAB valves not included in the reactor vent valve (RVV) system. The testing provisions specified in Table 3.9-18 should be reviewed for any necessary changes to reflect the requirements in the 2017 Edition of the ASME OM Code as incorporated by reference in 10 CFR 50.55a. The notes at the end of Table 3.9-18 should be updated to be consistent with the ECCS design changes, and to reflect that the implementation of 2017 Edition of the ASME OM Code will be a regulatory requirement for the NuScale US460 SDA.	3.9-17, “Valve Inservice Test Requirements per ASME OM Code,” has been made applicable to the testing of air-operated valves (AOVs), HOVs, and ECCS valves. These alternate authorizations and the Table 3.9-17 note were discussed with the NRC at the November 17, 2022, SDAA Pre-Application Presentation entitled, “SDAA: NuScale ECCS Valve Design and OM Code Testing Application.”