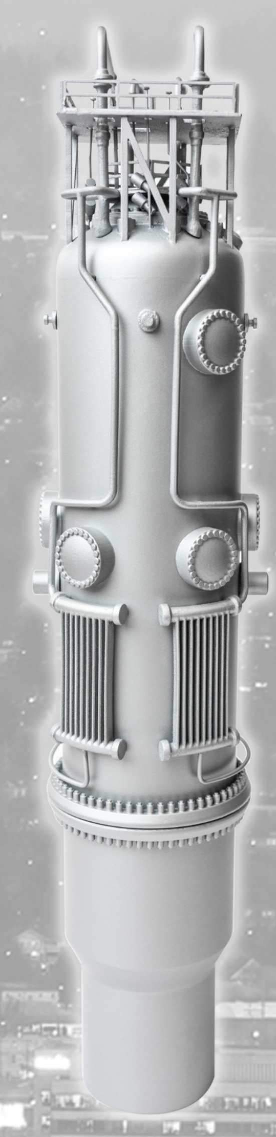
NuScale Standard Plant
Design Certification Application

Certified Design Descriptions and Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC)

PART 2 - TIER 1



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CHAPTER 1 INTRODUCTION

1.0 Introduction

This document presents the Tier 1 information developed for the NuScale, LLC Power Plant. The Tier 1 information is the information that is to be certified through rulemaking and includes the Inspections, Tests, Analyses, and Acceptance Criteria required by 10 CFR 52.47(b)(1).

Tier 1 includes the following information:

- definitions
- general provisions
- design descriptions
- Inspections, Tests, Analyses, and Acceptance Criteria
- site parameters
- interface requirements

The information presented in Tier 1 is consistent with the information presented in Tier 2.

A graded approach is employed relative to the level of design information presented in Tier 1, i.e., the amount of design information presented is proportional to the safety significance of the structures, systems, and components being addressed.

1.1 Definitions

The definitions below apply to terms that may be used in the design descriptions and associated Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC).

Acceptance Criteria refers to the performance, physical condition, or analysis result for structures, systems, and components (SSC), or program that demonstrates that the design commitment is met.

Analysis means a calculation, mathematical computation, or engineering or technical evaluation. Engineering or technical evaluations could include, but are not limited to, comparisons with operating experience or design of similar SSC.

Approved design means the design as described in the updated final safety analysis report (UFSAR), including any changes to the final safety analysis report (FSAR) since submission to the NRC of the last update of the FSAR.

As-built means the physical properties of an SSC following the completion of its installation or construction activities at its final location at the plant site. In cases where it is technically justifiable, determination of physical properties of the as-built SSC may be based on measurements, inspections, or tests that occur prior to installation, provided that subsequent fabrication, handling, installation, and testing do not alter the properties.

ASME Code means Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, as incorporated by reference in 10 CFR 50.55a with specific conditions or in accordance with relief granted or alternatives authorized by the NRC pursuant to 10 CFR 50.55a, unless a different section of the ASME Code is specifically referenced.

ASME Code Data Report means a document that certifies that a component or system is constructed in accordance with the requirements of the ASME Code. This data is recorded on a form approved by the ASME.

Common or Shared ITAAC means ITAAC that are associated with common or shared SSC and activities that support multiple NPMs. This includes (1) SSC that are common or shared by multiple NPMs, and for which the interface and functional performance requirements between the common or shared SSC and each NPM are identical, or (2) analyses or other generic design and qualification activities that are identical for each NPM (e.g., environmental qualification of equipment). For a multi-module plant, satisfactory completion of a common or shared ITAAC for the lead NPM shall constitute satisfactory completion of the common or shared ITAAC for associated NPMs.

Component, as used for reference to ASME Code components, means a vessel, concrete containment, pump, pressure relief valve, line valve, storage tank, piping system, or core support structure that is designed, constructed, and stamped in accordance with the rules of the ASME Code. ASME Code Section III classifies a metal containment as a vessel.

Design Commitment means that portion of the design description that is verified by ITAAC.

Design Description means that portion of the design that is certified. Design descriptions consist of a system description, system description tables, system description figures, and design commitments. System description tables and system description figures are only used when appropriate. The system description is not verified by ITAAC; only the design commitments are verified by ITAAC. System description tables and system description figures are only verified by ITAAC if they are referenced in the ITAAC table.

Inspect or **Inspection** means visual observations, physical examinations, or reviews of records based on visual observation or physical examination that compare (a) the SSC condition to one or more design commitments or (b) the program implementation elements to one or more program commitments, as applicable. Examples include walkdowns, configuration checks, measurements of dimensions, or nondestructive examinations. The terms, inspect and inspection, also apply to the review of Emergency Planning ITAAC requirements to determine whether ITAAC are met.

ITAAC are those Inspections, Tests, Analyses, and Acceptance Criteria identified in the combined license that if met by the licensee are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Atomic Energy Act, as amended, and the Commission's rules and regulations.

Module-Specific ITAAC means ITAAC that are associated with SSC that are specific to and support operation of a single, individual NuScale Power Module. Module-specific ITAAC shall be satisfactorily completed for each NuScale Power Module.

NuScale Power Module (NPM) is a collection of systems, sub-systems, and components that together constitute a modularized, movable, nuclear steam supply system. The NPM is composed of a reactor core, a pressurizer, and two steam generators integrated within a reactor pressure vessel and housed in a compact steel containment vessel.

Reconciliation or **Reconciled** means the identification, assessment, and disposition of differences between a design feature as described in the Updated Final Safety Analysis Report and an as-built plant design feature. For ASME Code piping systems, it is the reconciliation of differences between the design as described in the UFSAR and the as-built piping system. For structural features, it is the reconciliation of differences between the design as described in the UFSAR and the as-built structural feature.

Report, as used in the ITAAC table Acceptance Criteria column, means a document that verifies that the acceptance criteria of the subject ITAAC have been met and references the supporting documentation. The report may be a simple form that consolidates all of the necessary information related to the closure package for supporting successful completion of the ITAAC.

Safe Shutdown Earthquake (SSE) Ground Motion is the site-specific 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 ground motion response spectra. The SSE is characterized at the free ground surface. A combined license (COL) applicant may use the SSE for design of site-specific SSC.

System Description (Tier 1) includes

- a concise description of the system's or structure's safety-related functions, nonsafety-related functions that support safety-related functions, and certain nonsafety risk-significant functions.
- a listing of components required to perform those functions.
- identification of the system safety classification.
- the system components' general locations.

The system description may include system description tables and figures.

Test means actuation or operation, or establishment of specified conditions, to evaluate the performance or integrity of as-built SSC, unless explicitly stated otherwise, to determine whether ITAAC are met.

TF-3 is the test facility designed to study fluid elastic instability, vortex shedding, and turbulence due to primary side flow in helical steam generator tubes. Testing consists of modal testing in air and in water, and primary side flow testing with extensive instrumentation to detect vibration.

Tier 1 means the portion of the design-related information contained in the generic Design Control Document that is approved and certified by the design certification rule (Tier 1). The design descriptions, interface requirements, and site parameters are derived from Tier 2 information. Tier 1 includes:

- definitions and general provisions
- design descriptions
- ITAAC
- significant site parameters
- significant interface requirements

Top-Level Design Features means the principal performance characteristics and physical attributes that are important to performing the safety-related and certain nonsafety-related functions of the plant.

Type Test means a test on one or more sample components of the same type and manufacturer to qualify other components of the same type and manufacturer. A type test is not necessarily a test of an as-built SSC.

1.2 General Provisions

1.2.1 Design Descriptions

Design descriptions pertain only to the structures, systems, and components (SSC) of the standard design and not to their operation and maintenance after fuel load. In the event of an inconsistency between the design descriptions and the Tier 2 information, the design descriptions in Tier 1 shall govern.

Design descriptions consist of system descriptions, system description tables, system description figures, and design commitments. System description tables and system description figures are only used when appropriate. The system description provides a concise description of the top-level design features and performance characteristics of the SSC system functions, safety classification, and general location. The system description only describes those portions of the system or structure that are important to the top-level design features and performance characteristics of the system or structure. Design commitments are provided in numbered paragraphs that are used to develop the Design Commitment column in the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) table. These commitments address top-level design features and performance characteristics such as:

- seismic classification
- American Society of Mechanical Engineers Code classification
- Class 1E SSC
- equipment to be qualified for harsh environments
- instrumentation and controls equipment to be qualified for other than harsh environments

The absence of discussion or depiction of SSC in the design description shall not be construed as prohibiting a licensee from using such SSC, unless it would prevent SSC from performing a top-level design feature or performance characteristic, or impairing the performance of those functions, as discussed or depicted in the design description.

When the term “operate,” “operates,” or “operation” is used with respect to equipment discussed in the acceptance criteria, it refers to the actuation or control of the equipment.

1.2.2 Interpretation of System Description Tables

Cells with no values in system description tables contain an “N/A” to denote that the cell is “not applicable.”

1.2.3 Interpretation of System Description Figures

Figures are provided for some systems or structures with the amount of information depicted based on their safety significance. These figures may represent a functional diagram, general structural representation, or other general illustration. Unless specified, these figures are not indicative of the scale, location, dimensions, shape, or spatial relationships of as-built SSC. In particular, the as-built attributes of SSC may vary from the attributes depicted on these figures, provided that the top-level design features discussed in the design description pertaining to the figure are not adversely affected. Valve position indications shown on system description figures do not represent a specific operational state.

The figure legends in Tier 2 Section 1.7 are used to interpret Tier 1 system description figures.

1.2.4 Implementation of Inspections, Tests, Analyses, and Acceptance Criteria

Design commitments, inspections, tests, analyses, and acceptance criteria are provided in ITAAC tables with the following format:

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria

Each commitment in the "Design Commitment" column of the ITAAC tables has one or more associated requirements for inspections, tests or analyses specified in the "Inspections, Tests, Analyses" column. Each inspection, test, or analysis has an associated acceptance criterion in the third column of the ITAAC tables that demonstrate that the Design Commitment in the first column has been met.

Inspections, tests, or analyses may be performed by the licensee or by its authorized vendors, contractors, or consultants.

Inspections, tests, or analyses may be

- performed by more than a single individual or group.
- implemented through discrete activities separated by time.
- performed at any time prior to fuel load, including before issuance of the combined license for those ITAAC that do not require as-built equipment.
- performed at a location other than the construction site.

Additionally, inspections, tests, or analyses may be performed as part of other activities such as construction inspections and preoperational testing. Therefore, inspections, tests, or analyses need not be performed as a separate or discrete activity.

If an acceptance criteria does not specify the temperature, pressure, or other conditions under which an inspection or test must be performed, then the inspection or test conditions are not constrained.

Many of the Acceptance Criteria state that a report or analysis "exists and concludes that..." When these words are used, it indicates that the ITAAC for that Design Commitment will be met when it is confirmed that appropriate documentation exists and the documentation shows that the Design Commitment is met.

For the acceptance criteria, appropriate documentation may be a single document or a collection of documents that show that the stated acceptance criteria are met. Examples of appropriate documentation include:

- design reports
- test reports
- inspection reports
- analysis reports
- evaluation reports
- design and manufacturing procedures
- certified data sheets
- commercial grade dedication procedures and records
- quality assurance records
- calculation notes
- equipment qualification data packages

Conversion or extrapolation of test results from the test conditions to the design conditions may be necessary to satisfy an ITAAC. Suitable justification should be provided for any conversions or extrapolations of test results necessary to satisfy an ITAAC.

1.2.5 Acronyms and Abbreviations

The acronyms and abbreviations contained in Tier 2 Table 1.1-1 are applicable to Tier 1.

CHAPTER 2 UNIT SPECIFIC STRUCTURES, SYSTEMS, AND COMPONENTS DESIGN DESCRIPTIONS AND INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA

2.0 Unit Specific Systems, Structures, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria

This chapter of Tier 1 provides the structures, systems, and components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria for those structures, systems, and components that are specific to and support operation of a single NuScale Power Module. Unit-specific Inspections, Tests, Analyses, and Acceptance Criteria shall be satisfactorily completed for each NuScale Power Module in a multi-unit plant.

2.1 NuScale Power Module

2.1.1 Design Description

System Description

The scope of this section is the NuScale Power Module (NPM) and its associated systems. The NPM is installed in the reactor pool in the Reactor Building (RXB). Up to 12 NPMs may be installed in the Reactor Building. Figure 2.1-1 identifies the mechanical system boundaries for the mechanical systems within the NPM. A description of NPM piping systems is found in Table 2.1-1. The systems contained within the boundary of the NPM are the

- reactor coolant system (RCS), including the reactor pressure vessel (RPV), pressurizer, steam generator (SG), reactor vessel internals (RVI), and associated piping and valves. All RCS piping is located inside the containment vessel (CNV) and connects to containment piping located outside the CNV via CNV nozzles.
- control rod drive system (CRDS), including the control rod drive mechanisms (CRDM) with embedded cooling water tubes, cables, and associated cooling water piping. All CRDS piping is located inside the CNV and connects to containment piping located outside the CNV via CNV nozzles. The CRDS also includes instrumentation to provide control rod position indication information.
- containment system (CNTS), including the containment vessel (CNV) and containment isolation valves (CIVs) and associated piping. All containment piping is located outside the CNV with the exception of CNTS piping used for containment flooding and drain.
- emergency core cooling system (ECCS) valves.
- decay heat removal system (DHRS), including associated piping and valves. DHRS steam piping is located outside the CNV and connects to containment piping outside the CNV. The DHRS condensate lines connect the DHR condensers to the steam generator system (SGS) feedwater piping inside the CNV.
- All SGS piping is located inside the CNV. The SGS steam piping connects to CNTS steam piping located outside the CNV via CNV nozzles. The SGS feedwater piping connects to the DHRS condenser condensate line inside the CNV.

The NPM includes the pressure retaining structures of these systems because they are part of either the reactor coolant pressure boundary (RCPB) or the CNV pressure boundary. Therefore, the mechanical design and arrangement of the piping, CRDS, and NPM valves (emergency core cooling, reactor safety, and containment isolation) are included in this section.

The CRDM pressure housings form the pressure boundary between the environments inside the RPV and the CNV. The CRDM pressure housings consist of the latch housing, rod travel housing, and rod travel housing plug.

The ECCS consists of three reactor vent valves (RVVs), two reactor recirculation valves (RRVs), and associated actuators. The RRVs are designed with a minimum flow coefficient of 55 and a maximum flow coefficient of 100. Each RVV and diffuser, as a combined unit, are designed with a minimum flow coefficient of 375 and a maximum flow coefficient of 490.

Additionally, the RVVs are designed with a minimum terminal pressure drop ratio of 0.62 and a maximum terminal pressure drop ratio of 0.90.

Prototypes of the SG assembly will undergo TF-3 testing and meet the acceptance criteria in accordance with the Initial Test Program Steam Generator Flow-Induced Vibration Test. The results of the testing will be reviewed and approved in accordance with the NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report prior to loading fuel in the first ever NPM. This one-time testing satisfies TF-3 testing requirements for subsequent NPMs built in accordance with the approved design.

The NPM performs the following safety-related functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The RCS supports the CNTS by supplying the RCPB and a fission product boundary via the RPV and other appurtenances.
- The CRDS supports the RCS by maintaining the pressure boundary of the RPV.
- The SGS supports the RCS by supplying part of the RCPB.
- The ECCS supports the RCS by providing a portion of the RCPB for maintaining the RCPB integrity.
- The CNTS supports the RXB by providing a barrier to contain mass, energy, and fission product release from a degradation of the RCPB.
- The ECCS supports the CNTS by providing a portion of the containment boundary for maintaining containment integrity.
- The CNTS supports the DHRS by providing the required pressure boundary for DHR operation.
- The RCS supports the SGS by providing physical support for the SG tube supports and for the integral steam and feed plenums.
- The RCS supports the reactor core by the RVI providing mechanical support to orient, position, and seat the fuel assemblies.
- The RCS supports the CRDS by the RPV and the RVI supporting and aligning the control rods.
- The CNTS supports the DHRS by providing structural support for the DHRS piping.
- The CNTS supports the neutron monitoring system by providing structural support for the ex-core detectors.
- The RCS supports the ECCS by providing mechanical support for the ECCS valves.
- The RCS supports the in-core instrumentation system by providing structural support of the in-core instrumentation guide tubes.
- The CNTS supports the CRDS by providing structural support for the CRDMs.
- The CNTS supports the RCS by providing structural support for the RPV.
- The CNTS supports the ECCS by providing structural support of the trip and reset valves for the ECCS reactor vent valves (RVVs) and reactor recirculation valves (RRVs).

- The CNTS supports the RCS by closing the CIVs for pressurizer spray, chemical and volume control system (CVCS) makeup, CVCS letdown, and RPV high point degasification when actuated by module protection system (MPS) for RCS Isolation.
- The CNTS supports the RXB by providing a barrier to contain mass, energy, and fission product release by closure of the CIVs upon containment isolation signal.
- The CNTS supports the DHRS by closing CIVs for main steam and feedwater and opening DHRS actuation valves when actuated by MPS for DHRS operation.
- The ECCS supports the RCS by opening the ECCS reactor vent valves and RRVs when their respective trip valve is actuated by MPS.
- The DHRS supports the RCS by opening the DHRS actuation valves on a DHRS actuation signal.
- The CNTS supports the MPS by providing MPS actuation instrument information signals through the CNV.

The NPM performs the following nonsafety-related, risk-significant function that is verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The CNTS supports the Reactor Building crane (RBC) by providing lifting attachment points that the RBC can connect to so that the NPM can be lifted.

The NPM performs the following nonsafety-related functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The CNTS supports the SGS by providing structural support for the SGS piping.
- The CNTS supports the CRDS by providing structural support for the CRDS piping.
- The CNTS supports the RCS by providing structural support for the RCS piping.
- The CNTS supports the feedwater system (FWS) by providing structural support for the FWS piping.

<u>Design Commitments</u>

- The Nuscale Power Module American Society of Mechanical Engineers (ASME) Code Class 1, 2 and 3 piping systems listed in Table 2.1-1 and NuScale Power Module ASME Code Class 1, 2, 3, and CS components listed in Table 2.1-2 comply with ASME Code Section III requirements.
- The Nuscale Power Module ASME Code Class 1, 2, and 3 components listed in Table 2.1-2 conform to the rules of construction of ASME Code Section III.
- The Nuscale Power Module ASME Code Class CS components listed in Table 2.1-2 conform to the rules of construction of ASME Code Section III.
- Safety-related structures, systems, and components (SSC) are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.
- The Nuscale Power Module ASME Code Class 2 piping systems listed in Table 2.1-1 and interconnected equipment nozzles are evaluated for leak-before-break (LBB).
- The RPV beltline material has a Charpy upper-shelf energy of 75 ft-lb minimum.

- The CNV serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.
- Closure times for CIVs listed in Table 2.1-3 limit potential releases of radioactivity.
- The length of piping listed in Table 2.1-1 shall be minimized between the containment penetration and the associated outboard CIVs.
- The CNTS containment electrical penetration assemblies listed in Table 2.1-3 are sized to power their design loads.
- The RPV is provided with surveillance capsule holders to hold a capsule containing RPV material surveillance specimens at locations where the capsules will be exposed to a neutron flux consistent with the objectives of the RPV surveillance program.
- The remotely-operated CNTS containment isolation valves listed in Table 2.1-2 change position under design-basis temperature, differential pressure, and flow conditions.
- The ECCS valves listed in Table 2.1-2 change position under design-basis temperature, differential pressure, and flow conditions.
- The DHRS valves listed in Table 2.1-2 change position under design-basis temperature, differential pressure, and flow conditions.
- The CNTS hydraulic-operated valves listed in Table 2.1-2 fail to (or maintain) their safety-related position on loss of electrical power under design-basis temperature, differential pressure, and flow conditions.
- The ECCS RRVs and RVVs listed in Table 2.1-2 fail to (or maintain) their safety-related position on loss of electrical power to their corresponding trip valves under design-basis temperature, differential pressure, and flow conditions.
- The DHRS hydraulic-operated valves listed in Table 2.1-2 fail to (or maintain) their safety-related position on loss of electrical power under design-basis temperature, differential pressure, and flow conditions.
- The CNTS check valves listed in Table 2.1-2 change position under design-basis temperature, differential pressure, and flow conditions.
- Each CNTS containment electrical penetration assembly listed in Table 2.1-3 is rated either (i) to withstand fault and overload currents for the time required to clear the fault from its power source, or (ii) to withstand the maximum fault and overload current for its circuits without a circuit interrupting device.
- The NPM lifting fixture supports its rated load.
- The NPM lifting fixture is constructed to provide assurance that a single failure does not result in the uncontrolled movement of the lifted load.
- The ECCS valves, CIVs, and DHRS actuation valves listed in Table 2.1-2, and their associated hydraulic lines, are installed such that each valve can perform its safety function.

2.1.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.1-4 contains the inspections, tests, and analyses for the NPM.

Table 2.1-1: NuScale Power Module Piping Systems

Piping System Description	ASME Code Section III Class	High/ Moderate Energy	Evaluated for LBB	Length of Containment Piping (ft)
Outside CNV				
CNTS reactor coolant system injection line valves at CNV nozzle	N/A (see Note 2)	High	No	0 (see Note 1)
CNTS reactor coolant system pressurizer spray line valves at CNV nozzle	N/A (see Note 2)	High	No	0 (see Note 1)
CNTS reactor coolant system discharge line from valves at CNV nozzle to NPM disconnect flange	3	High	No	0 (see Note 1)
CNTS reactor coolant system RPV high point degasification line valves at CNV nozzle to NPM disconnect flange	3	High	No	0 (see Note 1)
CNTS containment evacuation line valves at CNV nozzle	N/A (see Note 2)	No	No	0 (see Note 1)
CNTS flood and drain line valves at CNV nozzle	N/A (see Note 2)	No	No	0 (see Note 1)
CNTS control rod drive mechanism cooling water supply line valves at CNV nozzle	N/A (see Note 2)	No	No	0 (see Note 1)
CNTS control rod drive mechanism cooling water return line valves at CNV nozzle	N/A (see Note 2)	No	No	0 (see Note 1)
CNTS steam generator #1 feedwater line valves at CNV nozzle	N/A (see Note 2)	High	No	0 (see Note 1)
CNTS steam generator #2 feedwater line valves at CNV nozzle	N/A (see Note 2)	High	No	0 (see Note 1)
CNTS steam generator #1 steam line from CNV nozzle to and including CNTS main steam CIV and CNTS main steam bypass valve CIV	2	High	No	4
CNTS steam generator #2 steam line from CNV nozzle to and including CNTS main steam CIV and CNTS main steam bypass valve CIV	2	High	No	4
DHRS #1 lines from steam generator #1 steam line to DHRS Passive Condenser Train 1 including DHRS actuation valves	2	High	No	N/A
DHRS #1 condensate line from DHRS Passive Condenser Train 1 to CNV nozzle	2	High	No	N/A
DHRS #2 lines from steam generator #2 steam line to DHRS Passive Condenser Train 2 including DHR actuation valves	2	High	No	N/A
DHRS #2 condensate line from DHRS Passive Condenser Train 2 to CNV nozzle	2	High	No	N/A

Table 2.1-1: NuScale Power Module Piping Systems (Continued)

Piping System Description	ASME Code Section III Class	High/ Moderate Energy	Evaluated for LBB	Length of Containment Piping (ft)
Inside CNV				
RCS injection line from RPV nozzle to CNV nozzle	1	High	No	N/A
RCS pressurizer spray line from RPV nozzles to CNV nozzle	1	High	No	N/A
RCS discharge line from RPV nozzle to CNV nozzle	1	High	No	N/A
RCS RPV high point degasification line from RPV nozzle to CNV nozzle	1	High	No	N/A
CNTS flood and drain line from CNV nozzle to open pipe end at bottom of CNV	2	No	No	N/A
CRDS control rod drive mechanism cooling water supply line from CNV nozzle to CRDM heat exchangers	2	No	No	N/A
CRDS control rod drive mechanism cooling water return line from CRDM heat exchangers to CNV nozzle	2	No	No	N/A
SGS steam generator #1 feedwater line from RPV nozzles to CNV nozzle	2	High	Yes	N/A
SGS steam generator #2 feedwater line from RPV nozzles to CNV nozzle	2	High	Yes	N/A
SGS steam generator #1 steam line from RPV nozzles to CNV nozzle	2	High	Yes	N/A
SGS steam generator #2 steam line from RPV nozzles to CNV nozzle	2	High	Yes	N/A
DHRS #1 condensate line from CNV nozzle to SG #1 feedwater line	2	High	No	N/A
DHRS #2 condensate line from CNV nozzle to SG #2 feedwater line	2	High	No	N/A

Note:

1) The listed component is welded directly to the safe end which is part of the containment vessel.
2) There is no ASME Class 1, 2, or 3 piping between the listed CNTS valves and the associated CNTS removable spool piece flange. The piping between the valves and the CNTS flange is classified as ASME B31.1.

Table 2.1-2: NuScale Power Module Mechanical Equipment

Equipment Name	ASME Code Section III Class	Valve Actuator Type	Containment Isolation Valve
RCS integral RPV/SG/Pressurizer	1	N/A	N/A
RVI upper core plate	CS	N/A	N/A
RVI core barrel	CS	N/A	N/A
RVI reflector blocks	CS	N/A	N/A
RVI lower core plate	CS	N/A	N/A
RVI core support blocks	CS	N/A	N/A
CNTS containment vessel	1	N/A	N/A
RCS reactor safety valves (2 Total)	1	N/A	No
CNTS pressurizer spray check valve	3	N/A	No
CNTS injection check valve	3	N/A	No
CNTS discharge excess flow check valve	3	N/A	No
ECCS reactor vent valves (3 Total)	1	Hydraulic	No
ECCS reactor vent valve trip valves (4 Total)	1	Solenoid	No
ECCS reactor vent valve reset valves (3 Total)	1	Solenoid	No
ECCS reactor recirculation valves (2 Total)	1	Hydraulic	No
ECCS reactor recirculation valve trip valves (2 Total)	1	Solenoid	No
ECCS reactor recirculation valve reset valves (2 Total)	1	Solenoid	No
CNTS solenoid valves	1	Solenoid	No
CNTS reactor coolant system injection inboard CIV	1	Electro-hydraulic	Yes
CNTS reactor coolant system injection outboard CIV	1	Electro-hydraulic	Yes
CNTS pressurizer spray inboard CIV	1	Electro-hydraulic	Yes
CNTS pressurizer spray outboard CIV	1	Electro-hydraulic	Yes
CNTS reactor coolant system discharge inboard CIV	1	Electro-hydraulic	Yes
CNTS reactor coolant system discharge outboard CIV	1	Electro-hydraulic	Yes
CNTS reactor pressure vessel high point degasification inboard CIV	1	Electro-hydraulic	Yes
CNTS reactor pressure vessel high point degasification outboard CIV	1	Electro-hydraulic	Yes
CNTS containment evacuation inboard CIV	1	Electro-hydraulic	Yes
CNTS containment evacuation outboard CIV	1	Electro-hydraulic	Yes
CNTS flood and drain inboard CIV	1	Electro-hydraulic	Yes
CNTS flood and drain outboard CIV	1	Electro-hydraulic	Yes
CNTS reactor component cooling water system supply inboard CIV	1	Electro-hydraulic	Yes
CNTS reactor component cooling water system supply outboard CIV	1	Electro-hydraulic	Yes
CNTS reactor component cooling water system return inboard CIV	1	Electro-hydraulic	Yes
CNTS reactor component cooling water system return outboard CIV	1	Electro-hydraulic	Yes
Control rod drive system thermal relief valve	2	N/A	No
CNTS feedwater #1 CIV	2	Electro-hydraulic	Yes
CNTS feedwater line #1 check valve	2	N/A	No
Steam generator #1 relief valve	2	N/A	Yes
CNTS feedwater #2 CIV	2	Electro-hydraulic	Yes
CNTS feedwater line #2 check valve	2	N/A	No

Table 2.1-2: NuScale Power Module Mechanical Equipment (Continued)

Equipment Name	ASME Code Section III Class	Valve Actuator Type	Containment Isolation Valve
Steam generator #2 relief valve	2	N/A	Yes
CNTS main steam #1 CIV	2	Electro-hydraulic	Yes
CNTS main steam line #1 bypass valve CIV	2	Electro-hydraulic	Yes
CNTS main steam #2 CIV	2	Electro-hydraulic	Yes
CNTS main steam line #2 bypass valve CIV	2	Electro-hydraulic	Yes
DHRS actuation valves (4 Total)	2	Electro-hydraulic	No
DHRS passive condensers (2 Total)	2	N/A	N/A
CRDM heat exchangers (16 Total)	2	N/A	N/A
CRDM cooling water supply flex hoses (16 Total)	2	N/A	N/A
CRDM cooling water return flex hoses (16 Total)	2	N/A	N/A
CRDM latch housing	1	N/A	N/A
CRDM rod travel housing	1	N/A	N/A
CRDM rod travel housing plug	1	N/A	N/A
CNTS I&C Division I Electrical Penetration Assembly (EPA)	1	N/A	N/A
CNTS I&C Division II Electrical Penetration Assembly (EPA)	1	N/A	N/A
CNTS PZR Heater Power #1 Electrical Penetration Assembly (EPA)	1	N/A	N/A
CNTS PZR Heater Power #2 Electrical Penetration Assembly (EPA)	1	N/A	N/A
CNTS I&C Channel A Electrical Penetration Assembly (EPA)	1	N/A	N/A
CNTS I&C Channel B Electrical Penetration Assembly (EPA)	1	N/A	N/A
CNTS I&C Channel C Electrical Penetration Assembly (EPA)	1	N/A	N/A
CNTS I&C Channel D Electrical Penetration Assembly (EPA)	1	N/A	N/A
CNTS CRD Power Electrical Penetration Assembly (EPA)	1	N/A	N/A
CNTS RPI Group #1 Electrical Penetration Assembly (EPA)	1	N/A	N/A
CNTS RPI Group #2 Electrical Penetration Assembly (EPA)	1	N/A	N/A
RPV Instrument Seal Assemblies (4 Total)	1	N/A	N/A

Table 2.1-3: NuScale Power Module Electrical Equipment

Equipment Name	Remotely Operated	Loss of Motive Power Position	CIV Closure Time (sec)[1]
ECCS reactor vent valve trip valves (4 Total)	Yes	Open	N/A
ECCS reactor vent valve reset valves (3 Total)	Yes	Close	N/A
ECCS reactor recirculation valve trip valves (2 Total)	Yes	Open	N/A
ECCSreactor recirculation valve reset valves (2 Total)	Yes	Close	N/A
CNTS reactor coolant system injection inboard CIV	Yes	Closed	≤ 7
CNTS reactor coolant system injection outboard CIV	Yes	Closed	≤ 7
CNTS pressurizer spray inboard CIV	Yes	Closed	≤ 7
CNTS pressurizer spray outboard CIV	Yes	Closed	≤ 7
CNTS reactor coolant system discharge inboard CIV	Yes	Closed	≤ 7
CNTS reactor coolant system discharge outboard CIV	Yes	Closed	≤ 7
CNTS reactor pressure vessel high point degasification inboard CIV	Yes	Closed	≤ 7
CNTS reactor pressure vessel high point degasification outboard CIV	Yes	Closed	≤ 7
CNTS containment evacuation inboard CIV	Yes	Closed	≤ 7
CNTS containment evacuation outboard CIV	Yes	Closed	≤ 7
CNTS flood and drain inboard CIV	Yes	Closed	≤ 7
CNTS flood and drain outboard CIV	Yes	Closed	≤ 7
CNTS reactor component cooling water system supply inboard CIV	Yes	Closed	≤ 7
CNTS reactor component cooling water system supply outboard CIV	Yes	Closed	≤ 7
CNTS reactor component cooling water system return inboard CIV	Yes	Closed	≤ 7
CNTS reactor component cooling water system return outboard CIV	Yes	Closed	≤ 7
CNTS feedwater #1 CIV	Yes	Closed	≤ 7
CNTS feedwater #2 CIV	Yes	Closed	≤ 7
CNTS main steam #1 CIV	Yes	Closed	≤ 7
CNTS main steam line #1 bypass valve CIV	Yes	Closed	≤7
CNTS main steam #2 CIV	Yes	Closed	7
CNTS main steam line #2 bypass valve CIV	Yes	Closed	≤7
DHRS actuation valves (4 Total)	Yes	Open	N/A
CNTS I&C Division I Electrical Penetration Assembly (EPA)	N/A	N/A	N/A
CNTS I&C Division II Electrical Penetration Assembly (EPA)	N/A	N/A	N/A
CNTS PZR Heater Power #1 Electrical Penetration Assembly (EPA)	N/A	N/A	N/A
CNTS PZR Heater Power #2 Electrical Penetration Assembly (EPA)	N/A	N/A	N/A
CNTS I&C Channel A Electrical Penetration Assembly (EPA)	N/A	N/A	N/A
CNTS I&C Channel B Electrical Penetration Assembly (EPA)	N/A	N/A	N/A
CNTS I&C Channel C Electrical Penetration Assembly (EPA)	N/A	N/A	N/A
CNTS I&C Channel D Electrical Penetration Assembly (EPA)	N/A	N/A	N/A
CNTS CRD Power Electrical Penetration Assembly (EPA)	N/A	N/A	N/A
CNTS RPI Group #1 Electrical Penetration Assembly (EPA)	N/A	N/A	N/A
CNTS RPI Group #2 Electrical Penetration Assembly (EPA)	N/A	N/A	N/A

Table 2.1-4: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The NuScale Power Module ASME Code Class 1, 2 and 3 piping systems listed in Table 2.1-1 and NuScale Power Module ASME Code Class 1, 2, 3, and CS components listed in Table 2.1-2 comply with ASME Code Section III requirements.	i. An inspection will be performed of the NuScale Power Module ASME Code Class 1, 2 and 3 as-built piping system Design Reports for systems listed in Table 2.1-1 required by ASME Code Section III.	i. The ASME Code Section III Design Reports (NCA-3550) exist and conclude that the NuScale Power Module ASME Code Class 1, 2 and 3 as-built piping systems listed in Table 2.1-1 meet the requirements of ASME Code Section III.
		ii. An inspection will be performed of the NuScale Power Module ASME Code Class 1, 2, and 3 as-built component Design Reports for components listed in Table 2.1-2 required by ASME Code Section III.	ii. The ASME Code Section III Design Reports (NCA-3550) exist and conclude that the NuScale Power Module ASME Code Class 1, 2, and 3 as-built components listed in Table 2.1-2 meet the requirements of ASME Code Section III.
		iii. An inspection will be performed of the NuScale Power Module ASME Code Class CS as-built component Design Reports for components listed in Table 2.1-2 required by ASME Code Section III.	iii. The ASME Code Section III Design Reports (NCA-3550) exist and conclude that the NuScale Power Module ASME Code Class CS as-built components listed in Table 2.1-2 meet the requirements of ASME Code Section III.
2.	The NuScale Power Module ASME Code Class 1, 2, and 3 components listed in Table 2.1-2 conform to the rules of construction of ASME Code Section III.	An inspection will be performed of the NuScale Power Module ASME Code Class 1, 2, and 3 as-built component Data Reports for components listed in Table 2.1-2 required by ASME Code Section III.	ASME Code Section III Data Reports for the NuScale Power Module ASME Code Class 1, 2, and 3 components listed in Table 2.1-2 and interconnecting piping exist and conclude that the requirements of ASME Code Section III are met.
3.	The NuScale Power Module ASME Code Class CS components listed in Table 2.1-2 conform to the rules of construction of ASME Code Section III.	An inspection will be performed of the NuScale Power Module ASME Code Class CS as-built component Data Reports for components listed in Table 2.1-2 required by ASME Code Section III.	ASME Code Section III Data Reports for the NuScale Power Module ASME Code Class CS components listed in Table 2.1-2 exist and conclude that the requirements of ASME Code Section III are met.
4.	Safety-related SSC are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.	An inspection and analysis will be performed of the as-built high- and moderate-energy piping systems and protective features for the safety-related SSC.	Protective features are installed in accordance with the as-built Pipe Break Hazard Analysis Report and safety-related SSC are protected against or qualified to withstand the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.
5.	The NuScale Power Module ASME Code Class 2 piping systems listed in Table 2.1-1 and interconnected equipment nozzles are evaluated for LBB.	An analysis will be performed of the ASME Code Class 2 as-built piping systems listed in Table 2.1-1 and interconnected equipment nozzles.	The as-built LBB analysis for the ASME Code Class 2 piping systems listed in Table 2.1-1 and interconnected equipment nozzles is bounded by the as-designed LBB analysis.

Table 2.1-4: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
6.	The RPV beltline material has a Charpy upper-shelf energy of 75 ft-lb minimum.	A vendor test will be performed of the Charpy V-Notch specimen of the RPV beltline material.	An ASME Code Certified Material Test Report exists and concludes that the initial RPV beltline material Charpy upper-shelf energy is 75 ft-lb minimum.
7.	The CNV serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.	A leakage test will be performed of the pressure containing or leakage-limiting boundaries, and CIVs.	The leakage rate for local leak rate tests (Type B and Type C) for pressure containing or leakage-limiting boundaries and CIVs meets the requirements of 10 CFR Part 50, Appendix J.
8.	Closure times for CIVs listed in Table 2.1-3 limit potential releases of radioactivity.	A test will be performed of the automatic CIVs listed in Table 2.1-3.	Each CIV listed in Table 2.1-3 travels from the full open to full closed position in less than or equal to the time listed in Table 2.1-3 after receipt of a containment isolation signal.
9.	The length of piping listed in Table 2.1-1 shall be minimized between the containment penetration and the associated outboard CIVs.	An inspection will be performed of the as-built piping listed in Table 2.1-1 between containment penetrations and associated outboard CIVs.	The length of piping between each containment penetration and its associated outboard CIV is less than or equal to the length identified in Table 2.1-1.
10.	The CNTS containment electrical penetration assemblies listed in Table 2.1-3 are sized to power their design loads.	i. An analysis will be performed of the CNTS as-designed containment electrical penetration assemblies listed in Table 2.1-3. ii. An inspection will be performed of CNTS as-built containment electrical penetration assemblies listed in Table 2.1-3.	i. An electrical rating report exists that defines and identifies the required design electrical rating to power the design loads of each CNTS containment electrical penetration assembly listed in Table 2.1-3. ii. The electrical rating of each CNTS containment electrical penetration assembly listed in Table 2.1-3 is greater than or equal to the required design electrical rating as specified in the electrical rating report.
11.	Not used.	Not used.	Not used.
12.	The RPV is provided with surveillance capsule holders to hold a capsule containing RPV material surveillance specimens at locations where the capsules will be exposed to a neutron flux consistent with the objectives of the RPV surveillance program.	An inspection will be performed of the as-built RPV surveillance capsule holders.	Four surveillance capsule holders are installed in the RPV beltline region at locations where the capsules will be exposed to a neutron flux consistent with the objectives of the RPV surveillance program.
13.	The remotely-operated CNTS containment isolation valves listed in Table 2.1-2 change position under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the remotely-operated CNTS containment isolation valves listed in Table 2.1-2 under preoperational temperature, differential pressure, and flow conditions.	Each remotely-operated CNTS containment isolation valve listed in Table 2.1-2 strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions.

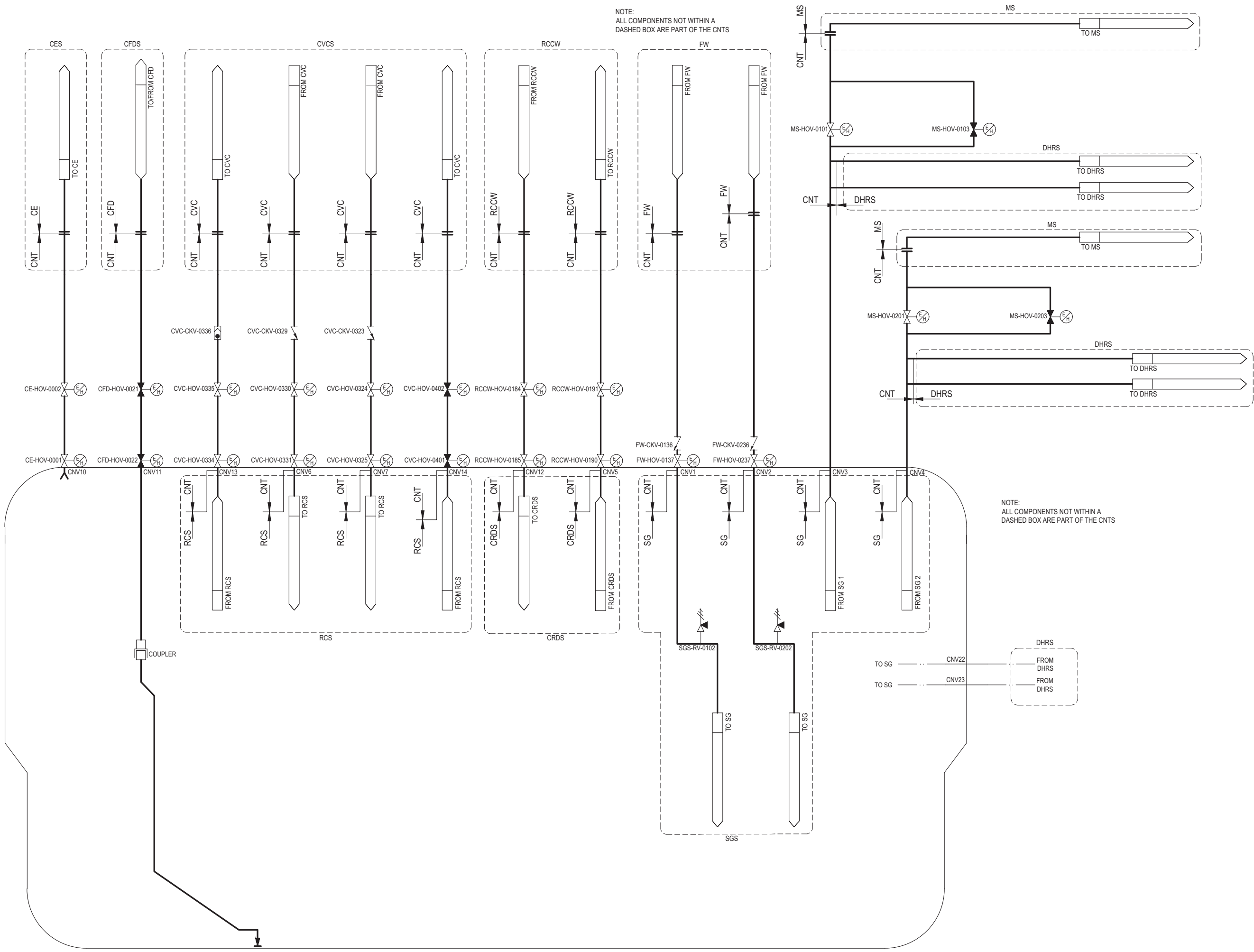
Table 2.1-4: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
14.	The ECCS valves listed in Table 2.1-2 change position under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the ECCS valves listed in Table 2.1-2 under preoperational temperature, differential pressure, and flow conditions.	Each ECCS valve listed in Table 2.1-2 strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions.
15.	The DHRS valves listed in Table 2.1-2 change position under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the DHRS valves listed in Table 2.1-2 under preoperational temperature, differential pressure, and flow conditions.	Each DHRS valve listed in Table 2.1-2 strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions.
16.	Not used.	Not used.	Not used.
17.	Not used.	Not used.	Not used.
18.	The CNTS hydraulic-operated valves listed in Table 2.1-2 fail to (or maintain) their safety-related position on loss of electrical power under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the CNTS hydraulic-operated valves listed in Table 2.1-2 under preoperational temperature, differential pressure, and flow conditions.	Each CNTS hydraulic-operated valve listed in Table 2.1-2 fails to (or maintains) its safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions.
19.	The ECCS RRVs and RVVs listed in Table 2.1-2 fail to (or maintain) their safety-related position on loss of electrical power to their corresponding trip valves under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the ECCS RRVs and RVVs listed in Table 2.1-2 under preoperational temperature, differential pressure, and flow conditions.	Each ECCS RRV and RVV listed in Table 2.1-2 fails to (or maintains) its safety-related position on loss of electrical power to its corresponding trip valve under preoperational temperature, differential pressure, and flow conditions.
20.	The DHRS hydraulic-operated valves listed in Table 2.1-2 fail to (or maintain) their safety-related position on loss of electrical power under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the DHRS hydraulic-operated valves listed in Table 2.1-2 under preoperational temperature, differential pressure, and flow conditions.	Each DHRS hydraulic-operated valve listed in Table 2.1-2 fails to (or maintains) its safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions.
21.	The CNTS check valves listed in Table 2.1-2 change position under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the CNTS check valves listed in Table 2.1-2 under preoperational temperature, differential pressure, and flow conditions.	Each CNTS check valve listed in Table 2.1-2 strokes fully open and closed (under forward and reverse flow conditions, respectively) under preoperational temperature, differential pressure, and flow conditions.

Table 2.1-4: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
22.	Each CNTS containment electrical penetration assembly listed in Table 2.1-3 is rated either (i) to withstand fault and overload currents for the time required to clear the fault from its power source, or (ii) to withstand the maximum fault and overload current for its circuits without a circuit interrupting device.	An analysis will be performed of each CNTS as-built containment electrical penetration assembly listed in Table 2.1-3.	For each CNTS containment electrical penetration assembly listed in Table 2.1-3, either (i) a circuit interrupting device coordination analysis exists and concludes that the current carrying capability for the CNTS containment electrical penetration assembly is greater than the analyzed fault and overload currents for the time required to clear the fault from its power source, or (ii) an analysis of the CNTS containment electrical penetration maximum fault and overload current exists and concludes the fault and overload current is less than the current carrying capability of the CNTS containment electrical penetration.
23.	The CNV serves as an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment.	A preservice design pressure leakage test of the CNV will be performed.	No water leakage is observed at CNV bolted flange connections.
24.	The NPM lifting fixture supports its rated load.	A rated load test will be performed of the NPM lifting fixture.	The NPM lifting fixture supports a load of at least 150 percent of the manufacturer's rated capacity.
25.	The NPM lifting fixture is constructed to provide assurance that a single failure does not result in the uncontrolled movement of the lifted load.	An inspection will be performed of the as-built NPM lifting fixture.	The NPM lifting fixture is single-failure-proof.
26.	The ECCS valves, CIVs, and DHRS actuation valves listed in Table 2.1-2, and their associated hydraulic lines, are installed such that each valve can perform its safety function.	An inspection will be performed of each ECCS valve, CIV, and DHRS actuation valve listed in Table 2.1-2, and associated hydraulic line.	A report exists and concludes each ECCS valve, CIV, and DHRS actuation valve listed in Table 2.1-2, and the associated hydraulic line, is installed in accordance with its associated installation specification.

Figure 2.1-1: Containment System (Isolation Valves)



2.2 Chemical and Volume Control System

2.2.1 Design Description

System Description

The scope of this section is the chemical and volume control system (CVCS). The system purifies reactor coolant, manages reactor coolant chemistry, provides reactor coolant inventory injection and discharge, and supplies spray flow to the pressurizer to reduce the reactor coolant system pressure. The CVCS is nonsafety-related. Each NuScale Power Module (NPM) has its own module-specific CVCS. The Reactor Building houses all CVCS equipment.

The CVCS performs the following safety-related system functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The CVCS supports the RCS by isolating dilution sources.

Design Commitments

- The chemical and volume control system American Society of Mechanical Engineers (ASME) Code Class 3 piping listed in Table 2.2-1 and chemical and volume control system ASME Code Class 3 components listed in Table 2.2-2 comply with ASME Code Section III requirements.
- The chemical and volume control system ASME Code Class 3 components listed in Table 2.2-2 conform to the rules of construction of ASME Code Section III.
- The chemical and volume control system ASME Code Class 3 air-operated demineralized water system supply isolation valves listed in Table 2.2-2 change position under design-basis temperature, differential pressure, and flow conditions.
- The chemical and volume control system ASME Code Class 3 air-operated demineralized water system supply isolation valves listed in Table 2.2-2 perform their function to fail to (or maintain) their position on loss of motive power under design-basis temperature, differential pressure, and flow conditions.

2.2.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.2-3 contains the inspections, tests, and analyses for the CVCS.

Table 2.2-1: Chemical and Volume Control System Piping

Piping System Description	ASME Code Section III Class
Demineralized water supply line between DWS Supply Isolation Valves	3
Reactor pressure vessel (RPV) discharge line from the NPM disconnect flange downstream of CVC Discharge containment isolation valve up to and including RPV discharge isolation valve and including NPM removable spool piece	3
RPV high point degasification line from the NPM disconnect flange downstream of RPV High Point Degas containment isolation valve up to and including RPV high point degasification isolation valve and NPM removable spool piece	3

Table 2.2-2: Chemical and Volume Control System Mechanical Equipment

Equipment Name	ASME Code Section III Class	Loss of Motive Power Position
Demineralized water system supply isolation valves (2 Total)	3	Closed
RPV discharge isolation valve	3	N/A
RPV high point degasification isolation valve	3	N/A

Table 2.2-3: Chemical and Volume Control System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The chemical and volume control system ASME Code Class 3 piping listed in Table 2.2-1 and chemical and volume control system ASME Code Class 3 components listed in Table 2.2-2 comply with the ASME Code Section III requirements.	i. An inspection will be performed of the chemical and volume control system ASME Code Class 3 as-built piping Design Report required by ASME Code Section III for piping listed in Table 2.2-1. ii. An inspection will be performed of the chemical and volume control system ASME Code Class 3 as-built component Design Reports required by ASME Code Section III for components listed in Table 2.2-2	i. The ASME Code Section III Design Report (NCA-3550) exists and concludes that the chemical and volume control system ASME Code Class 3 as-built piping listed in Table 2.2-1 meets the requirements of ASME Code Section III. ii. The ASME Code Section III Design Reports (NCA-3550) exist and conclude that the chemical and volume control system ASME Code Class 3 as-built components listed in Table 2.2-2 meet the requirements of ASME Code Section III.
2.	The chemical and volume control system ASME Code Class 3 components listed in Table 2.2-2 conform to the rules of construction of ASME Code Section III.	An inspection will be performed of the chemical and volume control system ASME Code Class 3 as-built component Data Reports required by ASME Code Section III for components listed in Table 2.2-2.	ASME Code Section III Data Reports for the chemical and volume control system ASME Code Class 3 components listed in Table 2.2-2 and interconnecting piping exist and conclude that the requirements of ASME Code Section III are met.
3.	The chemical and volume control system ASME Code Class 3 air-operated demineralized water system supply isolation valves listed in Table 2.2-2 change position under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the chemical and volume control system ASME Code Class 3 air-operated demineralized water system supply isolation valves listed in Table 2.2-2 under preoperational temperature, differential pressure, and flow conditions.	Each chemical and volume control system ASME Code Class 3 air-operated demineralized water system supply isolation valve listed in Table 2.2-2 strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions.
4.	Not used.	Not used.	Not used.
5.	The chemical and volume control system ASME Code Class 3 air-operated demineralized water system supply isolation valves listed in Table 2.2-2 perform their function to fail to (or maintain) their position on loss of motive power under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the chemical and volume control system ASME Code Class 3 air-operated demineralized water system supply isolation valves listed in Table 2.2-2 under preoperational temperature, differential pressure and flow conditions.	Each chemical and volume control system ASME Code Class 3 air-operated demineralized water system supply isolation valve listed in Table 2.2-2 its function to fail to (or maintain) its position performs on loss of motive power under preoperational temperature, differential pressure, and flow conditions.

2.3 Containment Evacuation System

2.3.1 Design Description

System Description

The scope of this section is the containment evacuation system (CES). Water vapor and non-condensable gases are removed from the containment vessel by the CES. The water vapor is collected and condensed into the CES sample vessel where it is monitored using level and temperature instrumentation. The CES pressure instrumentation and sample vessel level instrumentation is used to quantify and trend leak rates in the containment. The CES is a nonsafety-related system. Each NuScale Power Module (NPM) has its own module-specific CES. The Reactor Building houses all CES equipment.

The CES performs the following nonsafety-related system functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The CES supports the reactor coolant system (RCS) by providing RCS leak detection monitoring capability.

Design Commitments

- The CES sample vessel level instrumentation supports RCS leakage detection.
- The CES inlet pressure instrumentation supports RCS leakage detection.

2.3.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3-1 contains the inspections, tests, and analyses for the CES.

Table 2.3-1: Containment Evacuation System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The CES sample vessel level instrumentation supports RCS leakage detection.	A test will be performed of the CES sample vessel level instrumention.	The CES sample vessel level instrumentation detects a level increase in the CES sample vessel, which correlates to a detection of an unidentified RCS leakage rate of one gpm within one hour.
2.	The CES inlet pressure instrumentation supports RCS leakage detection.	A test will be performed of the CES inlet pressure instrumentation.	The CES inlet pressure instrumentation detects a pressure increase in CES inlet pressure, which correlates to a detection of an unidentified RCS leakage rate of one gpm within one hour.

2.4 Not Used

Table 2.4-1: Not Used

2.5 Module Protection System and Safety Display and Indication System

2.5.1 Design Description

System Description

The scope of this section is the module protection system (MPS) and its associated components in the safety display and indication system (SDIS). The primary purpose of the MPS is to monitor process variables and provide automatic initiating signals in response to out-of-normal conditions to provide protection against unsafe reactor operation during steady state and transient power operation. The MPS is a safety-related system. Each NuScale Power Module has its own independent MPS and SDIS. The Reactor Building and the Control Building house all MPS and SDIS equipment.

The MPS is comprised of the reactor trip system (RTS) and the engineered safety features actuation system (ESFAS). The RTS is responsible for monitoring key variables and shutting down the reactor when specified limits are reached. The ESFAS is responsible for monitoring key variables and actuating the engineered safety features (ESF) such as the emergency core cooling system (ECCS) and the decay heat removal system (DHRS) when specified limits are reached.

The MPS performs the following safety-related system functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The MPS supports the containment system (CNTS) by removing electrical power to the trip solenoids of the following containment isolation valves (CIVs) on a containment system isolation actuation signal:
  - reactor coolant system (RCS) injection CIVs
  - RCS discharge CIVs
  - pressurizer spray CIVs
  - reactor pressure vessel (RPV) high point degasification containment isolation valves
  - feedwater CIVs
  - main steam CIVs
  - main steam bypass valves
  - containment evacuation (CE) system CIVs
  - reactor component cooling water CIVs
  - containment flooding and drain system (CFDS) containment isolation valves
- The MPS supports the CNTS by removing electrical power to the trip solenoids of the following valves on a DHRS actuation signal:
  - DHRS actuation valves
  - main steam CIVs
  - main steam bypass valves

- feedwater CIVs

- The MPS supports the CNTS by removing electrical power to the trip solenoids of the following valves on a secondary system isolation actuation signal:
  - main steam CIVs
  - main steam bypass valves
  - feedwater CIVs
- The MPS supports the ECCS by removing electrical power to the trip solenoids of the following valves on an ECCS actuation signal:
  - reactor vent valves
  - reactor recirculation valves
- The MPS supports the CNTS by removing electrical power to the trip solenoids of the following CIVs on a chemical and volume control isolation actuation signal:
  - RCS injection CIVs
  - RCS discharge CIVs
  - Pressurizer spray CIVs
  - RPV high point degasification CIVs
- The MPS supports the chemical and volume control system (CVCS) by removing electrical power to the trip solenoids of the demineralized water system supply isolation valves on a demineralized water system isolation actuation signal.
- The MPS supports the ECCS by removing electrical power to the trip solenoids of the reactor vent valves on a low temperature overpressure protection actuation signal.
- The MPS supports the low voltage AC electrical distribution system (ELVS) by removing electrical power to the pressurizer heaters on a pressurizer heater trip actuation signal.
- The MPS supports the normal DC power system by removing electrical power to the control rod drive system for a reactor trip.
- The MPS supports the following systems by providing power to sensors for reactor trip and ESFAS actuation:
  - CNTS
  - RCS
  - DHRS (main steam system pressure sensors)

The MPS performs the following nonsafety-related system function that is verified by ITAAC.

- The MPS supports the following systems by providing power to sensors for post-accident monitoring (PAM) Type B and Type C variables:
  - CNTS
  - RCS

The primary purpose of the SDIS is to provide accurate, complete and timely information pertinent to MPS status and information displays. The SDIS provides display panels of MPS post-accident monitoring variables to support manually controlled protective actions if required.

The SDIS performs the following nonsafety-related system function that is verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The SDIS supports the main control room (MCR) by providing displays of PAM Type B and Type C variables.

Design Commitments

- The MPS design and software are implemented using a quality process composed of the following system design lifecycle phases, with each phase having outputs which satisfy the requirements of that phase:
  - system concept phase
  - system requirements phase
  - system design phase
  - system implementation phase
  - system test phase
  - system installation and checkout phase
- Protective measures are provided to restrict modifications to the MPS tunable parameters.
- Communications independence exists between Separation Groups A, B, C, and D of the Class 1E MPS.
- Communications independence exists between Divisions I and II of the Class 1E MPS.
- The MPS automatically initiates a reactor trip signal for reactor trip functions listed in Table 2.5-1.
- The MPS automatically initiates an ESF actuation signal for ESF functions listed in Table 2.5-2.
- The MPS automatically actuates a reactor trip.
- The MPS manually actuates a reactor trip.
- The reactor trip logic fails to a safe state such that loss of electrical power to a MPS separation group results in a trip state for that separation group.
- The ESFs logic fails to a safe state such that loss of electrical power to a MPS separation group results in a safe state listed in Table 2.1-3.
- The MPS interlocks listed in Table 2.5-4 automatically establish an operating bypass for the specified reactor trip or ESF actuations when the interlock condition is met, and the operating bypass is automatically removed when the interlock condition is no longer satisfied.

- The MPS permissives listed in Table 2.5-4 allow the manual bypass of the specified reactor trip or ESF actuations when the permissive condition is met, and the operating bypass is automatically removed when the permissive condition is no longer satisfied.
- The O-1 Override listed in Table 2.5-4 is established when the manual override switch is active and the RT-1 interlock is established. The Override switch must be manually taken out of Override when the O-1 Override is no longer needed.
- The MPS is capable of performing its safety-related functions when any one of its separation groups is out of service.
- The reactor trip breakers (RTBs) are installed and arranged as shown in Figure 2.5-2 in order to successfully accomplish the reactor trip function.
- Two of the four separation groups and one of the two divisions of RTS and ESFAS will utilize a different programmable technology.
- Physical separation exists (i) between each separation group of the MPS Class 1E instrumentation and control current-carrying circuits, (ii) between each division of the MPS Class 1E instrumentation and control current-carrying circuits, and (iii) between Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits.
- Electrical isolation exists (i) between each separation group of the MPS Class 1E instrumentation and control circuits, (ii) between each division of the MPS Class 1E instrumentation and control circuits, and (iii) between Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits to prevent the propagation of credible electrical faults.
- Electrical isolation exists between the highly reliable DC power system-module-specific (EDSS-MS) subsystem non-Class 1E circuits and connected MPS 1E circuits to prevent the propagation of credible electrical faults.
- Communications independence exists between the Class 1E MPS and non-Class 1E digital systems.
- The MPS automatically actuates the ESF equipment to perform its safety-related function listed in Table 2.5-2.
- The MPS manually actuates the ESF equipment to perform its safety-related function listed in Table 2.5-2.
- An MPS signal, once initiated (automatically or manually), results in an intended sequence of protective actions that continue until completion, and requires deliberate operator action in order to return the safety systems to normal.
- The MPS response times from sensor output through equipment actuation for the reactor trip functions and ESF functions are less than or equal to the value required to satisfy the design basis safety analysis response time assumptions.
- MPS operational bypasses are indicated in the MCR.
- MPS maintenance bypasses are indicated in the MCR.
- The MPS self-test features detect faults in the system and provide an alarm in the MCR.
- The PAM Type B and Type C displays are indicated on the SDIS displays in the MCR.

- The controls located on the operator workstations in the MCR operate to perform important human actions (IHAs).

2.5.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.5-7 contains the inspections, tests, and analyses for the MPS and SDIS.

Table 2.5-1: Module Protection System Automatic Reactor Trip Functions

Parameter	Input Variable	Interlock/Permissive
High source range count rate	Source range count rate	N-1 permissive
High source range log power rate	Source range log power	N-1 permissive
High intermediate range log power rate	Intermediate range log power	N-2L interlock
High-1 power range linear power	Power range linear power	N-2L permissive
High-2 power range linear power	Power range linear power	None
High power range positive rate	Power range rate (calculated from power range power)	N-2H interlock
High power range negative rate	Power range rate (calculated from power range power)	N-2H interlock
High narrow range containment pressure	Narrow range containment pressure	None
High narrow range RCS hot temperature	Narrow range RCS hot temperature (NR RCS T_{hot})	None
High pressurizer level	Pressurizer level	None
High pressurizer pressure	Pressurizer pressure	None
High main steam pressure	Main steam pressure (DHRS inlet pressure)	None
High main steam superheat	Main steam pressure (DHRS inlet pressure) Main steam temperature (DHRS inlet temperature)	None
Low AC voltage to battery chargers	ELVS voltage	None
Low low RCS flow	RCS flow	None
Low pressurizer level	Pressurizer level	None
Low pressurizer pressure	Pressurizer pressure	T-4 interlock
Low low pressurizer pressure	Pressurizer pressure	None
Low main steam pressure	Main steam pressure (DHRS inlet pressure)	N-2H interlock
Low low main steam pressure	Main steam pressure (DHRS inlet pressure)	None
Low main steam superheat	Main steam pressure (DHRS inlet pressure) Main steam temperature (DHRS inlet temperature)	V-1 interlock N-2H interlock
High under-the-bioshield temperature	Under-the-bioshield temperature	None

Table 2.5-2: Module Protection System Automatic Engineered Safety Feature Functions

Engineered Safety Feature Function	Protective	Input Variable	Interlock/Permissive
ESFAS - ECCS actuation	High containment water level	Containment water level	T-3 interlock L-2 interlock
	Low ELVS voltage 24-hour timer	ELVS voltage	None
ESFAS - DHRS actuation	High narrow range RCS hot temperature	Narrow range RCS hot temperature (NR RCS T_{hot})	None
	High pressurizer pressure	Pressurizer pressure	None
	High main steam pressure	Main steam pressure (DHRS inlet pressure)	None
	Low AC voltage to battery chargers	ELVS voltage	None
ESFAS - Secondary System Isolation	High pressurizer pressure	Pressurizer pressure	None
	High narrow range RCS hot temperature	Narrow range RCS hot temperature (NR RCS T_{hot})	None
	Low main steam pressure	Main steam pressure	N-2H interlock
	Low low main steam pressure	Main steam pressure	L-1 interlock
	High main steam pressure	Main steam pressure	None
	Low main steam superheat	Main steam pressure (DHRS inlet pressure) Main steam temperature (DHRS inlet temperature)	L-1 interlock V-1 interlock N-2H interlock
	High main steam superheat	Main steam pressure (DHRS inlet pressure) Main steam temperature (DHRS inlet temperature)	None
	High narrow range containment pressure	Narrow range containment pressure	T-3 interlock L-1 interlock
	Low low pressurizer pressure	Pressurizer pressure	T-5 interlock RT-1 interlock
	Low low pressurizer level	Pressurizer level	T-2 interlock L-1 interlock
	Low AC voltage to battery chargers	ELVS voltage	None
	High under-the-bioshield temperature	Under-the-bioshield temperature	None
ESFAS - containment system isolation	High narrow range containment pressure	Narrow range containment pressure	T-3 interlock
	Low AC voltage to battery chargers	ELVS voltage	None
	Low low pressurizer level	Pressurizer level	T-2 interlock L-1 interlock
	High under-the-bioshield temperature	Under-the-bioshield temperature	None
ESFAS - demineralized water system isolation	High subcritical multiplication	Source range count rate	N-1 interlock
	Low RCS flow	RCS flow	None
	Automatic reactor trip	N/A	T-5 interlock RT-1 interlock

Table 2.5-2: Module Protection System Automatic Engineered Safety Feature Functions (Continued)

Engineered Safety Feature Function	Protective	Input Variable	Interlock/Permissive
ESFAS - chemical and volume control system isolation	High narrow range containment pressure	Narrow range containment pressure	T-3 interlock
	High pressurizer level	Pressurizer level	None
	Low low pressurizer level	Pressurizer level	T-2 interlock L-1 interlock
	Low low pressurizer pressure	Pressurizer pressure	T-5 interlock RT-1 interlock
	Low low RCS flow	RCS flow	F-1 interlock RT-1 interlock
	Low AC voltage to battery chargers	ELVS voltage	None
	High under-the-bioshield temperature	Under-the-bioshield temperature	None
ESFAS - pressurizer heater trip	Low pressurizer level	Pressurizer level	None
	High pressurizer pressure	Pressurizer pressure	None
	High narrow range RCS hot temperature	Narrow range RCS hot temperature (NR RCS T_{hot})	None
	Low AC voltage to battery chargers	ELVS voltage	None
	High main steam pressure	Main steam pressure (DHRS inlet pressure)	None
Low temperature overpressure protection actuation	Low temperature interlock with high pressure	Wide range RCS cold temperature (WR RCS T_{cold}) Wide range RCS pressure	T-1 interlock

Table 2.5-3: Module Protection System Manual Switches

Reactor trip
Operating bypass
Emergency core cooling system actuation
Containment system isolation actuation
Decay heat removal system actuation
Secondary system isolation actuation
Chemical and volume control system isolation actuation
Demineralized water system isolation actuation
Pressurizer heater breaker trip
Low temperature overpressure protection actuation
Main control room isolation
Override
Enable nonsafety control

Table 2.5-4: Module Protection System Interlocks/Permissives/Overrides

	Interlock/Permissive/Override
F-1	RCS flow interlock
L-1	Containment water level interlock
L-2	Pressurizer level interlock
N-1	Intermediate range log power interlock/permissive
N-2H	Power range linear power interlock
N-2L	Power range linear power interlock/permissive
O-1	CNTS isolation override
RT-1	Reactor tripped interlock
T-1	Wide range RCS cold temperature interlock
T-2	Wide range RCS hot temperature interlock
T-3	Wide range RCS hot temperature interlock
T-4	Narrow range RCS hot temperature interlock
T-5	Wide range RCS hot temperature interlock
V-1	Feedwater isolation valve closed interlock

Table 2.5-5: Safety Display and Indication System Accident Monitoring Variables

Variable	Type B	Type C
Source range count rate	X	
Intermediate range log power	X	
Power range linear power	X	
Neutron monitoring system-flood	X	
Core exit temperature	X	X
Core inlet temperature	X	
Wide range RCS pressure	X	X
Degrees of subcooling (calculated from WR RCS T_{hot} and WR RCS pressure)	X	
Wide range RCS hot temperature	X	
RPV riser level	X	X
Wide range containment pressure	X	X
Containment water level	X	X
CIV positions	X	X
Inside bioshield area radiation monitor	X	X
Narrow range containment pressure	X	

Table 2.5-6: Important Human Actions Controls

Component Description	Operation
CFDS Emergency Flooding	
Division I enable nonsafety control switch	Enable
Division II enable nonsafety control switch	Enable
Division I override switch	Override
Division II override switch	Override
Containment drain inlet valve	Close
Containment drain discharge valve	Close
Containment drain separator gas discharge valve	Close
Module flood isolation valve	Open/Close
Pool suction isolation valve	Open
CFDS flood/drain selector valve	Open
CFDS pump discharge flow control valve	Open
System priming valve	Open/Close
CFDS pump A case vent valve	Open/Close
CFDS pump B case vent valve	Open/Close
Module flood outboard CIV	Open/Close
Module flood inboard CIV	Open/Close
CFDS pump A	Start/Stop
CFDS pump B	Start/Stop
CVCS Injection Following Containment Isolation	
Division I enable nonsafety control switch	Enable
Division II enable nonsafety control switch	Enable
Division I override switch	Override
Division II override switch	Override
Boric acid supply pump A	Start/Stop
Boric acid supply pump B	Start/Stop
CVCS makeup aligning valve	Open
Boric acid supply to CVCS makeup pumps	Open
CVCS three-way valve	Open
CVCS isolation valve	Open
CVCS to module heatup system isolation valve	Open
CVCS from module heatup system isolation valve	Open
CVCS isolation valve	Open
CVCS isolation valve	Open
CVCS isolation valve	Open
RCS injection CIVs (2 Total)	Open/Close
Pressurizer spray CIVs (2 Total)	Open/Close
CVCS makeup pump A	Start/Stop
CVCS makeup pump B	Start/Stop

Table 2.5-7: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	i. The MPS design and software are implemented using a quality process composed of the following system design lifecycle phases, with each phase having outputs which satisfy the requirements of that phase. i.a. System Concept Phase i.b. System Requirements Phase i.c. System Design Phase i.d. System Implementation Phase i.e. System Test Phase i.f. System Installation and Checkout Phase	i.a. An analysis will be performed of the output documentation of the System Concept Phase. i.b. An analysis will be performed of the output documentation of the System Requirements Phase. i.c. An analysis will be performed of the output documentation of the System Design Phase. i.d. An analysis will be performed of the output documentation of the System Implementation Phase. i.e. An analysis will be performed of the output documentation of the System Test Phase. i.f. An analysis will be performed of the output documentation of the System Installation and Checkout Phase.	i.a. The output documentation of the MPS Concept Phase satisfies the requirements of the System Concept Phase. i.b. The output documentation of the MPS Requirements Phase satisfies the requirements of the System Requirements Phase. i.c. The output documentation of the MPS Design Phase satisfies the requirements of the System Design Phase. i.d. The output documentation of the MPS Implementation Phase satisfies the requirements of the System Implementation Phase. i.e. The output documentation of the MPS Test Phase satisfies the requirements of the System Test Phase. i.f. The output documentation of the MPS Installation and Checkout Phase satisfies the requirements of the System Installation and Checkout Phase.
	ii. Protective measures are provided to restrict modifications to the MPS tunable parameters.	ii. Test will be performed on the access control features associated with MPS tunable parameters.	ii. Protective measures restrict modification to the MPS tunable parameters without proper configuration and authorization.
	iii.a. Communications independence exists between Separation Groups A, B, C, and D of the Class 1E MPS. iii.b. Communications independence exists between Divisions I and II of the Class 1E MPS.	iii. A test will be performed of the Class 1E MPS.	iii.a. Communications independence between Separation Groups A, B, C, and D of the Class 1E MPS is provided. iii.b. Communications independence between Division I and II of the Class 1E MPS is provided.
	iv. The MPS automatically initiates a reactor trip signal for reactor trip functions listed in Table 2.5-1.	iv. A test will be performed of the MPS.	iv. Reactor trip signal is automatically initiated for each reactor trip function listed in Table 2.5-1.
	v. The MPS automatically initiates an ESF actuation signal for ESF functions listed in Table 2.5-2.	v. A test will be performed of the MPS.	v. An ESF actuation signal is automatically initiated for each ESF function listed in Table 2.5-2.
	vi. The MPS automatically actuates a reactor trip.	vi. A test will be performed of the MPS.	vi. The RTBs open upon an injection of a single simulated MPS reactor trip signal.
	vii. The MPS manually actuates a reactor trip.	vii. A test will be performed of the MPS.	vii. The RTBs open when a reactor trip is manually initiated from the main control room.

Table 2.5-7: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	viii. The reactor trip logic fails to a safe state such that loss of electrical power to a MPS separation group results in a trip state for that separation group.	viii. A test will be performed of the MPS.	viii. Loss of electrical power in a separation group results in a trip state for that separation group.
	ix. The ESFs logic fails to a safe state such that loss of electrical power to a MPS separation group results in a safe state listed in Table 2.1-3.	ix. A test will be performed of the MPS.	ix. Loss of electrical power in a separation group results in the safe state listed in Table 2.1-3.
	x. The MPS interlocks listed in Table 2.5-4 automatically establish an operating bypass for the specified reactor trip or ESF actuations when the interlock condition is met, and the operating bypass is automatically removed when the interlock condition is no longer satisfied.	x. A test will be performed of the MPS.	x. The MPS interlocks listed in Table 2.5-4 automatically establish an operating bypass for the specified reactor trip or ESF actuations when the interlock condition is met. The operating bypass is automatically removed when the interlock condition is no longer satisfied.
	xi. The MPS permissives listed in Table 2.5-4 allow the manual bypass of the specified reactor trip or ESF actuations when the permissive condition is met, and the operating bypass is automatically removed when the permissive condition is no longer satisfied.	xi. A test will be performed of the MPS.	xi. The MPS permissives listed in Table 2.5-4 allow the manual bypass of the specified reactor trip or ESF actuations when the permissive condition is met. The operating bypass is automatically removed when the permissive condition is no longer satisfied.
	xii. The O-1 Override listed in Table 2.5-4 is established when the manual override switch is active and the RT-1 interlock is established. The Override switch must be manually taken out of Override when the O-1 Override is no longer needed.	xii. A test will be performed of the MPS.	xii. The O-1 Override listed in Table 2.5-4 is established when the manual override switch is active and the RT-1 interlock is established. The Override switch must be manually taken out of Override when the O-1 Override is no longer needed.
	xiii. The MPS is capable of performing its safety-related functions when any one of its separation groups is out of service.	xiii. A test will be performed of the MPS.	xiii. The MPS performs its safety-related functions if any one of its separation groups is out of service.
	xiv. The RTBs are installed and arranged as shown in Figure 2.5-2 in order to successfully accomplish the reactor trip function.	xiv. An inspection will be performed of the as-built RTBs, including the connections for the shunt and undervoltage trip mechanism and auxiliary contacts.	xiv. The RTBs have the proper connections for the shunt and undervoltage trip mechanisms and auxiliary contacts, and are arranged as shown in Figure 2.5-2 to successfully accomplish the reactor trip function.

Table 2.5-7: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	xv. Two of the four separation groups and one of the two divisions of RTS and ESFAS will utilize a different programmable technology.	xv. An inspection will be performed of the as-built MPS.	xv. Separation groups A & C and Division I of RTS and ESFAS utilize a different programmable technology from separation groups B & D and Division II of RTS and ESFAS.
2.	Not used.	Not used.	Not used.
3.	Physical separation exists (i) between each separation group of the MPS Class 1E instrumentation and control current-carrying circuits, (ii) between each division of the MPS Class 1E instrumentation and control current-carrying circuits, and (iii) between Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits.	An inspection will be performed of the MPS Class 1E as-built instrumentation and control current-carrying circuits.	i. Physical separation between each separation group of the MPS Class 1E instrumentation and control current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers. ii. Physical separation between each division of the MPS Class 1E instrumentation and control current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers. iii. Physical separation between MPS Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers.
4.	Electrical isolation exists (i) between each separation group of the MPS Class 1E instrumentation and control circuits, (ii) between each division of the MPS Class 1E instrumentation and control circuits, and (iii) between Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits to prevent the propagation of credible electrical faults.	An inspection will be performed of the MPS Class 1E as-built instrumentation and control circuits.	i. Class 1E electrical isolation devices are installed between each separation group of the MPS Class 1E instrumentation and control circuits. ii. Class 1E electrical isolation devices are installed between each division of the MPS Class 1E instrumentation and control circuits. iii. Class 1E electrical isolation devices are installed between MPS Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits.

Table 2.5-7: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.	Electrical isolation exists between the EDSS-MS subsystem non-Class 1E circuits and connected MPS Class 1E circuits to prevent the propagation of credible electrical faults.	i. A type test, analysis, or a combination of type test and analysis will be performed of the Class 1E isolation devices. ii. An inspection will be performed of the MPS Class 1E as-built circuits.	i. The Class 1E circuit does not degrade below defined acceptable operating levels when the non-Class 1E side of the isolation device is subjected to the maximum credible voltage, current transients, shorts, grounds, or open circuits. ii. Class 1E electrical isolation devices are installed between the EDSS-MS Subsystem non-Class 1E circuits and connected MPS Class 1E circuits.
6.	Not used.	Not used.	Not used.
7.	Communications independence exists between the Class 1E MPS and non-Class 1E digital systems.	A test will be performed of the Class 1E MPS.	Communications independence between the Class 1E MPS and non-Class 1E digital systems is provided.
8.	Not used.	Not used.	Not used.
9.	Not used.	Not used.	Not used.
10.	Not used.	Not used.	Not used.
11.	The MPS automatically actuates the ESF equipment to perform its safety-related function listed in Table 2.5-2.	A test will be performed of the MPS.	The ESF equipment automatically actuates to perform its safety-related function listed in Table 2.5-2 upon an injection of a single simulated MPS signal.
12.	Not used.	Not used.	Not used.
13.	The MPS manually actuates the ESF equipment to perform its safety-related function listed in Table 2.5-2.	A test will be performed of the MPS.	The MPS actuates the ESF equipment to perform its safety-related function listed in Table 2.5-2 when manually initiated.
14.	Not used.	Not used.	Not used.
15.	Not used.	Not used.	Not used.
16.	An MPS signal once initiated (automatically or manually), results in an intended sequence of protective actions that continue until completion, and requires deliberate operator action in order to return the safety systems to normal.	A test will be performed of the MPS reactor trip and engineered safety features signals.	i. Upon initiation of a real or simulated MPS reactor trip signal listed in Table 2.5-1, the RTBs open, and the RTBs do not automatically close when the MPS reactor trip signal clears. ii. Upon initiation of a real or simulated MPS engineered safety feature actuation signal listed in Table 2.5-2, the ESF equipment actuates to perform its safety-related function and continues to maintain its safety-related position and perform its safety-related function when the MPS engineered safety feature actuation signal clears.
17.	The MPS response times from sensor output through equipment actuation for the reactor trip functions and ESF functions are less than or equal to the value required to satisfy the design basis safety analysis response time assumptions.	A test will be performed of the MPS.	The MPS reactor trip functions listed in Table 2.5-1 and ESFs functions listed in Table 2.5-2 have response times that are less than or equal to the design basis safety analysis response time assumptions.
18.	Not used.	Not used.	Not used.

Table 2.5-7: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
19.	Not used.	Not used.	Not used.
20.	Not used.	Not used.	Not used.
21.	Not used.	Not used.	Not used.
22.	MPS operational bypasses are indicated in the MCR.	A test will be performed of the MPS.	Each operational MPS manual or automatic bypass is indicated in the MCR.
23.	MPS maintenance bypasses are indicated in the MCR.	A test will be performed of the MPS.	Each maintenance bypass is indicated in the MCR.
24.	The MPS self-test features detect faults in the system and provide an alarm in the MCR.	A test will be performed of the MPS.	A report exists and concludes that: • Self-testing features verify that faults requiring detection are detected. • Self-testing features verify that upon detection, the system responds according to the type of fault. • Self-testing features verify that faults are detected and responded within a sufficient timeframe to ensure safety function is not lost. • The presence and type of fault is indicated by the MPS alarms and displays.
25.	The PAM Type B and Type C displays are indicated on the SDIS displays in the MCR.	An inspection will be performed for the ability to retrieve the as-built PAM Type B and Type C displays on the SDIS displays in the MCR.	The PAM Type B and Type C displays listed in Table 2.5-5 are retrieved and displayed on the SDIS displays in the MCR.
26.	The controls located on the operator workstations in the MCR operate to perform IHAs.	A test will be performed of the controls on the operator workstations in the MCR.	The IHAs controls provided on the operator workstations in the MCR perform the functions listed in Table 2.5-6.
27.	Not used.	Not used.	Not used.
28.	Not used.	Not used.	Not used.

Figure 2.5-1: Module Protection System Safety Architecture Overview

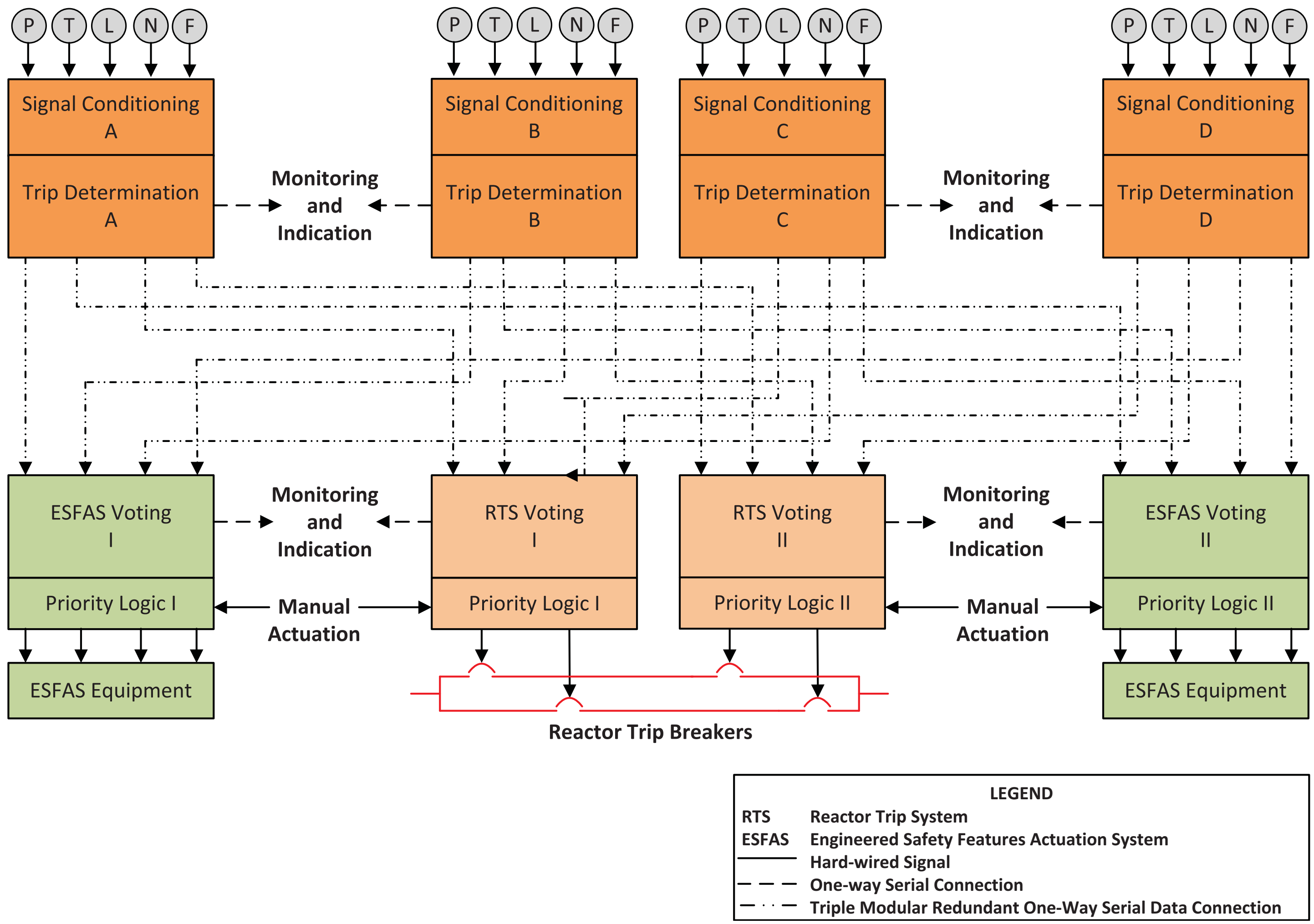
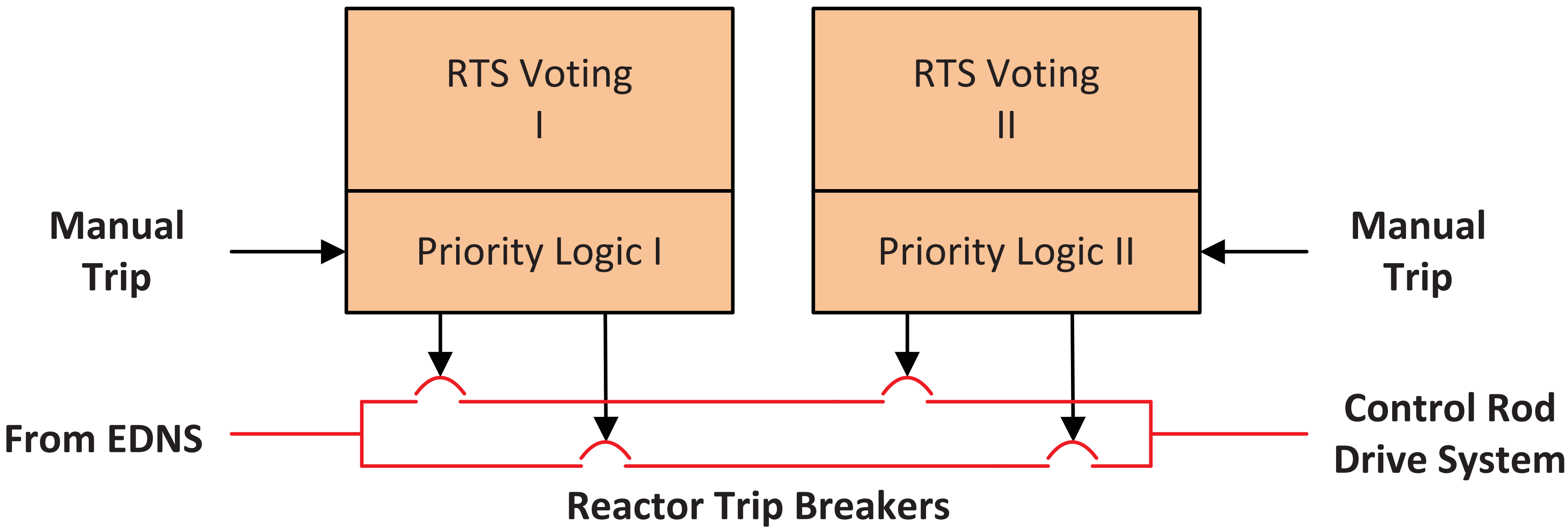


Figure 2.5-2: Reactor Trip Breaker Arrangement



2.6 Neutron Monitoring System

2.6.1 Design Description

System Description

The scope of this section is the neutron monitoring system (NMS). The NMS is a safety-related system. Each NuScale Power Module has its own module-specific NMS. The Reactor Building houses all NMS equipment.

The NMS monitors the neutron flux level of the reactor core by detecting neutron leakage from the core. The NMS measures neutron flux as an indication of core power and provides safety-related inputs to the module protection system.

The NMS performs the following safety-related system function that is verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The NMS supports the module protection system by providing neutron flux data for various reactor trips.

Design Commitments

- Electrical isolation exists between the NMS Class 1E circuits and connected non-Class 1E circuits to prevent the propagation of credible electrical faults.
- Physical separation exists between the redundant divisions of the NMS Class 1E instrumentation and control current-carrying circuits, and between Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits.
- Electrical isolation exists between the redundant divisions of the NMS Class 1E instrumentation and control circuits, and between Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits to prevent the propagation of credible electrical faults.

2.6.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6-1 contains the inspections, tests, and analyses for the NMS.

Table 2.6-1: Neutron Monitoring Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	Electrical isolation exists between the NMS Class 1E circuits and connected non-Class 1E circuits to prevent the propagation of credible electrical faults.	i. A type test, analysis, or a combination of type test and analysis will be performed of the Class 1E isolation devices. ii. An inspection will be performed of the NMS Class 1E as-built circuits.	i. The Class 1E circuit does not degrade below defined acceptable operating levels when the non-Class 1E side of the isolation device is subjected to the maximum credible voltage, current transients, shorts, grounds, or open circuits. ii. Class 1E electrical isolation devices are installed between NMS Class 1E circuits and connected non-Class 1E circuits.
2.	Physical separation exists between the redundant divisions of the NMS Class 1E instrumentation and control current-carrying circuits, and between Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits.	An inspection will be performed of the NMS Class 1E as-built instrumentation and control current-carrying circuits.	i. Physical separation between redundant divisions of NMS Class 1E instrumentation and control current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers. ii. Physical separation between NMS Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers.
3.	Electrical isolation exists between the redundant divisions of the NMS Class 1E instrumentation and control circuits, and between Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits to prevent the propagation of credible electrical faults.	An inspection will be performed of the NMS Class 1E as-built instrumentation and control circuits.	i. Class 1E electrical isolation devices are installed between redundant divisions of NMS Class 1E instrumentation and control circuits. ii. Class 1E electrical isolation devices are installed between NMS Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits.

2.7 Radiation Monitoring — Module Specific

2.7.1 Design Description

System Description

The scope of this section is automatic actions of various systems based on radiation monitoring. Automatic actions of systems based on radiation monitoring are nonsafety-related functions. The components actuated by these automatic radiation monitoring functions are contained in module-specific systems.

Design Commitments

- The containment evacuation system (CES) automatically responds to the CES high radiation signal listed in Table 2.7-1 to mitigate a release of radioactivity.
- The chemical and volume control system (CVCS) automatically responds to the CVCS and auxiliary boiler system (ABS) high radiation signals listed in Table 2.7-1 to mitigate a release of radioactivity.

2.7.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7-2 contains the inspections, tests, and analyses for the radiation monitoring - module-specific automatic actions.

Table 2.7-1: Radiation Monitoring - Module-Specific Automatic Actions

Variable Monitored	Actuated Component(s)	Component Action(s)
CES vacuum pump discharge	1. CES effluent to Reactor Building heating ventilation and air conditioning system isolation valve 2. CES effluent to gaseous waste management system isolation valve 3. CES effluent to process sample panel isolation valve 4. CES purge air solenoid valve to CES vacuum pump A 5. CES purge air solenoid valve to CES vacuum pump A 6. CES purge air solenoid valve to CES vacuum pump A 7. CES purge air solenoid valve to CES vacuum pump B 8. CES purge air solenoid valve to CES vacuum pump B 9. CES purge air solenoid valve to CES vacuum pump B	1. Close 2. Open 3. Close 4. Close 5. Close 6. Close 7. Close 8. Close 9. Close
Reactor coolant system discharge to regenerative heat exchanger	1. Reactor coolant system discharge to process sampling system isolation valve	1. Close
AB system steam flow to 0A module heatup system heat exchanger	1. CVCS module heatup system 0A & 0B heat exchanger isolation valve 2. CVCS module heatup system 0A & 0B heat exchanger isolation valve	1. Close 2. Close
Auxiliary boiler system steam flow to 0B module heatup system heat exchanger	1. CVCS module heatup system 0A & 0B heat exchanger isolation valve 2. CVCS module heatup system 0A & 0B heat exchanger isolation valve	1. Close 2. Close

Table 2.7-2: Radiation Monitoring - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The CES automatically responds to the CES high radiation signal listed in Table 2.7-1 to mitigate a release of radioactivity.	A test will be performed of the CES high radiation signal listed in Table 2.7-1.	Upon initiation of a real or simulated CES high radiation signal listed in Table 2.7-1, the CES automatically aligns/actuates the identified components to the positions identified in the table.
2.	The CVCS automatically responds to the CVCS and ABS high radiation signals listed in Table 2.7-1 to mitigate a release of radioactivity.	A test will be performed of the CVCS and ABS high radiation signals listed in Table 2.7-1.	Upon initiation of the real or simulated CVCS and ABS high radiation signals listed in Table 2.7-1, the CVCS automatically aligns/actuates the identified component(s) to the position identified in the table.

2.8 Equipment Qualification

2.8.1 Design Description

System Description

The scope of this section is equipment qualification (EQ) of equipment specific to each NuScale Power Module. Equipment qualification applies to safety-related electrical and mechanical equipment and safety-related digital instrumentation and controls equipment.

Additionally, this section applies to a limited population of module-specific, nonsafety-related equipment that has augmented Seismic Category I or environmental qualification requirements. The nonsafety-related equipment in this section has one of the following design features:

- Nonsafety-related mechanical and electrical equipment located within the boundaries of the NuScale Power Module that has an augmented Seismic Category I or environmental qualification design requirement.
- Nonsafety-related mechanical and electrical equipment that performs a credited function in Chapter 15 analyses (secondary main steam isolation valves (MSIV), feedwater regulating valves (FWRV) and secondary feedwater check valves).

Design Commitments

- The module-specific Seismic Category I equipment listed in Table 2.8-1, including its associated supports and anchorages, withstands design basis seismic loads without loss of its function(s) during and after a safe shutdown earthquake (SSE).
- The module-specific electrical equipment located in a harsh environment listed in Table 2.8-1, including associated connection assemblies, withstand the design basis harsh environmental conditions experienced during normal operations, anticipated operational occurrences (AOOs), design basis accidents (DBAs), and post-accident conditions, and performs its function for the period of time required to complete the function.
- The non-metallic parts, materials, and lubricants used in module-specific mechanical equipment listed in Table 2.8-1 perform their function up to the end of their qualified life in the design basis harsh environmental conditions (both internal service conditions and external environmental conditions) experienced during normal operations, AOOs, DBAs, and post-accident conditions.
- The Class 1E computer-based instrumentation and control systems listed in Table 2.8-1 located in a mild environment withstand design basis mild environmental conditions without loss of safety-related functions.
- The Class 1E digital equipment listed in Table 2.8-1 performs its safety-related function when subjected to the design basis electromagnetic interference, radio frequency interference, and electrical surges that would exist before, during, and following a DBA.
- The valves listed in Table 2.8-1 are functionally designed and qualified to perform their safety-related function under the full range of fluid flow, differential pressure, electrical, temperature, and fluid conditions up to and including DBA conditions.

- The safety-related relief valves listed in Table 2.8-1 provide overpressure protection.
- The DHRS condensers listed in Table 2.8-1 have the capacity to transfer their design heat load.
- The containment system (CNTS) containment electrical penetration assemblies listed in Table 2.8-1, including associated connection assemblies, withstand the design basis harsh environmental conditions experienced during normal operations, AOOs, DBAs, and post-accident conditions, and performs its function for the period of time required to complete the function.

2.8.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.8-2 contains the inspections, tests, and analyses for equipment qualification – module-specific equipment.

Table 2.8-1: Module Specific Mechanical and Electrical/I&C Equipment

Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
Containment System						
CNTS I&C Division I Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNTS I&C Division II Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNTS PZR Heater Power #1 Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNTS PZR Heater Power #2 Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNTS I&C Channel A Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	Yes	A
CNTS I&C Channel B Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	Yes	A
CNTS I&C Channel C Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	Yes	A
CNTS I&C Channel D Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	Yes	A
CNTS CRD Power Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNTS RPI Group #1 Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNTS RPI Group #2 Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
MS #1 CIV (MSIV #1)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
MS #2 CIV (MSIV #2)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
MS line #1 Bypass Valve (MSIV Bypass #1)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
MS line #2 Bypass Valve (MSIV Bypass #2)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
FW #1 CIV (FWIV #1)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B

Table 2.8-1: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[1]
FW #2 CIV (FWIV #2)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
FW line #1 Check Valve	RXB - Top of Module	Harsh	Mechanical	Yes	N/A	A B
FW line #2 Check Valve	RXB - Top of Module	Harsh	Mechanical	Yes	N/A	A B
CVC Discharge CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CVC Injection CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CVC PZR Spray CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
RPV High Point Degas CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
RCCW Supply CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
RCCW Return CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CE CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CFDS CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CNTS Check Valves and Excess Flow Check Valves (3 Total)	RXB - Top of Module	Harsh	Mechanical	Yes	N/A	A B
Hydraulic Skids (2 Total)	RXB - 100’ RXB - 120'	Harsh	Electrical Mechanical	Yes	No	A
Containment Pressure Transducers (Narrow Range) (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
Containment Pressure Transducers (Wide Range) (2 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	No	A
Containment Water Level Sensors (Radar Transceiver) (4 Total)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
SG #1 Steam Temperature Sensors (RTD) (4 Total)	RXB - Top of Module	Harsh	Electrical	Yes	Yes	A
SG #2 Steam Temperature Sensors (RTD) (4 Total)	RXB - Top of Module	Harsh	Electrical	Yes	Yes	A

Table 2.8-1: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
CE Inboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CE Inboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CE Outboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CE Outboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CFD Inboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CFD Inboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CFD Outboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CFD Outboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Inboard RCS Discharge CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Inboard RCS Discharge CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Outboard CIV RCS Discharge Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Outboard CIV RCS Discharge Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Inboard RCS Injection CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Inboard RCS Injection CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Outboard RCS Injection CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Outboard RCS Injection CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Inboard PZR Spray Line CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Inboard PZR Spray Line CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Outboard PZR Spray Line CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Outboard PZR Spray Line CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Inboard RPV High- Point Degasification CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A

Table 2.8-1: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
CVCS Inboard RPV High- Point Degasification CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Outboard RPV High-Point Degasification CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVCS Outboard RPV High-Point Degasification CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW Supply Inboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW Supply Inboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW Supply Outboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW Supply Outboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW Return Inboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW Return Inboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW Return Outboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW Return Outboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
FW Supply to SG1 and DHR HX1 CIV/FWIV Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
FW Supply to SG1 and DHR HX1 CIV/FWIV Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
FW Supply to SG2 and DHR HX2 CIV/FWIV Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
FW Supply to SG2 and DHR HX2 CIV/FWIV Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
SG1 Steam Supply CIV/ MSIV Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
SG1 Steam Supply CIV/ MSIV Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
SG1 Steam Supply CIV/ MS Bypass Isolation Valve Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
SG1 Steam Supply CIV/ MS Bypass Isolation Valve Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
SG2 Steam Supply CIV/ MSIV Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
SG2 Steam Supply CIV/ MSIV Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A

Table 2.8-1: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[(1)]
SG2 Steam Supply CIV/ MS Bypass Isolation Valve Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
SG2 Steam Supply CIV/ MS Bypass Isolation Valve Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
Steam Generator System						
SG Tubes and Tube Supports	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Steam Plenums (4 Total)	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Feedwater Plenums (4 Total)	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Flow Restrictors	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Thermal Relief Valves (2 Total)	RXB - Inside Containment	Harsh	Mechanical	Yes	N/A	B
Control Rod Drive System						
Rod Position Indication (RPI) Coils (32 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	No	B
Control Rod Drive Shafts	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Control Rod Drive Latch Mechanism	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
CRDM Pressure Boundary (Latch Housing, Rod Travel Housing, Rod Travel Housing Plug)	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Control Rod Assembly						
All components	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Neutron Source Assembly						
Primary and secondary neutron source rodlets	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Spider body, hub or coupling housing	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Reactor Coolant System						
Reactor Vessel Internals	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Reactor Safety Valve Position Indicators (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	No	B
Reactor Safety Valves (2 Total)	RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	N/A	A
Narrow Range Pressurizer Pressure Elements (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
Wide Range RCS Pressure Elements (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
PZR/RPV Level Elements (4 Total)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
Narrow Range RCS Hot Leg Temperature Elements (12 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A

Table 2.8-1: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
Wide Range RCS Hot Leg Temperature Elements (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
Wide Range RCS Cold Leg Temperature Elements (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	B
RCS Flow Transmitters (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
PZR Heaters (2 Total)	RXB - Inside Containment	N/A	N/A	Yes	No	N/A
Chemical and Volume Control System						
DWS Supply Isolation Valves (2 Total)	RXB - 50′	Harsh	Electrical Mechanical	Yes	Yes	A B
Emergency Core Cooling System						
Reactor Vent Valves (3 Total)	RXB - Inside Containment	Harsh	Mechanical	Yes	No	A
RVV Position Indications (8 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	No	A
Reactor Recirculation Valves (2 Total)	RXB - Inside Containment	Harsh	Mechanical	Yes	No	A
RRV Position Indications (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	No	A
RVV Trip Valves (4 Total)	RXB - Pool	Harsh	Electrical Mechanical	Yes	Yes	A B
RRV Trip Valves (2 Total)	RXB - Pool	Harsh	Electrical Mechanical	Yes	Yes	A B
RVV Trip Valve Position Indications (8 Total)	RXB - Pool	Harsh	Electrical	Yes	No	A
RRV Trip Valve Position Indications (4 Total)	RXB - Pool	Harsh	Electrical	Yes	No	A
RVV Reset Valves (3 Total)	RXB - Pool	Harsh	Electrical Mechanical	Yes	No	A
RRV Reset Valves (2 Total)	RXB - Pool	Harsh	Electrical Mechanical	Yes	No	A
Decay Heat Removal System						
DHRS Actuation Valves (4 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A
DHRS Condenser Outlet Temperature Transmitters (4 Total)	RXB - Pool	Harsh	Electrical	Yes	No	A
DHRS Condenser Outlet Pressure Transmitters (6 Total)	RXB - Pool	Harsh	Electrical	Yes	No	A
DHRS Valve Position Indicators (8 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
Condensers (2 Total)	RXB - Side of Module	N/A	N/A	Yes	N/A	N/A
SG Steam Pressure Transmitters (8 Total)	RXB - Top of Module	Harsh	Electrical	Yes	Yes	A

Table 2.8-1: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
Main Steam System						
Secondary Main Steam Isolation Valves (2 Total)	RXB - 100'	Harsh	Electrical Mechanical	Yes	No	A B
Secondary Main Steam Isolation Bypass Valves (2 Total)	RXB - 100'	Harsh	Electrical Mechanical	Yes	No	A B
Condensate and Feedwater System						
Feedwater Regulating Valves A/B (2 Total)	RXB - 100’	Harsh	Electrical Mechanical	Yes	No	A
Feedwater Supply Check Valves (2 Total)	RXB - 100’	Harsh	Mechanical	Yes	N/A	A
Module Protection System						
Safety-Related MPS Modules – Safety Function Modules – Hard-wired Modules – Scheduling and Bypass Modules – Equipment Interface Modules – Scheduling and Voting Modules	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
Power Isolation, Conversion and Monitoring Devices	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
ELVS Voltage Sensors	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
Under-the-Bioshield Temperature Sensors	RXB - Top of the Module	Harsh	Electrical	Yes	Yes	A
PZR Heater Trip Breakers (4 Total)	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
Reactor Trip Breakers (4 Total)	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
Safety Function Module Trip/Bypass Switches (60 Total)	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
Enable Nonsafety Control Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
MCR Isolation Switches (2 Total)	RXB - 75'	Harsh	Electrical	Yes	Yes	B
Manual PZR Heater Breaker Trip Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
Manual LTOP Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
Manual ECCS Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E

Table 2.8-1: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[1]
Manual DWSI Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
Manual DHRS Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
Manual CVCSI Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
Manual CSI Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
Manual Reactor Trip Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
Override Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
Operating Bypass Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
Neutron Monitoring System						
Excore Neutron Detectors	RXB - Pool	Harsh	Electrical	Yes	Yes	A
Excore Signal Conditioning and Processing Equipment	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
Excore Power Isolation, Conversion and Monitoring Devices	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
In-Core Instrumentation System						
In-core instrument string / temperature and flux sensors	RXB - Inside Containment	Harsh	Electrical	Yes	No	A
In-core instrument string sheath	RXB - Inside Containment	Harsh	Mechanical	Yes	N/A	B

Notes:

1. EQ Categories:

- A Equipment that will experience the environmental conditions of design basis accidents for which it must function to mitigate said accidents, and that will be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.
- B Equipment that will experience the environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that will be qualified to demonstrate the capability to withstand the accident environment for the time during which it must not fail with safety margin to failure.
- E Equipment that will not experience environmental conditions of design basis accidents and that will be qualified to demonstrate operability under the expected extremes of its nonaccident service environment.

Table 2.8-2: Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The module-specific Seismic Category I equipment listed in Table 2.8-1, including its associated supports and anchorages, withstands design basis seismic loads without loss of its function(s) during and after an SSE.	i. A type test, analysis, or a combination of type test and analysis will be performed of the module-specific Seismic Category I equipment listed in Table 2.8-1, including its associated supports and anchorages. ii. An inspection will be performed of the module-specific Seismic Category I as-built equipment listed in Table 2.8-1, including its associated supports and anchorages.	i. A Seismic Qualification Report exists and concludes that the module-specific Seismic Category I equipment listed in Table 2.8-1, including its associated supports and anchorages, will withstand the design basis seismic loads and perform its function(s) during and after an SSE. ii. The module-specific Seismic Category I equipment listed in Table 2.8-1, including its associated supports and anchorages, is installed in its design location in a Seismic Category I structure in a configuration bounded by the equipment's Seismic Qualification Report.
2.	The module-specific electrical equipment located in a harsh environment listed in Table 2.8-1, including associated connection assemblies, withstand the design basis harsh environmental conditions experienced during normal operations, AOOs, DBAs, and post-accident conditions and performs its function for the period of time required to complete the function.	i. A type test or a combination of type test and analysis will be performed of the module-specific electrical equipment listed in Table 2.8-1, including associated connection assemblies. ii. An inspection will be performed of the module-specific as-built electrical equipment listed in Table 2.8-1, including associated connection assemblies.	i. An EQ record form exists and concludes that the module-specific electrical equipment listed in Table 2.8-1, including associated connection assemblies, perform their function under the environmental conditions specified in the EQ record form for the period of time required to complete the function. ii. The module-specific electrical equipment listed in Table 2.8-1, including associated connection assemblies, are installed in their design location in a configuration bounded by the EQ record form.
3.	The non-metallic parts, materials, and lubricants used in module-specific mechanical equipment listed in Table 2.8-1 perform their function up to the end of their qualified life in the design basis harsh environmental conditions (both internal service conditions and external environmental conditions) experienced during normal operations, AOOs, DBAs, and post-accident conditions.	A type test or a combination of type test and analysis will be performed of the non-metallic parts, materials, and lubricants used in module-specific mechanical equipment listed in Table 2.8-1.	A qualification record form exists and concludes that the non-metallic parts, materials, and lubricants used in module-specific mechanical equipment listed in Table 2.8-1 perform their function up to the end of their qualified life under the design basis harsh environmental conditions (both internal service conditions and external environmental conditions) specified in the qualification record form.

Table 2.8-2: Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
4.	The Class 1E computer-based instrumentation and control systems listed in Table 2.8-1 located in a mild environment withstand design basis mild environmental conditions without loss of safety-related functions.	i. A type test or a combination of type test and analysis will be performed of the Class 1E computer-based instrumentation and control systems listed in Table 2.8-1 located in a mild environment. ii. An inspection will be performed of the Class 1E as-built computer-based instrumentation and control systems listed in Table 2.8-1 located in a mild environment.	i. An EQ record form exists and concludes that the Class 1E computer-based instrumentation and control systems listed in Table 2.8-1 located in a mild environment perform their function under the environmental conditions specified in the EQ record form. ii. The Class 1E computer-based instrumentation and control systems listed in Table 2.8-1 located in a mild environment are installed in their design location in a configuration bounded by the EQ record form.
5.	The Class 1E digital equipment listed in Table 2.8-1 performs its safety-related function when subjected to the design basis electromagnetic interference, radio frequency interference, and electrical surges that would exist before, during, and following a DBA.	A type test, analysis, or a combination of type test and analysis will be performed of the Class 1E digital equipment listed in Table 2.8-1.	An EQ record form exists and concludes that the Class 1E digital equipment listed in Table 2.8-1 withstands the design basis electromagnetic interference, radio frequency interference, and electrical surges that would exist before, during, and following a DBA without loss of safety-related function.
6.	The valves listed in Table 2.8-1 are functionally designed and qualified to perform their safety-related function under the full range of fluid flow, differential pressure, electrical, temperature, and fluid conditions up to and including DBA conditions.	A type test or a combination of type test and analysis will be performed of the valves listed in Table 2.8-1.	A Qualification Report exists and concludes that the valves listed in Table 2.8-1 are capable of performing their safety-related function under the full range of fluid flow, differential pressure, electrical, temperature, and fluid conditions up to and including DBA conditions.
7.	The safety-related relief valves listed in Table 2.8-1 provide overpressure protection.	i. A vendor test will be performed of each safety-related relief valve listed in Table 2.8-1. ii. An inspection will be performed of each safety-related as-built relief valve listed in Table 2.8-1.	i. An American Society of Mechanical Engineers Code Section III Data Report exists and concludes that the relief valves listed in Table 2.8-1 meet the valve's required set pressure, capacity, and overpressure design requirements. ii. Each relief valve listed in Table 2.8-1 is provided with an American Society of Mechanical Engineers Code Certification Mark that identifies the set pressure, capacity, and overpressure.

Table 2.8-2: Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
8.	The DHRS condensers listed in Table 2.8-1 have the capacity to transfer their design heat load.	A type test or a combination of type test and analysis will be performed of the DHRS condensers listed in Table 2.8-1.	A report exists and concludes that the DHRS condensers listed in Table 2.8-1 have a heat removal capacity sufficient to transfer their design heat load.
9.	The CNTS containment electrical penetration assemblies listed in Table 2.8-1, including associated connection assemblies, withstand the design basis harsh environmental conditions experienced during normal operations, AOOs, DBAs, and postaccident conditions and performs its function for the period of time required to complete the function.	i. A type test or a combination of type test and analysis will be performed of the CNTS containment electrical penetration assemblies listed in Table 2.8-1 including associated connection assemblies. ii. An inspection will be performed of the containment CNTS electrical penetration assembles listed in Table 2.8-1, including associated connection assemblies.	i. An EQ record form exists and concludes that the CNTS electrical penetration assemblies listed in Table 2.8-1, including associated connection assemblies, performs their function under the environmental conditions specified in the EQ record form for the period of time required to complete the function. ii. The CNTS electrical penetration assemblies listed in Table 2.8-1, including associated connection assemblies, are installed in their design location in a configuration bounded by the EQ record form.

2.9 Fuel Assembly Design

2.9.1 Fuel Assembly Design

System Description

The fuel assembly is designed to ensure that possible fuel damage will not result in the release of radioactive materials during normal operations, anticipated operational occurrences, and postulated accidents, in excess of prescribed limits. The fuel assembly is comprised of fuel rods, spacer grids, guide tubes, top and bottom nozzles, and plenum springs. The fuel assembly design is approved by the NRC for the NuScale reactor design.

2.9.2 Inspections, Tests, Analyses and Acceptance Criteria

None

CHAPTER 3 SHARED STRUCTURES, SYSTEMS, AND COMPONENTS AND NON-STRUCTURES, SYSTEMS, AND COMPONENTS DESIGN DESCRIPTIONS AND INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA

3.0 Shared Structures, Systems, and Components and Non-Structures, Systems, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria

This chapter of Tier 1 provides the structures, systems, and components (SSC) Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for those SSC that are common or shared by multiple NuScale Power Modules (NPMs). Shared systems in this chapter of Tier 1 are either shared by 1-12 NPMs or by 1- 6 NPMs as shown in Table 3.0-1. This chapter also includes non-SSC based Design Descriptions and ITAAC that are common or shared by multiple NPMs. For a multi-module plant, satisfactory completion of a shared ITAAC for the lead module shall constitute satisfactory completion of the shared ITAAC for associated modules. The ITAAC in Sections 3.1 through 3.17 shall only be completed once in conjunction with the ITAAC in Chapter 2 for the first NPM. The ITAAC in Section 3.18 shall only be completed once in conjunction with the ITAAC in Chapter 2 for NPM 7 or NPM 12, whichever is completed first.

Table 3.0-1: Shared Systems Subject to Inspections, Tests, Analyses, and Acceptance Criteria

Shared System	NPMs Supported
Balance-of-plant drain system	1 system per 6 modules
Containment flooding and drain system	1 system per 6 modules
Normal control room heating ventilation and air conditioning system	1 system per 12 modules
Control room habitability system	1 system per 12 modules
Reactor Building heating ventilation and air conditioning system	1 system per 12 modules
Fuel handling equipment system	1 system per 12 modules
Fuel storage system	1 system per 12 modules
Ultimate heat sink	1 system per 12 modules
Fire protection system	1 system per 12 modules
Plant lighting system	1 system per 12 modules
Gaseous radioactive waste system	1 system per 12 modules
Liquid radioactive waste system	1 system per 12 modules
Auxiliary boiler system	1 system per 12 modules
Pool surge control system	1 system per 12 modules
Reactor Building crane system	1 system per 12 modules
Reactor Building and Reactor Building components	1 system per 12 modules
Radioactive Waste Building	1 system per 12 modules
Control Building	1 system per 12 modules
Physical security system	1 system per 12 modules

3.1 Control Room Habitability

3.1.1 Design Description

System Description

The scope of this section is the control room habitability system (CRHS). The CRHS provides clean breathing air to the control room envelope and maintains a positive control room pressure during high radiation or loss of offsite power conditions for habitability and control of radioactivity. The CRHS is a nonsafety-related system which supports up to 12 NuScale Power Modules (NPMs). The Control Building houses all CRHS equipment.

The CRHS performs the following nonsafety-related system function that is verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The CRHS supports the Control Building by providing clean breathing air to the main control room (MCR) and maintains a positive control room pressure during high radiation or loss of normal AC power conditions.

Design Commitments

- The air exfiltration out of the control room envelope (CRE) is less than or equal to the assumptions used to size the CRHS inventory and the supply flow rate.
- The CRHS valves listed in Table 3.1-1 change position under design basis temperature, differential pressure, and flow conditions.
- The CRHS solenoid-operated valves listed in Table 3.1-1 perform their function to fail open on loss of motive power under design basis temperature, differential pressure, and flow conditions.
- The CRE heat sink passively maintains the temperature of the CRE within an acceptable range for the first 72 hours following a design basis accident (DBA).
- The CRHS maintains a positive pressure in the MCR relative to the adjacent areas.

3.1.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.1-2 contains the inspections, tests, and analyses for the CRHS.

Table 3.1-1: Control Room Habitability System Mechanical Equipment

Equipment Name	Failure Position
Air supply isolation solenoid valves (2 Total)	Open
CRE pressure relief isolation valves (2 Total)	Open

Table 3.1-2: Control Room Habitability System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The air exfiltration out of the CRE is less than or equal to the assumptions used to size the CRHS inventory and the supply flow rate.	A test will be performed of the CRE.	The air exfiltration measured by tracer gas testing is less than or equal to the CRE air infiltration rate assumed in the dose analysis.
2.	The CRHS valves listed in Table 3.1-1 change position under design basis temperature, differential pressure, and flow conditions.	A test will be performed of the CRHS valves listed in Table 3.1-1 under preoperational temperature, differential pressure, and flow conditions.	Each CRHS valve listed in Table 3.1-1 strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions.
3.	The CRHS solenoid-operated valves listed in Table 3.1-1 perform their function to fail open on loss of motive power under design basis temperature, differential pressure, and flow conditions.	A test will be performed of the CRHS solenoid-operated valves listed in Table 3.1-1 under preoperational temperature, differential pressure and flow conditions.	Each CRHS solenoid-operated valve listed in Table 3.1-1 performs its function to fail open on loss of motive power under preoperational temperature, differential pressure, and flow conditions.
4.	The CRE heat sink passively maintains the temperature of the CRE within an acceptable range for the first 72 hours following a DBA.	An analysis will be performed of the as-built CRE heat sinks.	A report exists and concludes that the CRE heat sink passively maintains the temperature of the CRE within an acceptable range for the first 72 hours following a DBA.
5.	The CRHS maintains a positive pressure in the MCR relative to adjacent areas.	A test will be performed of the CRHS.	The CRHS maintains a positive pressure of greater than or equal to 1/8 inches water gauge in the CRE relative to adjacent areas, while operating in DBA alignment.

3.2 Normal Control Room Heating Ventilation and Air Conditioning System

3.2.1 Design Description

System Description

The scope of this section is the normal control room HVAC system (CRVS). The CRVS serves the entire Control Building (CRB) and the access tunnel between the CRB and the Reactor Building (RXB). The CRVS is a nonsafety-related system. The CRVS supports up to 12 NuScale Power Modules. The CRB houses all CRVS equipment.

The CRVS performs the following nonsafety-related system functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The CRVS supports the CRB by providing isolation of the control room envelope (CRE) from the surrounding areas and outside environment via isolation dampers.
- The CRVS supports the CRB by maintaining the CRB at a positive pressure relative to the RXB and the outside atmosphere to control the ingress of potentially airborne radioactivity from the RXB or the outside atmosphere to the CRB.
- The CRVS supports the highly reliable DC power system by providing ventilation to maintain airborne hydrogen concentrations below the allowable limits.
- The CRVS supports the normal DC power system by providing ventilation to maintain airborne hydrogen concentrations below allowable limits.

Design Commitments

- The CRVS air-operated CRE isolation dampers listed in Table 3.2-1 perform their function to fail to the closed position on loss of motive power under design basis temperature, differential pressure, and flow conditions.
- The CRVS maintains a positive pressure in the CRB relative to the outside environment.
- The CRVS maintains the hydrogen concentration levels in the CRB battery rooms containing batteries below one percent by volume.

3.2.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.2-2 contains the inspections, tests, and analyses for the CRVS.

Table 3.2-1: Normal Control Room Heating Ventilation and Air Conditioning System Mechanical Equipment

Equipment Name	Actuator Type
CRE isolation dampers (8 Total)	Pneumatic

Table 3.2-2: Normal Control Room Heating Ventilation and Air Conditioning Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The CRVS air-operated CRE isolation dampers listed in Table 3.2-1 perform their function to fail to the closed position on loss of motive power under design basis temperature, differential pressure, and flow conditions.	A test will be performed of the air-operated CRE isolation dampers listed in Table 3.2-1 under preoperational temperature, differential pressure and flow conditions.	Each CRVS air-operated CRE isolation damper listed in Table 3.2-1 performs its function to fail to the closed position on loss of motive power under preoperational temperature, differential pressure, and flow conditions.
2.	The CRVS maintains a positive pressure in the CRB relative to the outside environment.	A test will be performed of the CRVS while operating in the normal operating alignment.	The CRVS maintains a positive pressure of greater than or equal to 1/8 inches water gauge in the CRB relative to the outside environment, while operating in the normal operating alignment.
3.	The CRVS maintains the hydrogen concentration levels in the CRB battery rooms containing batteries below one percent by volume.	A test will be performed of the CRVS while operating in the normal operating alignment.	The airflow capability of the CRVS maintains the hydrogen concentration levels in the CRB battery rooms containing batteries below one percent by volume.

3.3 Reactor Building Heating Ventilation and Air Conditioning System

3.3.1 Design Description

System Description

The scope of this section is the Reactor Building HVAC system (RBVS). The RBVS is designed to remove radioactive contaminants from the exhaust streams of the Reactor Building (RXB) general area, the Radioactive Waste Building (RWB) general area, and the Annex Building. The RBVS is a nonsafety-related system. The RBVS supports up to 12 NuScale Power Modules. The RXB and the RWB house the RBVS equipment.

The RBVS performs the following nonsafety-related system functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The RBVS supports the RXB by maintaining the RXB at a negative pressure relative to the outside atmosphere to control the movement of potentially airborne radioactivity from the RXB to the environment.
- The RBVS supports the RWB by maintaining the RWB at a negative ambient pressure relative to the outside atmosphere to control the movement of potentially airborne radioactivity from the RWB to the environment.
- The RBVS supports the highly reliable DC power system by providing ventilation to maintain airborne hydrogen concentrations below allowable limits.
- The RBVS supports the normal DC power system by providing ventilation to maintain airborne hydrogen concentrations below allowable limits.

Design Commitments

- The RBVS maintains a negative pressure in the RXB relative to the outside environment.
- The RBVS maintains a negative pressure in the RWB relative to the outside environment.
- The RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below one percent by volume.

3.3.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.3-1 contains the inspections, tests, and analyses for the RBVS.

Table 3.3-1: Reactor Building Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The RBVS maintains a negative pressure in the RXB relative to the outside environment.	A test will be performed of the RBVS while operating in the normal operating alignment.	The RBVS maintains a negative pressure in the RXB relative to the outside environment, while operating in the normal operating alignment.
2.	The RBVS maintains a negative pressure in the RWB relative to the outside environment.	A test will be performed of the RBVS while operating in the normal operating alignment.	The RBVS maintains a negative pressure in the RWB relative to the outside environment, while operating in the normal operating alignment.
3.	The RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below one percent by volume.	A test will be performed of the RBVS while operating in the normal operating alignment.	The airflow capability of the RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below one percent by volume.

3.4 Fuel Handling Equipment System

3.4.1 Design Description

System Description

The scope of this section is the fuel handling equipment (FHE) system. The FHE system is designed to support the periodic refueling of the reactor as well as movement of control rods and other radioactive components within the reactor core, refueling pool, and spent fuel pool. The FHE system is a nonsafety-related system. The FHE system supports up to 12 NuScale Power Modules (NPMs). The Reactor Building houses all FHE system equipment.

The FHE system performs the following nonsafety-related system function that is verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The FHE system supports the reactor fuel assembly by providing structural support during handling of fuel.

Design Commitments

- The fuel handling machine (FHM) main and auxiliary hoists are single-failure-proof in accordance with the approved design.
- The FHM main hoist is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.
- The FHM auxiliary hoist is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.
- The FHM welds comply with the American Society of Mechanical Engineers NOG-1 Code.
- The FHM travel is limited to maintain a water inventory for personnel shielding with the pool level at the lower limit of the normal operating low water level.
- The new fuel jib crane hook movement is limited to prevent carrying a fuel assembly over the fuel storage racks in the spent fuel pool.

3.4.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.4-1 contains the inspections, tests, and analyses for the FHE system.

Table 3.4-1: Fuel Handling Equipment System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The FHM main and auxiliary hoists are single-failure-proof in accordance with the approved design.	An inspection will be performed of the as-built FHM main and auxiliary hoists.	A report exists and concludes that the FHM main and auxiliary hoists are single-failure-proof in accordance with the approved design.
2.	The FHM main hoist is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.	A rated load test will be performed of the FHM main hoist.	The FHM main hoist lifts, supports, holds with the brakes, and transports a load of at least 125 percent of the manufacturer's rated capacity.
3.	The FHM auxiliary hoist is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.	A rated load test will be performed of the FHM auxiliary hoist.	The FHM auxiliary hoist lifts, supports, holds with the brakes, and transports a load of at least 125 percent of the manufacturer's rated capacity.
4.	The FHM welds comply with the American Society of Mechanical Engineers NOG-1 Code.	An inspection will be performed of the as-built FHM welds.	The results of the non-destructive examination of the FHM welds comply with American Society of Mechanical Engineers NOG-1 Code.
5.	The FHM travel is limited to maintain a water inventory for personnel shielding with the pool level at the lower limit of the normal operating low water level.	A test will be performed of the FHM gripper mast limit switches.	The FHM maintains at least 10 feet of water above the top of the fuel assembly when lifted to its maximum height with the pool level at the lower limit of the normal operating low water level.
6.	The new fuel jib crane hook movement is limited to prevent carrying a fuel assembly over the fuel storage racks in the spent fuel pool.	A test will be performed of new fuel jib crane interlocks.	The new fuel jib crane interlocks prevent the crane from carrying a fuel assembly over the spent fuel racks.

3.5 Fuel Storage System

3.5.1 Design Description

System Description

The scope of this section is the fuel storage system. The fuel storage system consists of the fuel storage racks in the spent fuel pool (SFP) that can store either spent fuel assemblies or new fuel assemblies. The fuel storage system is a nonsafety-related system. The fuel storage system supports up to 12 NuScale Power Modules (NPMs). The Reactor Building houses all fuel storage system equipment.

The fuel storage system performs the following nonsafety-related system functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The fuel storage system supports the reactor fuel assembly system by providing mechanical support for storage of new and spent fuel in a wet storage location.
- The fuel storage system supports the reactor fuel assembly system by providing neutron absorption to ensure subcriticality during storage of new and spent fuel.
- The fuel storage system supports the control rod assembly system by providing mechanical support for storage of control rods in fuel assemblies.

Design Commitments

- The fuel storage system American Society of Mechanical Engineers (ASME) Code Class NF components conform to the rules of construction of ASME Code Section III.
- The fuel storage racks maintain an effective neutron multiplication factor (k-effective) within the following limits at a 95 percent probability, 95 percent confidence level when loaded with fuel of the maximum reactivity to assure subcriticality during plant life, including normal operations and postulated accident conditions:
  – k-effective must not exceed 0.95 if flooded with borated water
  – k-effective must not exceed 1.0 if flooded with unborated water

3.5.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.5-1 contains the inspections, tests, and analyses for the fuel storage system.

Table 3.5-1: Fuel Storage System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The fuel storage system ASME Code Class NF components conform to the rules of construction of ASME Code Section III.	An inspection will be performed of the fuel storage system ASME Code Class NF as-built component Data Reports required by ASME Code Section III.	ASME Code Section III Data Reports for the fuel storage system ASME Code Class NF fuel storage racks exist and conclude that the requirements of ASME Code Section III are met.
2.	The fuel storage racks maintain an effective neutron multiplication factor (k-effective) within the following limits at a 95 percent probability, 95 percent confidence level when loaded with fuel of the maximum reactivity to assure subcriticality during plant life, including normal operations and postulated accident conditions: • k-effective must not exceed 0.95 if flooded with borated water • k-effective must not exceed 1.0 if flooded with unborated water	An inspection will be performed of the as-built fuel storage racks, their configuration in the SFP, and the associated documentation.	The as-built fuel storage racks, including any neutron absorbers, and their configuration within the SFP conform to the design values for materials and dimensions and their tolerances, as shown to be acceptable in the fuel storage criticality analysis described in the UFSAR.

3.6 Ultimate Heat Sink

3.6.1 Design Description

System Description

The scope of this section is the ultimate heat sink (UHS). The UHS is the system of structures and components credited for functioning as a heat sink for decay heat removal from the NuScale Power Modules during normal reactor operations or shutdown following an accident or transient, including a loss-of-coolant accident. The UHS is a safety-related system and supports up to 12 NuScale Power Modules. The Reactor Building (RXB) houses all UHS equipment.

The configuration of the UHS includes the combined volume of water in the reactor pool, refueling pool (RFP), and spent fuel pool (SFP). The pool areas are open to each other with a weir wall partially separating the SFP from the RFP. The dry dock area is not considered part of the UHS volume.

The structural components of the reactor pool, RFP, and SFP (i.e., structural walls, weir wall, and floor) and associated pool liners are a component of the RXB structure. The design commitments for the Reactor Building are provided in Tier 1 Section 3.11.

The UHS performs the following safety-related system functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The UHS supports the containment system by providing the removal of heat via direct water contact with the containment vessel.
- The UHS supports the decay heat removal system by accepting the heat from the decay heat removal heat exchanger.
- The UHS supports the spent fuel system by providing the removal of decay heat from the spent fuel via direct water contact with the spent fuel assemblies.

The UHS performs the following nonsafety-related system functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The UHS supports the containment system by providing the radiation shielding for the NPMs via the water surrounding the components.
- The UHS supports the spent fuel system by providing radiation shielding for spent fuel via the water surrounding the components.
- The UHS supports the RXB by having an assured water make-up line that can provide emergency make-up water to the UHS during off-normal events.

Design Commitments

- The UHS American Society of Mechanical Engineers (ASME) Code Class 3 piping system listed in Table 3.6-1 complies with ASME Code Section III requirements.
- The UHS Code Class 3 components listed in Table 3.6-1 conform to the rules of construction of ASME Code Section III.

- The spent fuel pool, refueling pool, reactor pool, and dry dock piping and connections are located to prevent the drain down of the SFP and reactor pool water level below the minimum safety water level.

3.6.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.6-2 contains the inspections, tests, and analyses for the UHS.

Table 3.6-1: Ultimate Heat Sink Piping System and Mechanical Equipment

Piping System Description	ASME Code Section III Class
Make-up line from the exterior of the RXB to the SFP.	3
Mechanical Equipment	
Equipment Name	ASME Code Section III Class
UHS make-up line isolation valve	3

Table 3.6-2: Ultimate Heat Sink Piping System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The ultimate heat sink ASME Code Class 3 piping system listed in Table 3.6-1 complies with ASME Code Section III requirements.	An inspection will be performed of the ultimate heat sink ASME Code Class 3 as-built piping system listed in Table 3.6-1 Design Report required by ASME Code Section III.	The ASME Code Section III Design Report (NCA-3550) exists and concludes that the ultimate heat sink ASME Code Class 3 as-built piping system listed in Table 3.6-1 meets the requirements of ASME Code Section III.
2.	The UHS Code Class 3 components listed in Table 3.6-1 conform to the rules of construction of ASME Code Section III.	An inspection will be performed of the UHS ASME Code Class 3 as-built component Data Report for the components listed in Table 3.6-1 required by ASME Code Section III.	The ASME Code Section III Data Report for the UHS ASME Code Class 3 components listed in Table 3.6-1 and interconnecting piping exists and concludes that the requirements of ASME Code Section III are met.
3.	The spent fuel pool, refueling pool, reactor pool, and dry dock piping and connections are located to prevent the drain down of the SFP and reactor pool water level below the minimum safety water level.	An inspection will be performed of the as-built SFP, RFP, reactor pool and dry dock piping and connections.	There are no gates, openings, drains, or piping within the SFP, RFP, reactor pool, and dry dock that are below 80 ft building elevation (55 ft pool level) as measured from the bottom of the SFP and reactor pool.

3.7 Fire Protection System

3.7.1 Design Description

System Description

The scope of this section is the fire protection system (FPS). The FPS is comprised of the equipment and components that provide early fire detection and suppression to limit the spread of fires. The FPS is a nonsafety-related system that supports up to 12 NuScale Power Modules (NPMs). The FPS equipment is located throughout the plant site.

The FPS includes the following equipment:

- fire water storage tanks, motor and diesel driven fire pumps, jockey pump, distribution piping, valves, and fire hydrants
- automatic fire detection, fire alarm notification, and fire suppression systems, including fire water supply and distribution systems
- manual firefighting capability, including portable fire extinguishers, standpipes, hydrants, hose stations, and fire department connections

The FPS performs the following nonsafety-related system functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The FPS supports the Reactor Building by providing fire prevention, detection, and suppression.
- The FPS supports the Radioactive Waste Building by providing fire prevention, detection, and suppression.
- The FPS supports the Control Building by providing fire prevention, detection, and suppression.

Design Commitments

- Two separate firewater storage tanks provide a dedicated volume of water for firefighting.
- The FPS has a sufficient number of fire pumps to provide the flow demand for the largest sprinkler or deluge system plus an additional 500 gpm for fire hoses assuming failure of the largest fire pump or loss of off-site power.
- Safe-shutdown can be achieved assuming that all equipment in any one fire area (except for the main control room (MCR) and under the bioshield) is rendered inoperable by fire damage and that reentry into the fire area for repairs and operator actions is not possible. An alternative shutdown capability that is physically and electrically independent of the MCR exists. Additionally, smoke, hot gases, or fire suppressant cannot migrate from the affected fire area into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions.
- A plant fire hazards analysis (FHA) considers potential fire hazards and ensures the fire protection features in each fire area are suitable for the hazards.

3.7.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.7-1 contains the inspections, tests, and analyses for the FPS.

Table 3.7-1: Fire Protection System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	Two separate firewater storage tanks provide a dedicated volume of water for firefighting.	An inspection will be performed of the as-built firewater storage tanks.	Each firewater storage tank provides a usable water volume dedicated for firefighting that is greater than or equal to 300,000 gallons.
2.	The FPS has a sufficient number of fire pumps to provide the flow demand for the largest sprinkler or deluge system plus an additional 500 gpm for fire hoses assuming failure of the largest fire pump or loss of off-site power.	i. An analysis will be performed of the as-built fire pumps. ii. A test will be performed of the fire pumps.	i. A report exists and concludes that the fire pumps can provide the flow demand for the largest sprinkler or deluge system plus an additional 500 gpm for fire hoses assuming failure of the largest fire pump or loss of off-site power. ii. Each fire pump delivers the design flow to the FPS.
3.	Safe-shutdown can be achieved assuming that all equipment in any one fire area (except for the MCR and under the bioshield) is rendered inoperable by fire damage and that reentry into the fire area for repairs and operator actions is not possible. An alternative shutdown capability that is physically and electrically independent of the MCR exists. Additionally, smoke, hot gases, or fire suppressant cannot migrate from the affected fire area into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions.	A safe-shutdown analysis of the as-built plant will be performed, including a post-fire safe-shutdown circuit analysis.	A safe-shutdown analysis report exists and concludes that: • Safe-shutdown can be achieved assuming that all equipment in any one fire area (except for the MCR and under the bioshield) is rendered inoperable by fire and that reentry into the fire area for repairs and operator actions is not possible • Smoke, hot gases, or fire suppressant cannot migrate from the affected fire area into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. • MPS equipment rooms within the Reactor Building used as the alternative shutdown capability are physically and electrically independent of the MCR.
4.	A plant FHA considers potential fire hazards and ensures the fire protection features in each fire area are suitable for the hazards.	A FHA of the as-built plant will be performed.	A FHA report exists and concludes that: • Combustible loads and ignition sources are accounted for, and • Fire protection features are suitable for the hazards they are intended to protect against.

3.8 Plant Lighting System

3.8.1 Design Description

System Description

The scope of this section is the plant lighting system (PLS). The PLS is a nonsafety-related system and supports up to 12 NuScale Power Modules (NPMs). The PLS provides artificial illumination for the entire plant: buildings (interior and exterior), rooms, spaces, and all outdoor areas of the plant. The PLS consists of normal and emergency lighting and includes miscellaneous non-lighting loads as required.

The PLS performs the following nonsafety-related system functions that are verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The PLS supports the Reactor Building by providing emergency lighting.
- The PLS supports the Control Building by providing normal lighting.
- The PLS supports the Control Building by providing emergency lighting in the main control room (MCR).

Design Commitments

- The PLS provides normal illumination of the operator workstations and auxiliary panels in the MCR.
- The PLS provides emergency illumination of the operator workstations and auxiliary panels in the MCR.
- Eight-hour battery-pack emergency lighting fixtures provide illumination for post-fire safe shutdown (FSSD) activities performed by operators outside the MCR and remote shutdown station (RSS) where post-FSSD activities are performed.

3.8.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.8-1 contains the inspections, tests, and analyses for the PLS.

Table 3.8-1: Plant Lighting System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The PLS provides normal illumination of the operator workstations and auxiliary panels in the MCR.	A test will be performed of the MCR operator workstations and auxiliary panel illumination.	The PLS provides at least 100 foot-candles illumination at the MCR operator workstations and at least 50 foot-candles at the auxiliary panels.
2.	The PLS provides emergency illumination of the operator workstations and auxiliary panels in the MCR.	A test will be performed of the MCR operator workstations and auxiliary panel illumination.	The PLS provides at least 10 foot-candles of illumination at the MCR operator workstations and auxiliary panels when it is the only MCR lighting system in operation.
3.	Eight-hour battery-pack emergency lighting fixtures provide illumination for post-FSSD activities performed by operators outside the MCR and RSS where post-FSSD activities are performed.	A test will be performed of the eight-hour battery-pack emergency lighting fixtures.	Eight-hour battery-pack emergency lighting fixtures illuminate their required target areas to provide at least one foot-candle illumination in the areas outside the MCR or RSS where post-FSSD activities are performed.

3.9 Radiation Monitoring - NuScale Power Modules 1 - 12

3.9.1 Design Description

System Description

The scope of this section is automatic actions of various systems based on radiation monitoring (RM). Automatic actions of systems based on RM are nonsafety-related functions. The systems actuated by these automatic RM functions are shared by NuScale Power Modules (NPMs) 1-12.

Design Commitments

- The normal control room HVAC system (CRVS) automatically responds to the CRVS high-radiation signals upstream of the CRVS filter unit listed in Table 3.9-1 to mitigate a release of radioactivity.
- The CRVS and the control room habitability system (CRHS) automatically respond to the CRVS high-radiation signals downstream of the CRVS filter unit listed in Table 3.9-1 to mitigate a release of radioactivity.
- The Reactor Building HVAC system (RBVS) automatically responds to the RBVS high-radiation signals listed in Table 3.9-1 to mitigate a release of radioactivity.
- The gaseous radioactive waste system (GRWS) automatically responds to the GRWS high-radiation signals listed in Table 3.9-1 to mitigate a release of radioactivity.
- The liquid radioactive waste system (LRWS) automatically responds to the LRWS high-radiation signals listed in Table 3.9-1 to mitigate a release of radioactivity.
- The auxiliary boiler system (ABS) automatically responds to the ABS high-radiation signals listed in Table 3.9-1 to mitigate a release of radioactivity.
- The pool surge control system (PSCS) automatically responds to the PSCS high-radiation signal listed in Table 3.9-1 to mitigate a release of radioactivity.

3.9.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.9-2 contains the inspections, tests, and analyses for radiation monitoring NPMs 1-12.

Table 3.9-1: Radiation Monitoring - NuScale Power Modules 1-12 Automatic Actions

Variable Monitored	Actuated Component(s)	Component Action(s)
CRVS outside air upstream of CRVS filter unit	1. CRVS filter unit bypass damper 2. CRVS filter unit bypass damper 3. CRVS filter unit inlet isolation damper 4. CRVS filter unit outlet isolation damper 5. CRVS filter unit fan	1. Close 2. Close 3. Open 4. Open 5. Start
CRVS outside air downstream of CRVS filter unit	1. CRVS outside air intake damper 2. CRVS outside air intake damper 3. CRVS filter unit fan 4. CRVS control room envelope supply damper 5. CRVS control room envelope supply damper 6. CRVS control room envelope return damper 7. CRVS control room envelope return damper 8. CRVS control room envelope smoke purge damper 9. CRVS control room envelope smoke purge damper 10. CRVS control room envelope exhaust damper 11. CRVS control room envelope exhaust damper 12. CRHS air supply isolation valve 13. CRHS air supply isolation valve 14. CRHS pressure relief isolation valve 15. CRHS pressure relief isolation valve	1. Close 2. Close 3. Stop 4. Close 5. Close 6. Close 7. Close 8. Close 9. Close 10. Close 11. Close 12. Open 13. Open 14. Open 15. Open
RBVS spent fuel pool exhaust	1. RBVS Reactor Building general exhaust isolation damper for the spent fuel pool and dry dock area 2. RBVS spent fuel pool filter unit A inlet isolation damper 3. RBVS spent fuel pool filter unit A outlet isolation damper 4. RBVS spent fuel pool filter unit A bypass isolation damper 5. RBVS spent fuel pool filter unit B inlet isolation damper 6. RBVS spent fuel pool filter unit B outlet isolation damper 7. RBVS spent fuel pool filter unit B bypass isolation damper 8. RBVS main supply AHU fan 9. RBVS main supply AHU fan 10. RBVS main supply AHU fan 11. RBVS main supply AHU fan	1. Close 2. Open 3. Open 4. Close 5. Open 6. Open 7. Close 8. Reduce flow to maintain Reactor Building (RXB) & Radioactive Waste Building (RWB) dP 9. Reduce flow to maintain RXB & RWB dP 10. Reduce flow to maintain RXB & RWB dP 11. Reduce flow to maintain RXB & RWB dP
GRWS train A charcoal decay bed discharge	1. GRWS train A charcoal bed effluent isolation valve	1. Close

Table 3.9-1: Radiation Monitoring - NuScale Power Modules 1-12 Automatic Actions (Continued)

Variable Monitored	Actuated Component(s)	Component Action(s)
GRWS train B charcoal decay bed discharge	1. GRWS train B charcoal bed effluent isolation valve	1. Close
GRWS effluent to RBVS	1. GRWS common charcoal bed effluent isolation valve 2. GRWS common charcoal bed effluent isolation valve	1. Close 2. Close
LRWS discharge to utility water system (UWS)	1. LRWS to UWS isolation valve 2. LRWS to UWS isolation valve	1. Close 2. Close
ABS flash tank vent	1. ABS flash tank vent pressure control valve 2. ABS high pressure steam supply isolation valve 3. ABS high pressure steam supply isolation valve	1. Close 2. Close 3. Close
ABS high pressure to low pressure steam supply	1. ABS high pressure to low pressure steam supply pressure control valve	1. Close
PSCS tank vent	1. PSCS tank inlet isolation valve 2. PSCS tank outlet isolation valve	1. Close 2. Close

Table 3.9-2: Radiation Monitoring - NuScale Power Modules 1-12 Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The CRVS automatically responds to the CRVS high-radiation signals upstream of the CRVS filter unit listed in Table 3.9-1 to mitigate a release of radioactivity.	A test will be performed of the CRVS high-radiation signals listed in Table 3.9-1.	Upon initiation of the real or simulated CRVS high-radiation signals upstream of the CRVS filter unit listed in Table 3.9-1, the CRVS automatically aligns/actuates the identified components to the positions identified in the table.
2.	The CRVS and the CRHS automatically respond to the CRVS high-radiation signals downstream of the CRVS filter unit listed in Table 3.9-1 to mitigate a release of radioactivity.	A test will be performed of the CRVS high-radiation signals listed in Table 3.9-1.	Upon initiation of the real or simulated CRVS high-radiation signals downstream of the CRVS filter unit listed in Table 3.9-1, the CRVS and the CRHS automatically align/actuate the identified components to the positions identified in the table.
3.	The RBVS automatically responds to the RBVS high-radiation signals listed in Table 3.9-1 to mitigate a release of radioactivity.	A test will be performed of the RBVS high-radiation signals listed in Table 3.9-1.	Upon initiation of the real or simulated RBVS high-radiation signals listed in Table 3.9-1, the RBVS automatically aligns/actuates the identified components to the positions identified in the table.
4.	The GRWS automatically responds to the GRWS high-radiation signals listed in Table 3.9-1 to mitigate a release of radioactivity.	A test will be performed of the GRWS high-radiation signals listed in Table 3.9-1.	Upon initiation of the real or simulated GRWS high-radiation signals listed in Table 3.9-1, the GRWS automatically aligns/actuates the identified components to the positions identified in the table.
5.	Not Used.	Not Used.	Not Used.
6.	Not Used.	Not Used.	Not Used.
7.	The LRWS automatically responds to the LRWS high-radiation signals listed in Table 3.9-1 to mitigate a release of radioactivity.	A test will be performed of the LRWS high-radiation signals listed in Table 3.9-1.	Upon initiation of the real or simulated LRWS high-radiation signals listed in Table 3.9-1, the LRWS automatically aligns/actuates the identified components to the positions identified in the table.
8.	The ABS automatically responds to the ABS high-radiation signals listed in Table 3.9-1 to mitigate a release of radioactivity.	A test will be performed of the ABS high-radiation signals listed in Table 3.9-1.	Upon initiation of the real or simulated ABS high-radiation signals listed in Table 3.9-1, the ABS automatically aligns/actuates the identified components to the positions identified in the table.
9.	Not Used.	Not Used.	Not Used.
10.	The PSCS automatically responds to the PSCS high-radiation signal listed in Table 3.9-1 to mitigate a release of radioactivity.	A test will be performed of the PSCS high-radiation signal listed in Table 3.9-1.	Upon initiation of the real or simulated PSCS high-radiation signal listed in Table 3.9-1, the PSCS automatically aligns/actuates the identified components to the positions identified in the table.

3.10 Reactor Building Crane

3.10.1 Design Description

System Description

The scope of this section is the Reactor Building crane (RBC). The RBC is a bridge crane that rides on rails anchored to the Reactor Building. The bridge crane can travel the length of the reactor pool, refueling pool, and the dry dock. The RBC is nonsafety-related and supports up to 12 NuScale Power Modules (NPMs). The Reactor Building houses all RBC equipment.

The RBC includes the following:

- RBC with auxiliary hoist
- below-the-hook lifting devices, including the module lifting adapter (MLA) and the wet hoist

The RBC performs the following risk-significant system function that is verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The RBC supports the NuScale Power Module by providing structural support and mobility while moving from refueling, inspection and operating bay.

Design Commitments

- The RBC main hoist is single-failure-proof in accordance with the approved design.
- The RBC auxiliary hoists are single-failure-proof in accordance with the approved design.
- The RBC wet hoist is single-failure-proof in accordance with the approved design.
- The RBC main hoist is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.
- The RBC auxiliary hoists are capable of lifting and supporting their rated load, holding the rated load, and transporting the rated load.
- The RBC wet hoist is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.
- All RBC weld joints whose failure could result in the drop of a critical load comply with the American Society of Mechanical Engineers NOG-1 Code.
- The MLA is capable of supporting its rated load.
- The MLA is single-failure-proof in accordance with the approved design.

3.10.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.10-1 contains the inspections, tests, and analyses for the RBC.

Table 3.10-1: Reactor Building Crane Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The RBC main hoist is single-failure-proof in accordance with the approved design.	An inspection will be performed of the as-built RBC main hoist.	A report exists and concludes that the RBC main hoist is single-failure-proof in accordance with the approved design.
2.	The RBC auxiliary hoists are single-failure-proof in accordance with the approved design.	An inspection will be performed of the as-built RBC auxiliary hoists.	A report exists and concludes that the RBC auxiliary hoists are single-failure-proof in accordance with the approved design.
3.	The RBC wet hoist is single-failure-proof in accordance with the approved design.	An inspection will be performed of the as-built RBC wet hoist.	A report exists and concludes that the RBC wet hoist is single-failure-proof in accordance with the approved design.
4.	The RBC main hoist is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.	A rated load test will be performed of the RBC main hoist.	The RBC main hoist lifts, supports, holds with the brakes, and transports a load of at least 125 percent of the manufacturer's rated capacity.
5.	The RBC auxiliary hoists are capable of lifting and supporting their rated load, holding the rated load, and transporting the rated load.	A rated load test will be performed of the RBC auxiliary hoists.	The RBC auxiliary hoists lift, support, hold with the brakes, and transport a load of at least 125 percent of the manufacturer's rated capacity.
6.	The RBC wet hoist is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.	A rated load test will be performed of the RBC wet hoist.	The RBC wet hoist lifts, supports, holds with the brakes, and transports a load of at least 125 percent of the manufacturer's rated capacity.
7.	All RBC weld joints whose failure could result in the drop of a critical load comply with the American Society of Mechanical Engineers NOG-1 Code.	An inspection will be performed of the as-built RBC weld joints whose failure could result in the drop of a critical load.	The results of the non-destructive examination of the RBC weld joints whose failure could result in the drop of a critical load comply with American Society of Mechanical Engineers NOG-1 Code.
8.	Not Used.	Not Used.	Not Used.
9.	The MLA is capable of supporting its rated load.	i. A rated load test will be performed of the MLA single load path elements. ii. A rated load test will be performed of the MLA dual load path elements.	i. The MLA single load path elements support a load of at least 300 percent of the manufacturer's rated capacity. ii. The MLA dual load path elements support a load of at least 150 percent of the manufacturer's rated capacity.
10.	The MLA is single-failure-proof in accordance with the approved design.	An inspection will be performed of the as-built MLA.	A report exists and concludes that the MLA is single-failure-proof in accordance with the approved design.

3.11 Reactor Building

3.11.1 Design Description

Building Description

The scope of this section is the Reactor Building (RXB). The RXB is a safety-related structure. The RXB supports up to 12 NuScale Power Modules. The RXB is a reinforced-concrete structure that is embedded in soil and supported on a basemat foundation. The RXB houses all Reactor Building components equipment.

The RXB performs the following safety-related system function that is verified by Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC):

- The RXB supports the following systems by housing and providing structural support:
  - NuScale Power Module
  - chemical and volume control system (CVCS)
  - ultimate heat sink
  - module protection system
  - nuclear monitoring system

The RXB performs the following nonsafety-related, risk-significant system function that is verified by ITAAC:

- The RXB supports the RXB crane by housing and providing structural support.

Design Commitments

- Fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the RXB fire area of origin.
- Internal flooding barriers provide confinement so that the impact from internal flooding is contained within the RXB flooding area of origin.
- The Seismic Category I RXB is protected against external flooding in order to prevent flooding of safety-related structures, systems, and components (SSC) within the structure.
- The RXB includes radiation shielding barriers for normal operation and post-accident radiation shielding.
- The RXB includes radiation attenuating doors for normal operation and post-accident radiation shielding. These doors have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed.
- The RXB is Seismic Category I and maintains its structural integrity under the design basis loads.
- Non-Seismic Category I SSC located where there is a potential for adverse interaction with the RXB or a Seismic Category I SSC in the RXB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following a safe shutdown earthquake (SSE).

- Safety-related SSC are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.

3.11.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.11-2 contains the inspections, tests, and analyses for the RXB.

Table 3.11-1: Not Used

Table 3.11-2: Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	Fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases,or fire suppressants is contained within the RXB fire area of origin.	An inspection will be performed of the RXB as-built fire and smoke barriers.	The following RXB fire and smoke barriers exist in accordance with the fire hazards analysis, and have been qualified for the fire rating specified in the fire hazards analysis: • fire-rated doors • fire-rated penetration seals • fire-rated dampers • fire-rated walls, floors, and ceilings • smoke barriers
2.	Internal flooding barriers provide confinement so that the impact from internal flooding is contained within the RXB flooding area of origin.	An inspection will be performed of the RXB as-built internal flooding barriers.	The following RXB internal flooding barriers exist in accordance with the internal flooding analysis report and have been qualified as specified in the internal flooding analysis report: • flood resistant doors • curbs and sills • walls • water tight penetration seals • National Electrical Manufacturer's Association enclosures
3.	The Seismic Category I RXB is protected against external flooding in order to prevent flooding of safety-related SSC within the structure.	An inspection will be performed of the RXB as-built floor elevation at ground entrances.	The RXB floor elevation at ground entrances is higher than the maximum external flood elevation.
4.	The RXB includes radiation shielding barriers for normal operation and post-accident radiation shielding.	An inspection and analysis will be performed of the as-built RXB radiation shielding barriers.	A report exists and concludes the radiation attenuation capability of RXB radiation shielding barriers is greater than or equal to the required attenuation capability of the approved design.
5.	The RXB includes radiation attenuating doors for normal operation and for post-accident radiation shielding. These doors have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed.	An inspection will be performed of the as-built RXB radiation attenuating doors.	The RXB radiation attenuating doors are installed in their design location and have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed.
6.	The RXB is Seismic Category I and maintains its structural integrity under the design basis loads.	A reconciliation analysis will be performed of the as-built RXB under the actual design basis loads.	A design summary report exists and concludes that (1) the as-built RXB maintains its structural integrity in accordance with the approved design under the actual design basis loads, and (2) the in-structure responses for the as-built RXB are enveloped by those in the approved design.

Table 3.11-2: Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
7.	Non-Seismic Category I SSC located where there is a potential for adverse interaction with the RXB or a Seismic Category I SSC in the RXB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following a SSE.	An inspection and analysis will be performed of the as-built non-Seismic Category I SSC located where there is a potential for adverse interaction with the RXB or a Seismic Category I SSC in the RXB.	A report exists and concludes that the Non-Seismic Category I SSC located where there is a potential for adverse interaction with the RXB or a Seismic Category I SSC in the RXB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following an SSE as demonstrated by one or more of the following criteria: • Seismic Category I SSC are isolated from non-Seismic Category I SSC, so that interaction does not occur. • Seismic Category I SSC are analyzed to confirm that the ability to perform their safety functions is not impaired as a result of impact from non-Seismic Category I SSC. • A non-Seismic Category I restraint system designed to Seismic Category I requirements is used to assure that no interaction occurs between Seismic Category I SSC and non-Seismic Category I SSC.
8.	Safety-related SSC are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.	An inspection and analysis will be performed of the as-built high- and moderate-energy piping systems and protective features for the safety-related SSC located in the RXB outside the Reactor Pool Bay.	Protective features are installed in accordance with the as-built Pipe Break Hazard Analysis Report and safety-related SSC are protected against or qualified to withstand the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.

3.12 Radioactive Waste Building

3.12.1 Design Description

Building Description

The scope of this section is the Radioactive Waste Building (RWB). The RWB is a nonsafety-related building which supports up to 12 NuScale Power Modules (NPMs). The RWB is located west of the Reactor Building (RXB) and serves as the primary radioactive waste facility to collect waste from the RXB and the Annex Building.

The RWB is a reinforced-concrete structure with a concrete roof supported on a steel frame. It is embedded in soil and is supported on a foundation basemat. There are penetrations in the east wall and in the west wall through which the NuScale Power Module is transported into the RXB using a module import trolley.

Design Commitments

- The RWB includes radiation shielding barriers for normal operation and post-accident radiation shielding.
- The RWB includes radiation attenuating doors for normal operation and for post-accident radiation shielding. These doors have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed.
- The RWB is an RW-IIa structure and maintains its structural integrity under the design basis loads.

3.12.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.12-2 contains the inspections, tests, and analyses for the RWB.

Table 3.12-1: Not Used

Table 3.12-2: Radioactive Waste Building ITAAC

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The RWB includes radiation shielding barriers for normal operation and post-accident radiation shielding.	An inspection and analysis will be performed of the as-built RWB radiation shielding barriers.	A report exists and concludes the radiation attenuation capability of RWB radiation shielding barriers is greater than or equal to the required attenuation capability of the approved design.
2.	The RWB includes radiation attenuating doors for normal operation and for post-accident radiation shielding. These doors have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed.	An inspection will be performed of the as-built RWB radiation attenuating doors.	The RWB radiation attenuating doors are installed in their design location and have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed.
3.	The RWB is an RW-IIa structure and maintains its structural integrity under the design basis loads.	A reconciliation analysis will be performed of the as-built RW-IIa RWB under the actual design basis loads.	A design summary report exists and concludes that (1) the as-built RWB maintains its structural integrity in accordance with the approved design under the actual design basis loads, and (2) the in-structure responses for the as-built RWB are enveloped by those in the approved design.

3.13 Control Building

3.13.1 Design Description

Building Description

The scope of this section is the Control Building (CRB). The CRB is a safety-related building that supports up to 12 NuScale Power Modules (NPMs). The CRB houses the main control room, the technical support center, the control room habitability system, the normal control room HVAC system, and safety and nonsafety control and instrumentation systems.

The CRB is designated as Seismic Category I except for the following areas which are designated Seismic Category II:

- above the 120'-0" elevation
- inside the elevator shaft (full building height)
- inside the two stairwells (full building height)
- the fire protection vestibule located on the East side of the CRB

The CRB is a reinforced-concrete building with an upper steel structure supporting the roof and has an underground equipment tunnel that connects to the Reactor Building. The tunnel is comprised of two levels-- an upper tunnel for personnel access to the Reactor Building and a lower tunnel that is a utilities tunnel between the CRB and for the Reactor Building. Above Elevation 120'-0", the CRB is a steel structure supporting a steel roof.

The CRB performs the following safety-related system function that is verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The CRB supports the module protection system by housing and providing structural support.

The CRB performs the following nonsafety-related system function that is verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The CRB supports the normal control room HVAC system by providing a portion of the control room envelope.

Design Commitments

- Fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the CRB fire area of origin.
- Internal flooding barriers provide confinement so that the impact from internal flooding is contained within the CRB flooding area of origin.
- The Seismic Category I CRB is protected against external flooding in order to prevent flooding of safety-related structures, systems, and components (SSC) within the structure.
- The CRB at Elevation 120'-0" and below (except for the elevator shaft, the stairwells and the fire protection vestibule which are Seismic Category II) is Seismic Category I and maintains its structural integrity under the design basis loads.

- Non-Seismic Category I SSC located where there is a potential for adverse interaction with the CRB or a Seismic Category I SSC in the CRB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following a safe shutdown earthquake.

3.13.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.13-1 contains the inspections, tests, and analyses for the CRB.

Table 3.13-1: Control Building Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	Fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the CRB fire area of origin.	An inspection will be performed of the CRB as-built fire and smoke barriers.	The following CRB fire and smoke barriers exist in accordance with the fire hazards analysis, and have been qualified for the fire rating specified in the fire hazards analysis: • fire-rated doors • fire-rated penetration seals • fire-rated dampers • fire-rated walls, floors, and ceilings • smoke barriers
2.	Internal flooding barriers provide confinement so that the impact from internal flooding is contained within the CRB flooding area of origin.	An inspection will be performed of the CRB as-built internal flooding barriers.	The following CRB internal flooding barriers exist in accordance with the internal flooding analysis report and have been qualified as specified in the internal flooding analysis report: • flood resistant doors • walls • water tight penetration seals • National Electrical Manufacturer's Association (NEMA) enclosures
3.	The Seismic Category I CRB is protected against external flooding in order to prevent flooding of safety-related SSC within the structure.	An inspection will be performed of the CRB as-built floor elevation at ground entrances.	The CRB floor elevation at ground entrances is higher than the maximum external flood elevation.
4.	The CRB at Elevation 120’-0” and below (except for the elevator shaft, the stairwells, and the fire protection vestibule which are Seismic Category II) is Seismic Category I and maintains its structural integrity under the design basis loads.	A reconciliation analysis will be performed of the as-built CRB at Elevation 120'-0" and below under the actual design basis loads.	A design summary report exists and concludes that (1) the as-built CRB at Elevation 120'-0" and below maintains its structural integrity in accordance with the approved design under the actual design basis loads, and (2) the in-structure responses for the as-built CRB at Elevation 120'-0" and below are enveloped by those in the approved design.

Table 3.13-1: Control Building Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5.	Non-Seismic Category I SSC located where there is a potential for adverse interaction with the CRB or a Seismic Category I SSC in the CRB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following a safe shutdown earthquake.	An inspection and analysis will be performed of the as-built non-Seismic Category I SSC located where there is a potential for adverse interaction with the CRB or a Seismic Category I SSC in the CRB.	A report exists and concludes that the Non-Seismic Category I SSC located where there is a potential for adverse interaction with the CRB or a Seismic Category I SSC in the CRB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following an SSE as demonstrated by one or more of the following criteria: • Seismic Category I SSC are isolated from non-Seismic Category I SSC, so that interaction does not occur. • Seismic Category I SSC are analyzed to confirm that the ability to perform their safety functions is not impaired as a result of impact from non-Seismic Category I SSC. • A non-Seismic Category I restraint system designed to Seismic Category I requirements is used to assure that no interaction occurs between Seismic Category I SSC and non-Seismic Category I SSC.

3.14 Equipment Qualification - Shared Equipment

3.14.1 Design Description

System Description

The scope of this section is equipment qualification (EQ) of equipment shared by NuScale Power Modules 1 through 12.

This section applies to the safety-related reactor pressure vessel (RPV) support stand and Reactor Building (RXB) over-pressurization vents, and a limited population of common, nonsafety-related equipment that has augmented Seismic Category I or environmental qualification requirements. The nonsafety-related equipment in this section provides one of the following nonsafety-related functions:

- Provides physical support of irradiated fuel (fuel handling machine, spent fuel storage racks, reactor building crane, and module lifting adapter).
- Provides a path for makeup water to the ultimate heat sink (UHS).
- Provides containment of the UHS water.
- Monitors UHS water level.

Additionally, this section applies to the nonsafety-related, RW-IIa components and piping used for processing gaseous radioactive waste.

Design Commitments

- The common, Seismic Category I equipment listed in Table 3.14-1, including its associated supports and anchorages, withstands design basis seismic loads without loss of its function(s) during and after a safe shutdown earthquake.
- The common electrical equipment listed in Table 3.14-1 located in a harsh environment, including its connection assemblies, withstands the design basis harsh environmental conditions experienced during normal operations, anticipated operational occurrences, design basis accidents, and post-accident conditions, and performs its function for the period of time required to complete the function.
- The RW-IIa components and piping used for processing gaseous radioactive waste listed in Table 3.14-1 are constructed to the standards of RW-IIa.

3.14.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.14-2 contains the inspections, tests, and analyses for EQ-shared equipment.

Table 3.14-1: Mechanical and Electrical/Instrumentation and Controls Shared Equipment

Description	Location	EQ Environment	EQ Program	Seismic Category	Class 1E	EQ Category(1)
Module Assembly Equipment - Bolting						
RPV Support Stand	RXB - UHS	N/A	N/A	I	N/A	N/A
Fuel Handling Equipment						
Fuel handling machine (FHM)	RXB 100'-0" Elevation	Harsh	Electrical Mechanical	I	No	B
Spent Fuel Storage System						
Fuel Storage Racks(14 Total)	RXB - Spent Fuel Pool	N/A	N/A	I	N/A	N/A
Ultimate Heat Sink						
Pool level instruments (8 Total)	RXB - UHS	Harsh	Electrical	I	No	A
Water Makeup Line	RXB - UHS	N/A	N/A	I	N/A	N/A
Reactor Building Cranes						
Reactor Building crane	RXB 100’-0”’ thru 145’-6” Elevation	Harsh	Electrical Mechanical	I	No	B
Module Lifting Adapter	RXB - Various	N/A	N/A	I	N/A	N/A
Reactor Building Components						
UHS Pool Liner and Dry Dock Liner	RXB - UHS	N/A	N/A	I	N/A	N/A
RXB over-pressurization vents (34 Total)	RXB	N/A	N/A	I	N/A	A
Liquid Radioactive Waste System						
Degasifiers (2 Total)	RXB	N/A	N/A	RW-IIa	N/A	N/A
For each line connected to a degasifier, piping and components up to and including the first isolation valve	RXB	N/A	N/A	RW-IIa	N/A	N/A
Gaseous Radioactive Waste System						
Charcoal Guard Bed	RWB	N/A	N/A	RW-IIa	N/A	N/A
For each line connected to the charcoal guard bed, piping and components up to and including the first isolation valve	RWB	N/A	N/A	RW-IIa	N/A	N/A
Charcoal Decay Beds (8 Total)	RWB	N/A	N/A	RW-IIa	N/A	N/A

Table 3.14-1: Mechanical and Electrical/Instrumentation and Controls Shared Equipment (Continued)

Description	Location	EQ Environment	EQ Program	Seismic Category	Class 1E	EQ Category(1)
For each line connected to a charcoal decay bed, piping and components up to and including the first isolation valve	RWB	N/A	N/A	RW-IIa	N/A	N/A

Notes:

1. EQ Categories:

- A - Equipment that will experience the environmental conditions of design basis accidents for which it must function to mitigate said accidents, and that will be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.
- B - Equipment that will experience the environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that will be qualified to demonstrate the capability to withstand the accident environment for the time during which it must not fail with safety margin to failure.

Table 3.14-2: Equipment Qualification - Shared Equipment ITAAC

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The common Seismic Category I equipment listed in Table 3.14-1, including its associated supports and anchorages, withstands design basis seismic loads without loss of its function(s) during and after a safe shutdown earthquake.	i. A type test, analysis, or a combination of type test and analysis will be performed of the common Seismic Category I equipment listed in Table 3.14-1, including its associated supports and anchorages.	i. A Seismic Qualification Report exists and concludes that the common Seismic Category I equipment listed in Table 3.14-1, including its associated supports and anchorages, will withstand the design basis seismic loads and perform its function during and after a safe shutdown earthquake.
		ii. An inspection will be performed of the common Seismic Category I as-built equipment listed in Table 3.14-1, including its associated supports and anchorages.	ii. The common Seismic Category I equipment listed in Table 3.14-1, including its associated supports and anchorages, is installed in its design location in a Seismic Category I structure in a configuration bounded by the equipment's Seismic Qualification Report.
2.	The common electrical equipment listed in Table 3.14-1 located in a harsh environment, including its connection assemblies, withstands the design basis harsh environmental conditions experienced during normal operations, anticipated operational occurrences, DBA, and post-accident conditions and performs its function for the period of time required to complete the function.	i. A type test or a combination of type test and analysis will be performed of the common electrical equipment listed in Table 3.14-1, including its connection assemblies.	i. An equipment qualification record form exists and concludes that the common electrical equipment listed in Table 3.14-1, including its connection assemblies, performs its function under the environmental conditions specified in the equipment qualification record form for the period of time required to complete the function.
		ii. An inspection will be performed of the common as-built electrical equipment listed in Table 3.14-1, including its connection assemblies.	ii. The common electrical equipment listed in Table 3.14-1, including its connection assemblies, is installed in its design location in a configuration bounded by the EQ record form.
3.	The RW-IIa components and piping used for processing gaseous radioactive waste listed in Table 3.14-1 are constructed to the standards of RW-IIa.	i. An inspection and reconciliation analysis will be performed of the as-built RW-IIa components and piping used for processing gaseous radioactive waste listed in Table 3.14-1.	i. A report exists and concludes that the as-built RW-IIa components and piping used for processing gaseous radioactive waste listed in Table 3.14-1 meet the RW-IIa design criteria.

3.15 Human Factors Engineering

3.15.1 Design Description

System Description

The human factors engineering (HFE) program design process is employed to design the control rooms and the human-system interfaces (HSIs) and associated equipment while relating the high-level goal of plant safety into individual, discrete focus areas for the design.

The HFE and control room design team establish design guidelines, define program-specific design processes, and verify that the guidelines and processes are followed. The scope of the HFE program includes the following:

- location and accessibility requirements for the control rooms and other control stations
- layout requirements of the control rooms, including requirements regarding the locations and design of individual displays and panels
- basic concepts and detailed design requirements for the information displays, controls, and alarms for HSI control stations
- coding and labeling conventions for control room components and plant displays
- HFE design requirements and guidelines for the screen-based HSI, including the actual screen layout and the standard dialogues for accessing information and controls
- requirements for the physical environment of the control rooms (e.g., lighting, acoustics, heating, ventilation, and air conditioning)
- HFE requirements and guidelines regarding the layout of operator workstations and work spaces
- corporate policies and procedures regarding the verification and validation of the design of HSI

The HFE program applies to the design of the main control room (MCR) and the remote shutdown station. The HSI of the technical support center, the emergency operations facility, and local control stations (LCS) are derivatives of the MCR HSI. The design of LCS is accomplished concurrently with the applicable system design and follows guidelines established by the HFE and control room design team.

Design Commitments

- The configuration of the MCR HSI is consistent with the design verified and validated by the integrated system validation as reconciled by the Design Implementation Implementation Plan.

3.15.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.15-1 contains the inspections, tests, and analyses for the HFE.

Table 3.15-1: Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The configuration of the main control room HSI is consistent with the design verified and validated by the integrated system validation as reconciled by the Design Implementation Implementation Plan.	An inspection will be performed of the as-built configuration of MCR HSI.	A report exists and concludes the as-built configuration of main control room HSI is consistent with the design verified and validated by the integrated system validation as reconciled by the Design Implementation Implementation Plan.

3.16 Physical Security System

3.16.1 Design Description

System Description

The NuScale Power Plant physical security system design provides the capabilities to detect, assess, impede and delay threats up to and including the design basis threat, and to provide for defense-in-depth through the integration of systems, technologies, and equipment.

Design Commitments

- Vital equipment will be located only within a vital area.
- Access to vital equipment requires passage through at least two physical barriers.
- The external walls, doors, ceilings, and floors in the main control room (MCR) and central alarm station (CAS) will be bullet-resistant.
- An access control system will be installed and designed for use by individuals who are authorized access to vital areas within the nuclear island and structures without escort.
- Unoccupied vital areas within the nuclear island and structures will be designed with locking devices and intrusion-detection devices that annunciate in the CAS.
- The CAS will be located inside the protected area and will be designed so that the interior is not visible from the perimeter of the protected area.
- Security alarm devices in the Reactor Building (RXB) and Control Building (CRB), including transmission lines to annunciators, will be tamper-indicating and self-checking, and alarm annunciation indicates the type of alarm and its location.
- Intrusion detection and assessment systems in the RXB and CRB will be designed to provide visual display and audible annunciation of alarms in the CAS.
- Intrusion detection systems' recording equipment will record security alarm annunciations with the nuclear island and structures, including each alarm, false alarm, alarm check, and tamper indication and the type of alarm, location, alarm circuit, date, and time.
- Emergency exits through the vital area boundaries within the nuclear island and structures will be alarmed with intrusion detection devices and are secured by locking devices that allow prompt egress during an emergency.
- The CAS will have landline telephone service with the control room and local law enforcement authorities.
- The CAS will be capable of continuous communication with on-duty security force personnel.
- Non-portable communications equipment in the CAS will remain operable from an independent power source in the event of the loss of normal power.

3.16.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.16-1 contains the inspections, tests, and analyses for physical security system.

Table 3.16-1: Physical Security System Inspections, Tests, Analyses, and Acceptance Criteria

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	Vital equipment will be located only within a vital area.	All vital equipment locations will be inspected.	Vital equipment is located only within a vital area.
2.	Access to vital equipment requires passage through at least two physical barriers.	All vital equipment physical barriers will be inspected.	Vital equipment is located within a protected area such that access to the vital equipment requires passage through at least two physical barriers.
3.	The external walls, doors, ceilings, and floors in the MCR and CAS will be bullet-resistant.	Type test, analysis, or a combination of type test and analysis of the external walls, doors, ceilings, and floors in the MCR and CAS, will be performed.	A report exists and concludes that the walls, doors, ceilings, and floors in the MCR and CAS are bullet-resistant.
4.	An access control system will be installed and designed for use by individuals who are authorized access to vital areas within the nuclear island and structures without escort.	The access control system will be tested.	The access control system is installed and provides authorized access to vital areas within the nuclear island and structures only to those individuals with authorization for unescorted access.
5.	Unoccupied vital areas within the nuclear island and structures will be designed with locking devices and intrusion detection devices that annunciate in the CAS.	Tests, inspections, or a combination of tests and inspections of unoccupied vital areas' intrusion detection equipment and locking devices will be performed.	Unoccupied vital areas within the nuclear island and structures are locked and alarmed and intrusion is detected and annunciated in the CAS.
6.	The CAS will be located inside the protected area and will be designed so that the interior is not visible from the perimeter of the protected area.	The CAS will be inspected.	The CAS is located inside the protected area, and the interior of the alarm station is not visible from the perimeter of the protected area.
7.	Security alarm devices in the RXB and CRB, including transmission lines to annunciators, will be tamper-indicating and self-checking, and alarm annunciation indicates the type of alarm and its location.	All security alarm devices and transmission lines in the RXB and CRB will be tested.	Security alarm devices, in the RXB and CRB including transmission lines to annunciators, are tamper-indicating and self-checking; an automatic indication is provided when failure of the alarm system or a component thereof occurs or when the system is on standby power; the alarm annunciation indicates the type of alarm and location.
8.	Intrusion detection and assessment systems in the RXB and CRB will be designed to provide visual display and audible annunciation of alarms in the CAS.	Intrusion detection and assessment systems in the RXB and CRB will be tested.	The intrusion detection systems, in the RXB and CRB provide a visual display and audible annunciation of all alarms in the CAS.
9.	Intrusion detection systems' recording equipment will record security alarm annunciations within the nuclear island and structures including each alarm, false alarm, alarm check, and tamper indication, and the type of alarm, location, alarm circuit, date, and time.	The intrusion detection systems' recording equipment will be tested.	Intrusion detection systems' recording equipment is capable of recording each security alarm annunciation within the nuclear island and structures, including each alarm, false alarm, alarm check, and tamper indication and the type of alarm, location, alarm circuit, date, and time.

Table 3.16-1: Physical Security System Inspections, Tests, Analyses, and Acceptance Criteria (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10.	Emergency exits through the vital area boundaries within the nuclear island and structures will be alarmed with intrusion detection devices and are secured by locking devices that allow prompt egress during an emergency.	Tests, inspections, or a combination of tests and inspections of emergency exits through vital area boundaries within the nuclear island and structures will be performed.	Emergency exits through the vital area boundaries within the nuclear island and structures are alarmed with intrusion detection devices and secured by locking devices that allow prompt egress during an emergency.
11.	The CAS will have a landline telephone service with the control room and local law enforcement authorities.	Tests, inspections, or a combination of tests and inspections of the CAS's landline telephone service will be performed.	The CAS is equipped with landline telephone service with the control room and local law enforcement authorities.
12.	The CAS will be capable of continuous communication with on-duty security force personnel.	Tests, inspections, or a combination of tests and inspections of the CAS's continuous communication capabilities will be performed.	The CAS is capable of continuous communication with on-duty watchmen, armed security officers, armed responders, or other security personnel who have responsibilities within the physical protection program and during contingency response events.
13.	Non-portable communications equipment in the CAS will remain operable from an independent power source in the event of the loss of normal power.	Tests, inspections, or a combination of tests and inspections of the nonportable communications equipment will be performed.	All nonportable communication devices in the CAS remain operable from an independent power source in the event of the loss of normal power.

3.17 Radiation Monitoring - NuScale Power Modules 1 - 6

3.17.1 Design Description

System Description

The scope of this section is automatic actions of various systems based on radiation monitoring. Automatic actions of systems based on radiation monitoring are nonsafety-related functions. The systems actuated by these automatic radiation monitoring functions are shared by NuScale Power Modules (NPMs) 1 through 6.

Design Commitments

- The containment flooding and drain system (CFDS) automatically responds to the CFDS high-radiation signal listed in Table 3.17-1 to mitigate a release of radioactivity.
- The balance-of-plant drain system (BPDS) automatically responds to the BPDS high-radiation signals listed in Table 3.17-1 to mitigate a release of radioactivity.

3.17.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.17-2 contains the inspections, tests, and analyses for radiation monitoring -- NuScale Power Modules 1 - 6.

Table 3.17-1: Radiation Monitoring - Automatic Actions for NuScale Power Modules 1 - 6

Variable Monitored	Actuated Component(s)	Component Action(s)
CFDS containment drain separator gaseous discharge to Reactor Building heating ventilation and air conditioning system	1. CFDS containment drain separator gaseous discharge isolation valve	1. Close
BPDS 0A condensate polishing system regeneration skid waste effluent	1. North chemical waste water sump pump A 2. North chemical waste water sump pump B 3. North chemical water sump to BPDS collection tank flow control valve 4. North chemical water sump to liquid radioactive waste system (LRWS) isolation valve	1. Stop 2. Stop 3. Close 4. Close
BPDS north turbine building floor drains	1. North waste water sump pump A 2. North waste water sump pump B 3. North waste water sump to BPDS collection tank flow control valve 4. North waste water sump to LRWS isolation valve	1. Stop 2. Stop 3. Close 4. Close
BPDS auxiliary blowdown cooler condensate	1. North waste water sump pump A 2. North waste water sump pump B 3. North waste water sump to BPDS collection tank flow control valve 4. North waste water sump to LRWS isolation valve	1. Stop 2. Stop 3. Close 4. Close

Table 3.17-2: Radiation Monitoring - Inspections, Tests, Analyses, and Acceptance Criteria for NuScale Power Modules 1-6

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The CFDS automatically responds to the CFDS high-radiation signal listed in Table 3.17-1 to mitigate a release of radioactivity.	A test will be performed of the CFDS high-radiation signal listed in Table 3.17-1.	Upon initiation of a real or simulated CFDS high-radiation signal listed in Table 3.17-1, the CFDS automatically aligns/actuates the identified components to the positions identified in the table.
2.	The BPDS automatically responds to the BPDS high-radiation signals listed in Table 3.17-1 to mitigate a release of radioactivity.	A test will be performed of the BPDS high-radiation signals listed in Table 3.17-1.	Upon initiation of the real or simulated BPDS high-radiation signals listed in Table 3.17-1the BPDS automatically aligns/actuates the identified components to the positions identified in the table.

3.18 Radiation Monitoring - NuScale Power Modules 7 - 12

3.18.1 Design Description

System Description

The scope of this section is automatic actions of various systems based on radiation monitoring. Automatic actions of systems based on radiation monitoring are nonsafety-related functions. The systems actuated by these automatic radiation monitoring functions are shared by NuScale Power Modules (NPMs) 7 through 12.

Design Commitments

- The containment flooding and drain system (CFDS) automatically responds to the CFDS high-radiation signal listed in Table 3.18-1 to mitigate a release of radioactivity.
- The balance-of-plant drain system (BPDS) automatically responds to the BPDS high-radiation signals listed in Table 3.18-1 to mitigate a release of radioactivity.

3.18.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.18-2 contains the inspections, tests, and analyses for radiation monitoring of NuScale Power Modules 7 - 12.

Table 3.18-1: Radiation Monitoring - Automatic Actions For NuScale Power Modules 7 - 12

Variable Monitored	Actuated Component(s)	Component Action(s)
CFDS containment drain separator gaseous discharge to Reactor Building heating ventilation and air conditioning system	1. CFDS containment drain separator gaseous discharge isolation valve	1. Close
BPDS 0B condensate polishing system regeneration skid waste effluent	1. South chemical waste water sump pump A 2. South chemical waste water sump pump B 3. South chemical water sump to BPDS collection tank flow control valve 4. South chemical water sump to liquid radioactive waste system (LWRS) isolation valve	1. Stop 2. Stop 3. Close 4. Close
BPDS south turbine building floor drains	1. South waste water sump pump A 2. South waste water sump pump B 3. South waste water sump to BPDS collection tank flow control valve 4. South waste water sump to liquid radioactive waste system isolation valve	1. Stop 2. Stop 3. Close 4. Close

Table 3.18-2: Radiation Monitoring Inspections, Tests, Analyses, and Acceptance Criteria For NuScale Power Modules 7 - 12

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.	The CFDS automatically responds to the CFDS high-radiation signal listed in Table 3.18-1 to mitigate a release of radioactivity.	A test will be performed of the CFDS high-radiation signal listed in Table 3.18-1.	Upon initiation of a real or simulated CFDS high-radiation signal listed in Table 3.18-1, the CFDS automatically aligns/actuates the identified components to the positions identified in the table.
2.	The BPDS automatically responds to the BPDS high-radiation signals listed in Table 3.18-1 to mitigate a release of radioactivity.	A test will be performed of the BPDS high-radiation signals listed in Table 3.18-1.	Upon initiation of the real or simulated BPDS high-radiation signals listed in Table 3.18-1, the BPDS automatically aligns/actuates the identified components to the positions identified in the table.

CHAPTER 4 INTERFACE REQUIREMENTS

4.0 Interface Requirements

As noted in10 CFR 52.47(a)(25), identification of the interface requirements is to be met by those portions of the plant for which the application does not seek certification. Also, 10 CFR 52.47(a)(26) requires justification that compliance with the interface requirements be verifiable through inspection, testing (either in the plant or elsewhere), or analysis. The method to be used for verification of interface requirements must be included as part of the proposed Inspections, Tests, Analyses, and Acceptance Criteria required by 10 CFR 52.47 (b)(1).

In additiion, 10 CFR 52.79(d)(2) requires that if the combined license application references a standard design certification, then the Final Safety Analysis Report must demonstrate that the interface requirements established for the design under § 52.47 have been met.

This section provides the Tier 1 material for interface items. No Tier 1 information is provided for the conceptual design portions that are combined license applicant scope.

4.1 Site-Specific Structures

Failure of any of the site-specific structures not within the scope of the NuScale Power Plant certified design will not cause any of the Seismic Category I structures within the scope of the NuScale Power Plant-certified design to fail.

CHAPTER 5 SITE PARAMETERS

5.0 Site Parameters

The NuScale Power Plant design certification may be deployed over a wide variety of sites; therefore, it is necessary to specify a set of parameters that bound the site conditions that are suitable for NuScale Power Plant operation. A site for construction of a NuScale Power Plant is acceptable if the site-specific characteristics fall within the site parameter values specified in Table 5.0-1 and Figure 5.0-1 through Figure 5.0-4. In case of deviation from these parameters, justification may be provided that the proposed facility is acceptable at the proposed site.

Table 5.0-1: Site Parameters

Site Characteristic	Site Parameter
Nearby Industrial, Transportation, and Military Facilities	
External hazards on plant structures, systems, and components (SSC) (e.g., explosions, fires, release of toxic chemicals and flammable clouds, pressure effects) on plant SSC	No external hazards
Aircraft hazards on plant SSC	No design basis aircraft hazards
Meteorology	
Maximum precipitation rate	19.4 inches per hour 6.3 inches for a 5 minute period
Normal roof snow load	50 psf
Extreme roof snow load	75 psf
100-year return period 3-second wind gust speed	145 mph (Exposure Category C) with an importance factor of 1.15 for Reactor Building, Control Building, and Radioactive Waste Building
Design Basis Tornado maximum wind speed translational speed maximum rotational speed radius of maximum rotational speed pressure drop rate of pressure drop	 230 mph 46 mph 184 mph 150 ft 1.2 psi 0.5 psi/sec
Tornado missile spectra	Table 2 of Regulatory Guide 1.76, Revision 1, Region 1.
Maximum wind speed design basis hurricane	290 mph
Hurricane missile spectra	Tables 1 and 2 of Regulatory Guide 1.221, Revision 0.
Zero percent exceedance value (historical limit excluding peaks <2 hours) Maximum outdoor design dry bulb temperature Minimum outdoor design dry bulb temperature	 115°F -40°F
Accident release χ/Q values at exclusion area boundary and outer boundary of low population zone 0-2 hr 2-8 hr 8-24 hr 24-96 hr 96-720 hr	 6.22E-04 s/m^3 5.27E-04 s/m^3 2.41E-04 s/m^3 2.51E-04 s/m^3 2.46E-04 s/m^3
Accident release χ/Q values at main control room/ technical support center door and heating ventilation and air conditioning intake 0-2 hr 2-8 hr 8-24 hr 1-4 day 4-30 day	Door — Heating Ventilation and Air Conditioning Intake 6.50E-03 s/m^3 — 6.50E-03 s/m^3 5.34E-03 s/m^3 — 5.34E-03 s/m^3 2.32E-03 s/m^3 — 2.32E-03 s/m^3 2.37E-03 s/m^3 — 2.37E-03 s/m^3 2.14E-03 s/m^3 — 2.14E-03 s/m^3
Hydrologic Engineering	
Maximum flood elevation Probable maximum flood and coincident wind wave and other effects on maximum flood level	 1 foot below the baseline plant elevation
Maximum elevation of groundwater	2 feet below the baseline plant elevation

Table 5.0-1: Site Parameters (Continued)

Site Characteristic	Site Parameter
Geology, Seismology, and Geotechnical Engineering	
Ground motion response spectra/safe shutdown earthquake	See Figure 5.0-1 and Figure 5.0-2 for horizontal and vertical certified seismic design response spectra (CSDRS) for all Seismic Category I SSC. See Figure 5.0-3 and Figure 5.0-4 for horizontal and vertical high frequency certified seismic design response spectra (CSDRS-HF) for Reactor Building and Control Building.
Fault displacement potential	No fault displacement potential
Minimum soil bearing capacity (Q_{ult}) beneath safety-related structures	75 ksf
Lateral soil variability	Uniform site (< 20 degree dip)
Minimum soil angle of internal friction	30 degrees
Minimum shear wave velocity	≥ 1000 fps at bottom of foundation
Maximum settlement for the Reactor Building, Control Building, and Radioactive Waste Building: • total settlement • tilt settlement • differential settlement (between Reactor Building and Control Building, and Reactor Building and Radioactive Waste Building)	 4 inches Maximum of 0.5 inch per 50 feet of building length or 1 inch total in any direction at any point in these structures 0.5 inch
Slope failure potential	No slope failure potential

Figure 5.0-1: NuScale Horizontal Certified Seismic Design Response Spectra 5% Damping

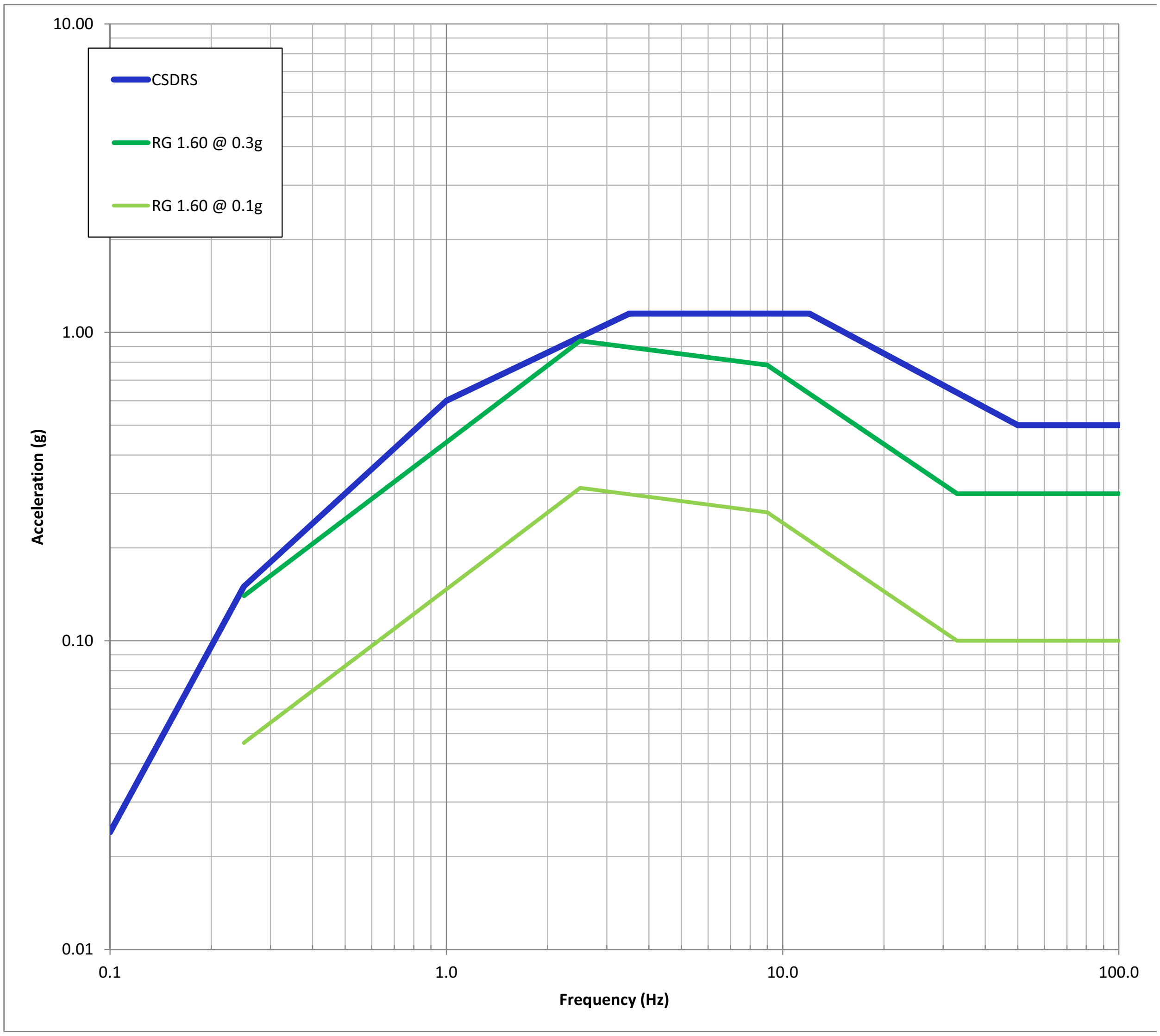


Figure 5.0-2: NuScale Vertical Certified Seismic Design Response Spectra 5% Damping

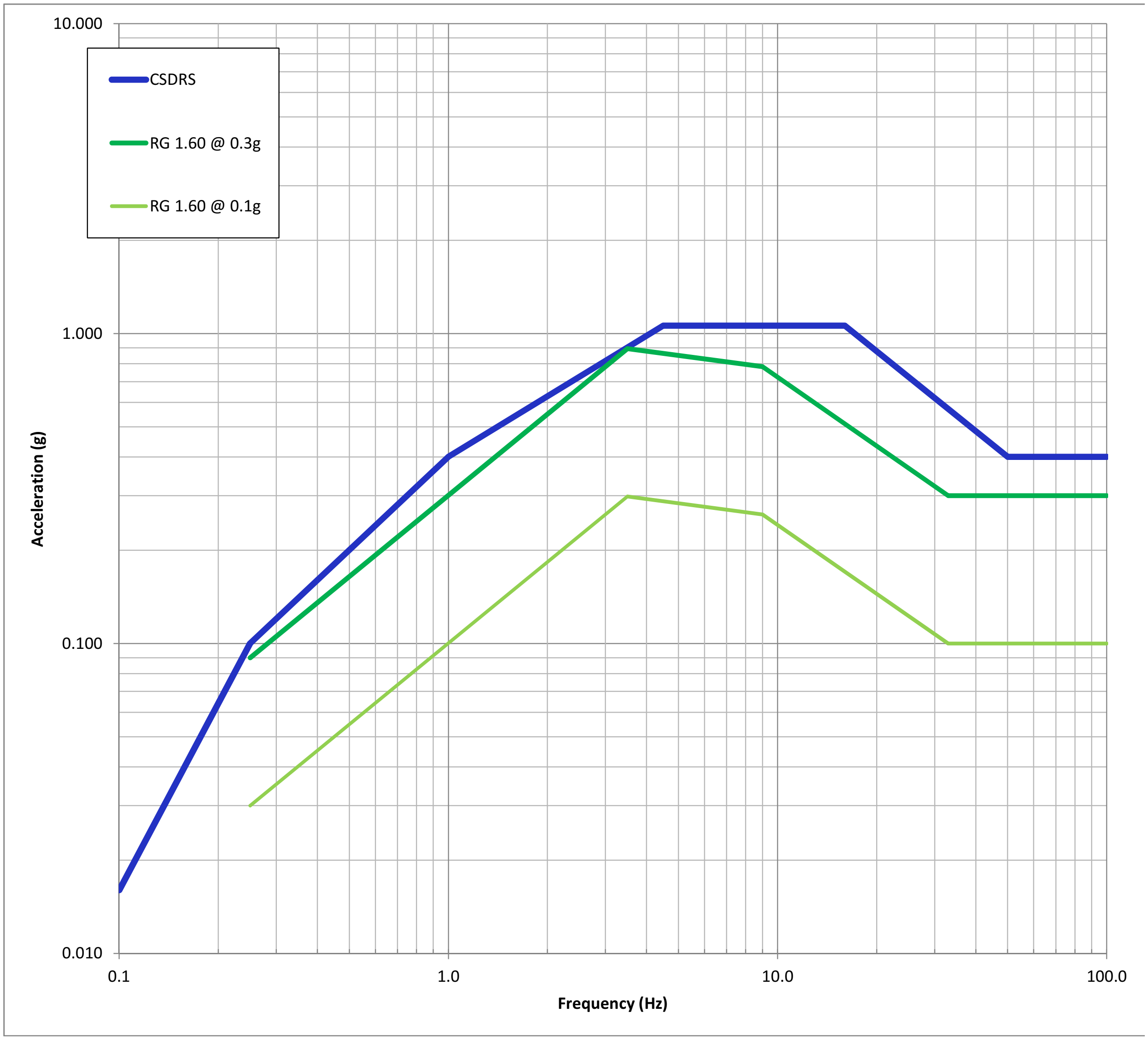
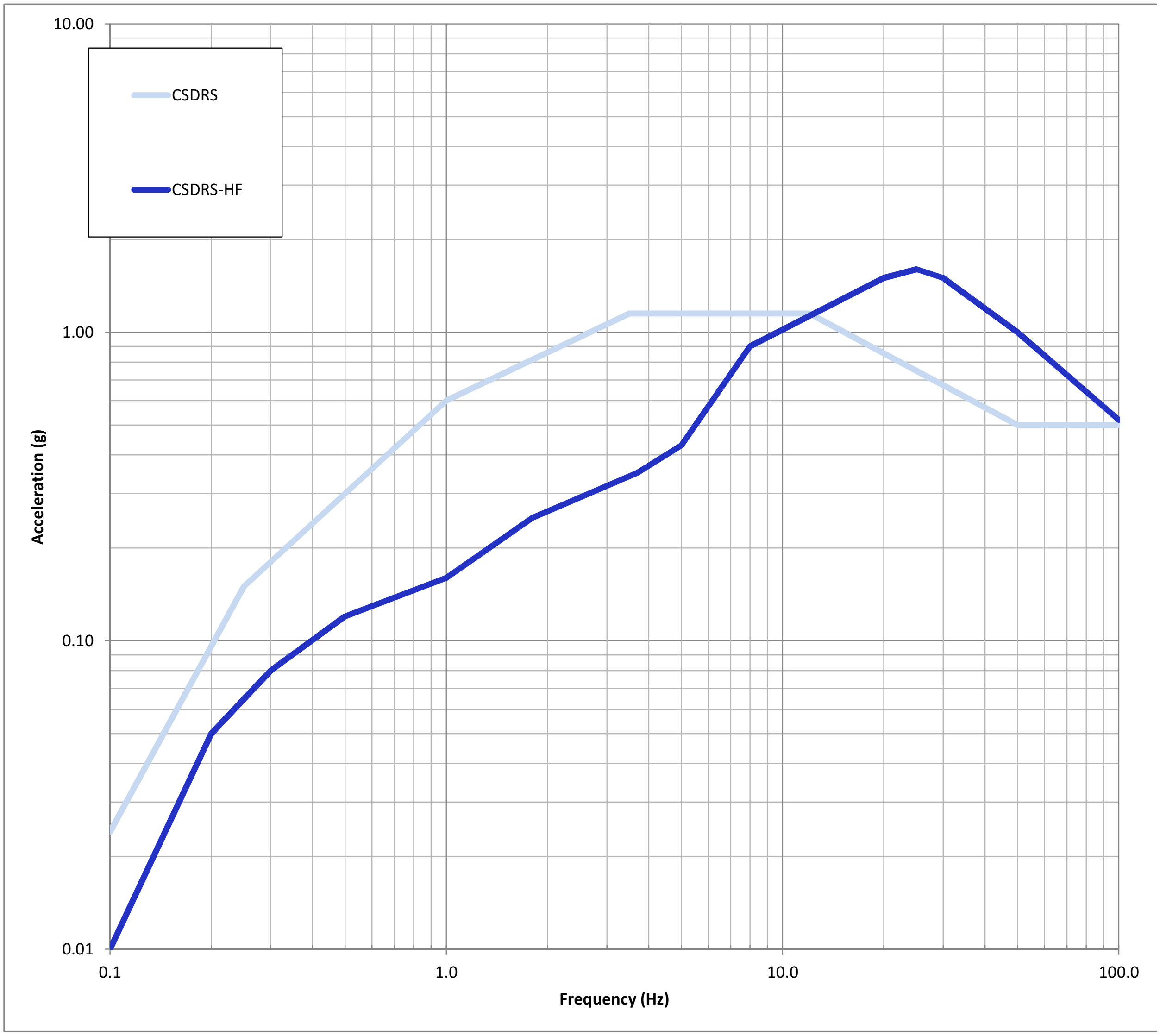
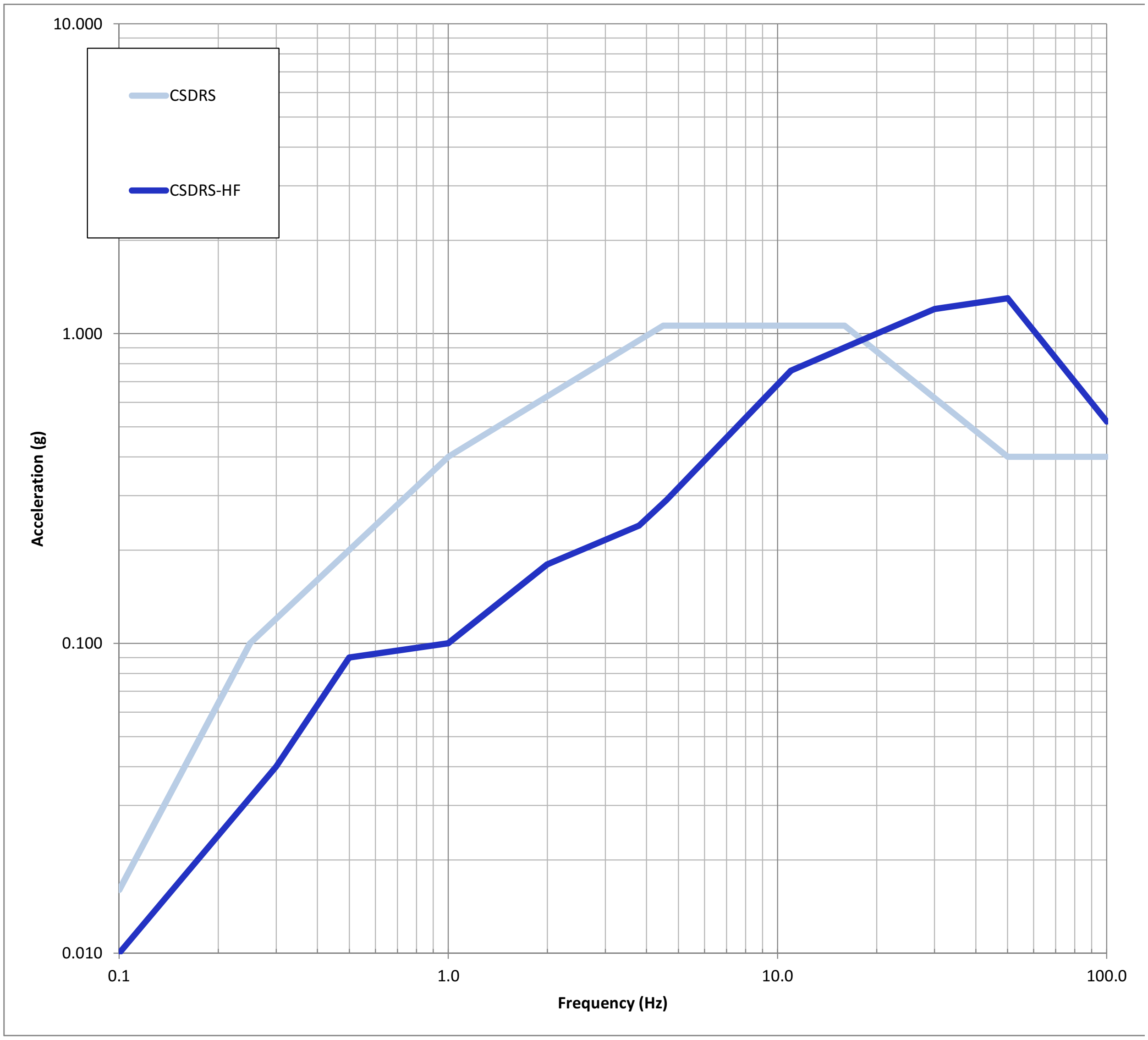


Figure 5.0-3: NuScale Horizontal Certified Seismic Design Response Spectra - High Frequency 5% Damping



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

Figure 5.0-4: NuScale Vertical Certified Seismic Design Response Spectra - High Frequency 5% Damping



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