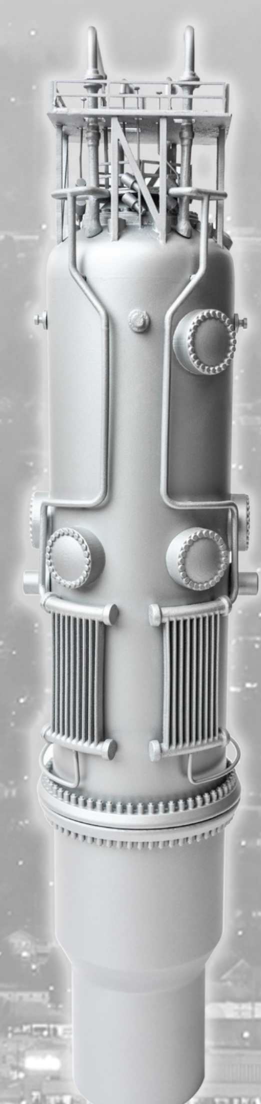
NuScale Standard Plant
Design Certification Application

Chapter Fourteen

Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria

PART 2 - TIER 2



Revision 4
January 2020

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CHAPTER 14 INITIAL TEST PROGRAM AND INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA

14.0 Verification Programs

Verification programs include the initial test programs for the NuScale Power, LLC (NuScale) Power Plant. The initial test programs are comprised of preoperational tests, initial fuel loading, initial criticality, low-power tests, and power-ascension tests. The verification programs ensure that the as-built facility configuration and operation comply with the approved plant design and applicable regulations.

The verification programs also include Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). The methodology associated with developing ITAAC is described in Section 14.3. The ITAAC are presented in Tier 1.

The initial test program addresses structures, systems, and components and design features for both the nuclear portion of the facility and the balance-of-plant. The initial test program contains information that:

- addresses the major phases of the test program including preoperational tests, initial fuel loading, initial criticality, low-power tests, and power-ascension tests, including scope and general plans for demonstrating that due consideration has been given to matters that normally require advance planning.
- demonstrates that an adequate number of qualified personnel support the program.
- demonstrates the adequacy of administrative controls to govern the conduct of the program.
- allows plant staff the ability to train using the plant's operating procedures.
- demonstrates and verifies the adequacy of plant operating and emergency procedures to the extent practicable during the period of the initial test program.
- allows for the verification of functional requirements.
- demonstrates sequence of testing such that the safety of the plant does not depend on untested structures, systems, and components.

14.1 Specific Information to be Addressed for the Initial Plant Test Program

The initial test program establishes procedures and controls used to conduct and evaluate the results of tests as described in Section 14.2 and to satisfy the relevant requirements of the following regulations:

- 10 CFR 30.53, as it relates to testing radiation detection equipment and monitoring instruments
- 10 CFR 50.34(b)(6)(iii), as it relates to providing information associated with preoperational testing and initial operations
- Section XI of Appendix B to 10 CFR Part 50, as it relates to test programs to demonstrate that systems, structures, and components will perform satisfactorily
- Option A or Option B of Appendix J of 10 CFR Part 50, as it relates to preoperational leakage rate testing
- 10 CFR 52.79 as it pertains to preoperational testing and initial operations
- Subpart A, Subpart B, and Subpart C of 10 CFR Part 52 as they relate to the Inspections, Tests, Analyses, and Acceptance Criteria that the applicant must submit

14.2 Initial Plant Test Program

14.2.1 Summary of Initial Test Program and Objectives

The Initial Test Program (ITP) consists of a series of preoperational and startup tests conducted by the Startup organization. Preoperational testing is conducted for each NuScale Power Module (NPM) following completion of construction testing but prior to fuel load. Completion of preoperational testing for each NPM is necessary to ensure the NPM is ready for fuel loading and startup testing.

Startup tests of an NPM are performed following the completion of preoperational testing. Startup testing includes the following:

- initial fuel loading and pre-critical testing
- initial criticality testing
- low-power testing
- power-ascension testing

Startup testing is performed to confirm the design bases of the NPM and to demonstrate, to the extent practical, that the NPM will operate in accordance with its design and is capable of responding to anticipated transients and postulated accidents as described in Section 15.0.

The objectives of the ITP are to

- provide assurance that structures, systems, and components (SSC) operate in accordance with their design.
- provide assurance that construction and installation of equipment in the facility has been completed in accordance with the design.
- demonstrate to the extent practical the validity of analytical models used to predict plant responses to anticipated transients and postulated accidents, and to demonstrate to the extent practical the correctness and conservatism of assumptions used in those models.
- familiarize the plant's operating and technical staff with the operation of the facility.
- perform testing to the extent practical using the plant conditions that simulate the actual operating, abnormal operating occurrences, and emergency conditions to which the SSC may be subjected.
- verify to the extent practical by trial use that the facility operating procedures, surveillance procedures and emergency procedures are adequate.
- verify that interfaces and system and component interactions are in accordance with the design.
- complete and document the ITP testing required to satisfy preoperational and startup testing requirements and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) testing requirements.

Preoperational and startup testing is performed on those SSC that are:

- relied upon for safe shutdown and cooldown of the NPM under normal conditions and for maintaining a safe condition for an extended shutdown period.
- relied upon for safe shutdown and cooldown of the NPM under transient and postulated accident conditions and for maintaining a safe condition for an extended shutdown period following such conditions.
- relied upon for establishing conformance with safety limits or limiting conditions for operation that are included in the technical specifications (TS).
- assumed to function or for which credit is taken in the accident analysis as described in Chapter 15.
- used to process, store, control, or limit the release of radioactive materials.
- relied upon to maintain their structural integrity during normal operation, anticipated transients, simulated test parameters, and design basis event conditions to avoid damage to safety-related SSC.
- identified as risk-significant in the probabilistic risk assessment.

The ITP is implemented consistent with the requirements of Section XI of 10 CFR 50 Appendix B. Implementation of the ITP ensures that the testing required to demonstrate that SSC perform satisfactorily in service, are identified and performed in accordance with written test procedures that incorporate the requirements and acceptance limits in the applicable design documents.

Leakage rate testing of the NPM and related systems and components penetrating the containment pressure boundary is described in Section 6.2. Leakage rate testing test abstracts are presented in Section 14.2.12.

The methodology associated with the development of the ITAAC necessary to demonstrate that the facility has been constructed and will be operated in conformity with the final safety analysis report and the applicable Nuclear Regulatory Commission (NRC) regulations is presented in Section 14.3.

NuScale Power Plant compliance with the proposed technical resolution of unresolved safety issues and medium- and high-priority generic safety issues that are identified in NUREG-0933 are addressed in Section 1.9.3. Operating experience insights are addressed in Section 1.9.4 and Section 14.2.8. Compliance with technically relevant portions of the Three Mile Island requirements are addressed in Section 1.9.5.

14.2.1.1 Construction Organization Testing

The objective of construction testing is to verify, on a system basis, that the system is constructed and installed in accordance with design requirements. Construction tests include but are not limited to:

- flushing
- cleaning
- hydrostatic pressure tests

- wiring continuity and separation checks

The testing and installation of digital I&C systems is described in FSAR Section 7.2.1, and includes factory acceptance testing and site acceptance testing which is completed as part of construction and installation tests performed prior to, and as a prerequisite of, preoperational tests. Factory acceptance tests are performed during the digital I&C system testing phase described in FSAR Section 7.2.1. Site installation and checkout activities are performed as part of the integrated site acceptance testing during the system installation phase as described in FSAR Section 7.2.1. Software integration and testing is governed by the NuScale Digital I&C Software Master Test Plan described in FSAR Section 7.2.1.

14.2.1.2 Preoperational Test Phase Objectives

Preoperational tests are performed to demonstrate that SSC operate in accordance with design requirements so that initial fuel loading, initial criticality, and subsequent power operation can be safely undertaken. The objectives of the preoperational test phase are to

- demonstrate that SSC will perform their functions in accordance with their design during the preoperational test phase.
- verify and demonstrate expected operation following a loss of power sources and in degraded modes for which the systems are designed to remain operational.
- test the backup power supply system (BPSS) to ensure that backup sources of alternating current (AC) electrical power are available when the normal AC power sources are not available.
- verify and demonstrate the operational readiness of valves and dynamic restraints before relying on those components to perform their safety functions.
- perform inspections or testing for flow-induced vibration loads on components that must maintain their structural integrity.
- obtain baseline test and operating data on equipment and systems for future reference.
- operate equipment for a sufficient period of time to achieve normal equilibrium conditions (e.g., temperatures and pressures) so that design, manufacturing, and installation defects can be detected and corrected.
- ensure to the extent practical plant systems operate properly on an integrated basis.
- evaluate normal, abnormal, and emergency operating procedures to the extent practical.
- demonstrate equipment performance.
- test, as appropriate, manual operation and automatic operation of systems and their components.
- test the proper functioning of controls, permissives, interlocks, and equipment protective devices for which malfunction or premature actuation may shut down or defeat the operation of systems or equipment.

- provide the plant operating staff with the opportunity to obtain practical experience in the operation and maintenance of equipment and systems including instrument calibrations and functional tests of components.
- demonstrate equipment performance is satisfactory to proceed to initial fuel loading and initial criticality.

Test abstracts associated with preoperational testing are included in Section 14.2.12.

14.2.1.3 Startup Test Phase Objectives

14.2.1.3.1 Initial Fuel Loading and Pre-Critical Tests

This phase of testing is performed in order to ensure that initial fuel loading of an NPM can be accomplished in an orderly and safe manner. A description of the fuel loading process is presented Section 14.2.10. The objectives of the initial fuel loading and pre-critical tests are to:

- conduct initial fuel loading cautiously to preclude inadvertent criticality. Establish and follow specific safety measures, such as:
  - ensuring that the applicable TS requirements and other prerequisites have been satisfied
  - continuous monitoring of the neutron flux throughout core loading so that changes in the multiplication factor are observed
  - verifying that the fuel and control components have been properly installed
- establish that the required SDM exists, without achieving criticality
- establish the functionality of plant systems and components, including reactivity control systems and other systems and components necessary to ensure the safety of plant personnel and the public in the event of errors or malfunctions
- confirm the proper operation of plant systems and design features that could not be completely tested during preoperational testing
- confirm interdependent effects among the safety features of the design are acceptable

14.2.1.3.2 Initial Criticality

The objectives associated with the initial criticality phase of the startup testing program are to achieve initial criticality in a safe and controlled manner. In order to meet this objective the following are performed:

- The initial approach to criticality is performed in a deliberate and orderly manner using the same rod withdrawal sequences and patterns that will be used during subsequent startups.
- The neutron flux levels are continuously monitored and periodically evaluated. A neutron count rate of at least 1/2 counts per second should register on the

startup channels before startup begins, and the signal to noise ratio should be known to be greater than 2.

- The systems required for startup or protection of the plant, including the reactor protection system and engineered safety features (ESFs), are operable and in a state of readiness.
- The control rod or poison removal sequence is accomplished using approved plant procedures.
- The reactor achieves initial criticality by boron dilution. Control rods are withdrawn before dilution begins.
- The control rod insertion limits defined in the Technical Specifications are observed and followed.
- Criticality predictions for boron concentration and control rod positions are provided.
- The reactivity addition sequence is prescribed, and plant procedures require a cautious approach to achieving criticality to prevent passing through criticality in a period shorter than approximately 30 seconds (<1 decade per minute).

A description of the process followed to achieve initial criticality is provided in Section 14.2.10.

14.2.1.3.3 Low - Power Testing

Following criticality, low-power testing is performed. The objectives associated with performing low-power testing are to

- confirm the design and validate analytical models.
- verify the correctness of assumptions used in the safety analyses.
- confirm the functionality of plant systems and design features that could not be completely tested during the preoperational test phase because of the lack of an adequate heat source for the reactor coolant and main steam systems.

14.2.1.3.4 Power-Ascension Testing

Following low-power testing, power-ascension testing is performed. Power-ascension testing is performed to bring the reactor to full power with testing at power levels of approximately 25 percent, 50 percent, 75 percent, and 100 percent. The objectives associated with performing power-ascension testing are to

- achieve reactor full power in a safe and controlled manner.
- demonstrate that the plant operates in accordance with its design bases during normal steady-state conditions and, to the extent practical, during and following anticipated transients.
- validate models used to predict plant response.
- demonstrate the ability of major or principal plant control systems to automatically control process variables within design limits.

- demonstrate that the facility's integrated dynamic response is in accordance with design for plant events such as reactor scram, turbine trip, and loss of feedwater heaters or pumps.

14.2.2 Organization and Staffing

COL Item 14.2-1: A COL applicant that references the NuScale Power Plant design certification will describe the site-specific organizations that manage, supervise, or execute the Initial Test Program, including the associated training requirements.

14.2.3 Test Procedures

14.2.3.1 Initial Test Program Procedures

Test procedures are developed and reviewed by individuals with the appropriate technical background and expertise. Once the test procedures have been developed they are reviewed by plant management personnel who upon acceptance designate the procedures as final.

Input from the principal design organization is utilized to establish the test objectives and acceptance criteria for the system. Operating experience, as discussed in Section 14.2.8 is used in the development of test procedures.

Test procedure testing and acceptance criteria are founded upon the information contained in design specifications, design documents, the Final Safety Analysis Report, and regulatory documents. A test procedure is prepared for each specific system test to be performed during the test program.

Preoperational and startup testing procedures include checklists and signature blocks to control the sequence and performance of testing. The administrative controls associated with test procedure development address the following:

- test procedure format
- application, to the extent practical, of normal plant operating procedures, emergency operating procedures, and surveillance procedures in support of test procedure development
- test procedure review and approval
- test procedure change and revision

The content of each test procedure addresses

- objectives.
- detailed step-by-step procedures specifying how testing is to be performed.
- special precautions.
- test instrumentation.
- test equipment calibration.

- initial test conditions, including provisions to perform testing under environmental conditions as close as practical to those the equipment will experience in both normal and accident situations.
- methods to direct and control test performance.
- acceptance criteria by which testing is evaluated. Acceptance criteria account for measurement errors and uncertainties associated with normal operation as well as operation during transients and accidents. Acceptance criteria are biased conservatively. In some cases the acceptance criteria is qualitative. Where applicable, quantitative values, with appropriate tolerances, are used as acceptance criteria.
- test prerequisites including as necessary prerequisite statements to ensure that nonstandard arrangements are restored to their normal status after the test is completed (for example, electric jumper cable use does not invalidate electrical separation; jumper cables are removed following testing; valve configurations, and instrument settings are returned to their normal orientations and settings).
- identification of the data to be collected and the method of documentation.
- actions to take if unanticipated errors or malfunctions occur while testing.
- remedial actions to take if acceptance criteria are not satisfied.
- actions to take if an unexpected or unanalyzed condition occurs.

14.2.3.2 Graded Approach to Testing

The ITP allows for the application of a graded approach to testing. The graded approach to testing is founded in the requirements of General Design Criterion 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 that requires, in part, that SSC important to safety shall be tested to quality standards commensurate with the importance of the safety functions to be performed. Criterion XI of Appendix B to 10 CFR Part 50 also includes a graded approach for important to safety SSC in the Quality Assurance Program. The administrative requirements that govern the conduct of the test program (e.g., test program objectives, organizational elements, personnel qualifications, evaluation and approval of test results, and test records retention) contain provisions that allow for testing of SSC in a manner commensurate with the safety significance of the SSC within its scope. This provides a systematic approach to the "defense-in-depth" concept. This concept dictates that the plant be designed, constructed, and tested to (1) provide for safe normal operation, (2) ensure that, in the event of errors, malfunctions, and off-normal conditions, the reactor protection systems and other design features will mitigate the event or limit its consequences to defined and acceptable levels, and (3) ensure that adequate safety margin exists for events of extremely low probability or arbitrarily postulated hypothetical events without substantial reduction in the safety margin for the protection of public health and safety.

Application of the graded approach to testing provides reasonable assurance that the SSC being tested will perform satisfactorily while accomplishing the testing in a cost-effective manner. The administrative requirements that govern the conduct of the test program allow for the preparation of documentation (such as procedures and records) associated with testing to be prepared commensurate with the importance to safety of the SSC being tested.

During the SSC classification process, the subject matter expert identified all functions of the system. Each of these functions was compared to safety functional requirements and regulatory functional requirements to establish a functional hierarchy. This hierarchy established a supporting to supported relationship between the systems and tied it to a set of plant functions as described in Section 17.4 to identify a classification for the functions. The functions were categorized as A1 (safety-related, risk-significant), A2 (safety-related, not risk-significant), B1 (nonsafety-related, risk-significant), or B2 (nonsafety-related, not risk-significant). This safety significance evaluation was the basis for the graded approach in the ITP.

The hierarchy in the NuScale approach to preoperational testing is:

- Testing of active, safety-related system functions (A1 or A2 functions)
- Testing of active, nonsafety-related functions which require ITAAC verification (B1 and B2)
- Testing of active nonsafety-related functions which do not require ITAAC verification (B1 and B2)

The preoperational test abstracts contained in Table 14.2-1 through Table 14.2-69 provide a definition of the test scope for each system by listing the associated active system functions and their safety categorization. The test abstract also provides system functions tested by another test abstract, thereby providing an "inventory" of all testable system functions.

Table 17.4-1 contains a list of all A1 and B1 system functions. All active, safety-related A1 functions are tested by the safety-related module protection system (MPS) logic testing found in Table 14.2-63. The remaining safety-related functions categorized as A2 are also tested by the MPS test abstract. The NuScale graded approach provides for testing of A2 functions to the same rigor as A1 functions.

As indicated by Table 14.2-63, all active, safety-related functions are one of the following types:

- provides safety-related instrument information signals to MPS
- removes electrical power to the control rod drive
- removes electrical power to the pressurizer heaters
- removes electrical power to the trip solenoids of safety-related valves
- closes safety-related valves

The MPS test abstract also describes testing of the following safety-related design features:

- Safety-related containment isolation valve response time
- MPS safety-related sensor response time

Section 14.3 provides guidance regarding the development of certified design material (CDM) in Tier 1, including Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) required under Title 10 of the Code of Federal Regulations (10 CFR) 52.47(b)(1). The scope of ITAAC is sufficient to provide reasonable assurance that, if the ITAAC are successfully completed, the facility has been constructed and can be operated in accordance with the Atomic Energy Act, relevant Nuclear Regulatory Commission (NRC) regulations, and the combined license (COL). The successful completion of ITAAC constitutes the basis for the NRC determination to allow operation of a facility certified under 10 CFR 52. ITAAC may be verified by an inspection, test, or analysis, or a combination thereof. Some ITAAC are verified by successful completion of preoperational testing.

Table 14.3-1 and Table 14.3-2 identify each ITAAC by its unique ITAAC number, for example, ITAAC 03.01.02. The two tables provide a discussion of each ITAAC, including a reference to a verifying preoperational test abstract, if required. This results in a cross-reference between the Tier 1 ITAAC number and the associated Tier 2 test abstract.

If an ITAAC is verified by the successful completion of a preoperational test, then Table 14.3-1 or Table 14.3-2 identifies the associated preoperational test abstract number contained in Section 14.2. The acceptance criteria of the associated test in Section 14.2 contain a bracketed reference to the verified ITAAC. An example annotation is [ITAAC 03.01.02] in Table 14.2-18 where 03.01.02 is the number of the verified ITAAC contained in Tier 1 Table 3-1 item 2.

Preoperational testing of nonsafety-related systems is necessary to verify ITAAC for the following design features. The acceptance criteria of the associated test abstract acceptance criteria are annotated with the ITAAC number.

- Radiation isolation
- Battery room ventilation for hydrogen control
- Control room building and reactor building differential pressure
- Control room envelope design features
- Post-accident monitoring (PAM) signals
- Fire protection pump flow
- Plant lighting illumination in the main control room and for post-fire shutdown
- Important Human Actions for containment flooding and drain system (CFDS) addition of water to containment
- Important Human Actions for chemical and volume control system (CVCS) addition of water to the reactor coolant system (RCS)

Credit is taken for the logic testing performed for the nonsafety-related module control system (MCS) described in Section 7.0.4, and the nonsafety-related plant control system (PCS) described in Section 7.0.4. Therefore, if the component is controlled by MCS or PCS, the component-level logic testing in the preoperational test is limited to the testing of component-level design features described below (if the design feature is applicable to the system) unless the preoperational test verifies an ITAAC. The component tests are standardized to provide the same level of test detail across all systems. This graded approach does not affect system-level tests which require integrated system operation. The standardized component tests are:

- Remote operation of equipment.
- Manual control of variable-speed pump or fan.
- Automatic start of standby pump or fan.
- Automatic operation of pump recirculation valve.
- Pump start does not create a water hammer.
- Remote operation of valve or damper.
- Valve or damper fails to its safe position on loss of air.
- Valve or damper fails to its safe position on loss of electrical power to its solenoid.
- Damper or fan responds to fire or smoke alarm.
- Equipment response to automatic signals to protect plant equipment.
- Automatic operation of tank or basin level control valve.
- Local grab sample can be obtained from a system grab sample device.
- Automatic bus transfer via bus tie breaker.
- System instrument calibration.
- Each instrument is monitored in the MCR and the remote shutdown station (RSS), if the signal is designed to be displayed in the RSS. (Test not required if the instrument calibration verified the MCR and RSS display.)

14.2.3.3 Testing of First-of-a-Kind Design Features

First-of-a-kind (FOAK) tests are new, unique, or special tests used to verify design features that are being reviewed for the first time by the NRC. The NuScale Power Plant contains design features which are new and unique and have not been tested previously; therefore, testing of these design features is treated as FOAK. For the FOAK tests, the testing frequency is specified in the test abstract. The NuScale comprehensive vibration assessment program (CVAP) is a FOAK program. The program is implemented consistent with the requirements of the “NuScale Comprehensive Vibration Assessment Program Technical Report", TR-0716-50439, and the “NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report,” TR-0918-60894. The CVAP is addressed in Section 3.9.2.

The following ITP test abstracts describe the on-site CVAP testing of FOAK design features:

- Table 14.2-72: Steam Generator Flow-Induced Vibration Test #72.
- Table 14.2-108: NuScale Power Module Vibration Test #108

The test results for the CVAP program testing of the first NPM are to inform the required CVAP testing on subsequent NPMs as described in Section 6.0 of TR-0716-50439. All other ITP testing of FOAK design features is performed for each NPM, except as described below.

Table 14.2-47: Emergency Core Cooling Test #47 includes a one-time in-situ system performance test of the emergency core cooling system (ECCS). The test demonstrates valve and containment response to manual emergency safety feature actuation of the ECCS at hot functional test pressure and temperature.

Section 5.4.3.4 contains a description of the decay heat removal system (DHRS) one-time in-situ RCS heat removal test. The test will be performed per test abstract Table 14.2-48: Decay Heat Removal System Test # 48.

Table 14.2-110 provides a summary of the ITP testing (i.e., preoperational and startup testing) for new design features. Each test will be performed for all NPMs.

Section 1.5.1 contains a description of testing programs which have been completed or are currently in progress for NuScale design features for which applicable data or operational experience did not previously exist. The section describes tests specific to fuel design, steam generator (SG) and control rod assemblies.

14.2.3.4 Generic Component Testing

Component testing is generally executed after a system's transfer from the construction organization to the startup organization. Generic component testing executes standardized tests for a family of related component types, independent of the component's system assignment. Each generic component test procedure will be completed and approved before the component is required as a prerequisite to a preoperational test performance. The completion of generic component testing will be listed as a prerequisite in each preoperational test procedure as applicable.

Examples of components that may require generic component testing are as follows:

- Mechanical Components
  - pumps
  - motors
  - chillers
  - compressors
- Valves
  - pressure relief valves
  - air operated valves
  - motor operated valves
  - hydraulically operated valves
  - control valves
- Electrical distribution components
  - initiating devices

- trip devices
- heaters
- breakers
- motor controllers
- switchgear
- transformers
- cables
- batteries
- battery chargers
- transfer devices
- converters
- inverters
- protective devices

- HVAC components
  - fans
  - dampers
  - fan coil units
  - high-efficiency particulate air filters
  - charcoal absorbers
- Instrument calibrations
- Other component types
  - heaters
  - heat tracing
  - cathodic protection

14.2.4 Conduct of the Test Program

The ITP activities are controlled by administrative procedures contained within the Startup Administrative Manual.

COL Item 14.2-2: A COL applicant that references the NuScale Power Plant design certification is responsible for the development of the Startup Administration Manual that will contain the administrative procedures and requirements that control the activities associated with the Initial Test Program. The COL applicant will provide a milestone for completing the Startup Administrative Manual and making it available for NRC inspection.

Administrative controls are established to ensure that the designated construction-related inspections and tests are completed prior to initiating preoperational testing. In addition controls are established to ensure completion of preoperational testing prior to initiating startup testing. Administrative controls address adherence to approved test procedures during the conduct of the test program and the methods for effecting changes to approved test procedures.

The controls used to ensure that test prerequisites associated with each major phase of testing, as well as individual system or component testing are met, include requirements for performing inspections and checks, identification of test personnel, completing data forms or check sheets, and identification of dates of completion.

The controls provided to implement plant modification and repairs ensure that the required modifications and repairs are made. Retesting is conducted following modifications or repairs. Reviews of proposed facility modifications by designated design organizations is conducted prior to performing the modification or repair.

Controls are established to ensure that retesting that is required for modifications or maintenance remains in compliance with ITAAC commitments.

The documentation associated with the conduct of the test plan is captured and auditable.

14.2.5 Review, Evaluation, and Approval of Test Results

Administrative procedures control the review and approval of preoperational and startup test results for each phase of the test program. This includes approval of test data for each major test phase before proceeding to the next test phase as well as approval of test data at each power test plateau (during the power-ascension phase) before increasing the power level. Test exceptions or results that do not meet acceptance criteria are identified to the responsible design organization as well as plant operations and plant technical staff and corrective actions and retests, as required, are performed.

These administrative procedures address the following:

- notification of responsible design organizations when test acceptance criteria are not met
- methods and schedules for approval of test data for each major phase
- methods used for initial review of individual parts of multiple tests
- technical evaluation of test results by qualified personnel and approval of such results by personnel in designated management positions
- provisions to allow design organizations to participate in the resolution of design-related problems that result in, or contribute to, a failure to meet test acceptance criteria
- provisions to retain test reports, including test procedures and results, as part of the plant historical records

14.2.6 Test Records

Initial test program reports, test procedures and results are retained as part of the plant's historical record in accordance with 10 CFR 50.36, "Technical Specification," 10 CFR 50.71, "Maintenance of Records, Making of Reports," and 10 CFR 50, Appendix B, Criterion XVII, "Quality Assurance Records." The test reports include test results associated with the testing of SSC identified in the ITP. A summary of the startup testing is included in a startup report. This summary includes the following information:

- description of the method and objectives for each test
- comparison of applicable test data with the related acceptance criteria, including the systems' responses to major plant transients (such as reactor scram and turbine trip)
- design and construction related deficiencies discovered during testing, and system modifications, the corrective actions required to correct those deficiencies, and the schedule for implementing the identified modifications and corrective actions
- justification for acceptance of systems or components that are not in conformance with design predictions or performance requirements
- conclusions about system or component adequacy
- identity of test observers and recorders
- type of observation
- identifying numbers of test or measuring equipment
- results of tests

14.2.7 Test Programs Conformance with Regulatory Guides

The ITP conforms to Regulatory Guide (RG) 1.68 Initial Test Programs for Water-Cooled Nuclear Power Plants, Rev. 4 except for aspects that address specific SSC design features not in the design.

The following list of regulatory guides provides information used to supplement the information, recommendations, and guidance presented in RG 1.68 Rev. 4 relative to testing of SSC:

- RG 1.20 - Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing, Rev. 3
- RG 1.29 - Seismic Design Classification for Nuclear Power Plants, Rev. 5
- RG 1.41 - Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments, March 1973
- RG 1.45 - Guidance on Monitoring and Responding to Reactor Coolant System Leakage, Rev. 1
- RG 1.68.1 - Initial Test Program of Condensate and Feedwater Systems for Light-Water Reactors, Rev. 2
- RG 1.68.2 - Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water- Cooled Nuclear Power Plants, Rev 2
- RG 1.68.3 - Preoperational Testing of Instrument and Control Air Systems, Rev.1
- RG 1.69 - Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants, Rev. 1

- RG 1.79 - Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors, Rev. 2
- RG 1.118 - Periodic Testing of Electric Power and Protection Systems, Rev. 3
- RG 1.128 - Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants Rev. 2
- RG 1.140 - Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water- Cooled Nuclear Power Plants, Rev. 2
- RG 1.155 - Station Blackout, Rev.1
- RG 8.38 - Control of Access to High and Very High Radiation Areas of Nuclear Power Plants, Rev. 1

14.2.8 Utilization of Reactor Operating and Testing Experience in Test Program Development

The operational experience gained from pressurized-water and other reactor designs is factored into the design and testing.

Operations and technical staff review the following documents for information that can be included in the ITP:

- NRC licensee event reports
- NRC generic communications (i.e., inspection and enforcement bulletins, circulars, generic letters, administrative letters, information notices, and regulatory issue summaries)
- Institute of Nuclear Power Operations issuances

The administrative procedures control the review of reactor operating experience and its incorporation in the ITP.

14.2.9 Trial Use of Plant Operating Procedures, Emergency Procedures, and Surveillance Procedures

Plant emergency, operating, and surveillance test procedures are, to the extent practical, developed, trial tested, and corrected during the ITP before fuel load to establish their adequacy. Trial testing of procedures is accomplished by having plant operators trained to these procedures to the extent practicable during the ITP. Following completion of trial testing these procedures are used as part of the ITP.

Additionally, emergency, operating, and surveillance test procedures are incorporated into the plant reference simulator, which meets the requirements of 10 CFR 55.46(c), and are trial tested as part of the operator training program.

The administrative procedures control the trial use of approved plant operating procedures, emergency operating procedures, and surveillance procedures.

14.2.10 Initial Fuel Loading, and Initial Criticality

Approved startup tests are used to control startup testing for initial fuel loading, pre-critical tests, initial criticality, low-power tests, and power-ascension tests in a controlled, deliberate, and safe manner. Technical specification compliance is met prior to initiation of startup testing. Startup test procedures are prepared based upon test abstracts provided in Section 14.2.12.

Startup tests procedures contain general provisions, precautions, prerequisites, and measures consistent with the requirements of RG 1.68 Rev. 4.

14.2.10.1 Initial Fuel Loading and Pre-Criticality Testing

As part of the startup test program, initial fuel loading and pre-criticality testing are performed by first implementing the prerequisite and precautionary measures that are contained in test procedures and identified below:

- Technical Specification (TS) compliance is met
- successful completion of all ITAAC
- actions to be taken in the event of unanticipated errors or malfunctions are clearly identified
- completion of a review of preoperational test results (the Startup Administrative Manual contains administrative procedures to control the verification process for successful completion of preoperational tests required for fuel load)
- review and status of design changes
- review of retests that were performed due to preoperational test deficiencies
- review of test exceptions

14.2.10.2 Initial Fuel Loading

Initial fuel loading is conducted to preclude inadvertent criticality. Specific safety measures are followed including (1) ensuring that the applicable TS are met, (2) performing continuous monitoring of the neutron flux throughout core loading so that changes in the multiplication factor are observed, (3) establishing requirements for periodic data taking, and (4) independently verifying that the fuel and control components have been properly installed.

Predictions of core reactivity are prepared in advance of the initial fuel loading to aid in evaluating the measured responses to specified loading increments. Comparative data on neutron detector responses from previous loadings of essentially identical core designs may be used in lieu of these predictions. Criteria and requirements for actions to be taken if the measured results deviate from expected values are established prior to the initial fuel loading. In addition, prior to initial fuel loading the required shutdown margin (SDM) is confirmed.

To provide further assurance of safe loading, requirements for the functionality of plant systems and components are established, including reactivity control systems and other systems and components necessary to ensure the safety of plant personnel and the public in the event of errors or malfunctions. The initial core loading is directly supervised by a senior licensed operator having no other concurrent duties, and the loading operation is conducted in strict accordance with detailed approved procedures.

14.2.10.3 Initial Criticality Testing

Control rods are withdrawn in the normal sequence to a configuration that does not violate the zero power rod insertion limits. Initial criticality is then achieved in a deliberate, orderly, and controlled fashion using boron dilution. Core neutron flux is continuously monitored during the approach to critical. Changes in reactivity are continuously monitored, and inverse multiplication plots are maintained and interpreted.

The following conditions exist prior to initial criticality:

- A minimum crew is required to support initial criticality, including a senior reactor operator with no other concurrent duties who is in charge of the operation.
- Critical rod position and boron concentration predictions are identified so that anomalies can be noted and evaluated.
- Systems needed for startup are aligned and in proper operation.
- Emergency systems are operable and in readiness.
- TS compliance is met.
- Nuclear instruments are calibrated.
- Neutron count rate of at least 1/2 counts per second registers on startup channels before the startup begins.
- Signal to noise ratio is greater than two.
- Conservative startup rate limit (greater than approximately a 30-second period) is established.
- High flux scram trips are set at their lowest value.
- Implementation of the radiation monitoring program as it pertains to operation of radiation barriers, airborne radiation monitors, air sampling, as well as performance of baseline surveys before pulling control rods for the approach to critical.

14.2.10.4 Low-Power Testing

Following initial criticality, low-power tests (at less than 5 percent power) are conducted to (1) confirm the design and, to the extent practical, validate the analytical models, and verify the correctness or conservatism of assumptions used in the safety analyses for the facility, and (2) confirm the functionality of plant systems and design features that could not be completely tested during the preoperational test phase because of the lack of an adequate heat source for the reactor coolant and main steam systems.

Low-power testing is performed in a controlled manner in accordance with written procedures. The minimum crew required to support low-power testing is available in addition to a senior reactor operator with no other concurrent duties who is in charge of low-power testing operations. Low-power testing procedures include instructions and precautions necessary for conducting tests such as adherence to TS requirements, testing sequence, measurement to be taken and test conditions as well as actions to be taken in the event of unanticipated errors or malfunctions. These procedures provide direction for restoration to normal following the test.

Refer to Section 14.2.12 for a list of low-power tests.

COL Item 14.2-3: A COL applicant that references the NuScale Power Plant design certification will identify the specific operator training to be conducted during low-power testing related to the resolution of TMI Action Plan Item I.G.1, as described in NUREG-0660, NUREG-0694, and NUREG-0737.

14.2.10.5 Power-Ascension Tests

Power-ascension testing is performed following the successful completion of low-power testing. Power-ascension testing is performed to bring the reactor to full power and while doing so performing major testing at power levels of approximately 25 percent, 50 percent, 75 percent, and 100 percent. The purpose of the testing is to demonstrate that the plant operates in accordance with its design bases during normal steady state conditions and, to the extent practicable, during and following anticipated transients as well as to demonstrate the validity of analytical models by comparing measured responses with predicted responses. Predicted responses are developed using real or expected values of attributes such as beginning of life core reactivity coefficients, flow rates, pressures, temperatures, and response times of equipment, as well as the actual status of the plant (not those values or plant conditions assumed for conservative evaluations of postulated accidents).

Tests and acceptance criteria are prescribed to demonstrate the ability of principal plant control systems to automatically control process variables within design limits. Such tests are expected to provide assurance that the facility's integrated dynamic response is in accordance with the design for plant events such as reactor scram, turbine trip, and loss of feedwater heaters or pumps. The testing performed is sufficiently comprehensive to establish that the facility can operate in the operating modes for which it has been designed. Testing is not conducted in operating modes or plant configurations that have not been analyzed or that fall outside the range of assumptions used in analyzing postulated accidents described in the Final Safety Analysis Report.

Power-ascension testing is performed in a controlled manner in accordance with written procedures. The minimum crew required to support power-ascension testing is available in addition to a senior reactor operator with no other concurrent duties who is in charge of power-ascension testing operations. Power-ascension testing procedures include instructions and precautions necessary for conducting tests such as adherence to TS requirements, testing sequence, measurement to be taken and test conditions as well as actions to be taken in the event of unanticipated errors or malfunctions. These procedures provide direction for restoration to normal following the test.

Refer to Section 14.2.12 for a list of power-ascension tests.

The completed power-ascension testing program is reviewed at each plateau. Test results are evaluated and the required approvals are received before ascending to the next power level or test condition.

14.2.11 Test Program Schedule and Sequence

Testing schedules are developed taking into account development and approval of plant procedures for use as part of the ITP.

Testing schedules are developed so that SSC that are required to prevent or mitigate the consequences of postulated accidents are tested prior to fuel loading.

Approved test procedures are submitted to the NRC approximately 60 days before their intended use or at least 60 days prior to fuel loading for fuel loading and startup test procedures. The NRC is notified of test procedure changes prior to performance.

Test procedures are essentially identical for each NPM. SSC identification numbering is specific to each NPM.

For individual startup tests, test requirements are completed in accordance with plant TS requirements associated with SSC functionality before changing plant modes.

Testing required to be completed prior to fuel load that is intended to satisfy the requirements for completing ITAAC is identified and documented as such.

Vibration testing is performed in accordance to the requirements of the NuScale CVAP as described in the "NuScale Comprehensive Vibration Assessment Program Technical Report," TR-0716-50439. The technical report contains a schedule for the CVAP testing. Test results are verified following power-ascension testing. See Section 3.9.2 for information pertaining to the CVAP.

The sequential schedule for individual startup tests establishes, insofar as practicable, that test requirements are completed prior to exceeding 25 percent power for the plant SSC that are relied upon to prevent, limit, or mitigate the consequences of postulated accidents. The schedule establishes that, insofar as practicable, the sequencing of testing is accomplished as early in the test program as feasible and that the safety of the plant is not dependent on the performance of untested systems, components, or features. Startup test data is reviewed and approved prior to moving onto the next power plateau. Startup testing is discussed in Section 14.2.1.

The NuScale Power Plant is comprised of up to 12 NPMs. A schedule is developed for startup of each NPM. Preoperational and startup testing schedule considerations include:

- preoperational test schedule duration will be greatest for the first NPM because the first NPM will require testing of systems common to other NPMs

- preoperational and startup test schedule duration should decrease for each successive NPM due to increase in personnel experience and refinement of test procedures
- scheduling such that overlapping test program schedules will not result in significant divisions of responsibilities or dilute staff provided to implement the test program
- plant safety will not be dependent on the performance of untested SSC during the startup test program

Refer to Section 21.3.3 for information pertaining to phased construction and testing activities due to addition of individual NPMs.

COL Item 14.2-4: A COL applicant that references the NuScale Power Plant design certification will provide a schedule for the Initial Test Program.

14.2.12 Individual Test Descriptions

Individual test abstracts are provided in Table 14.2-1 through Table 14.2-108. Table 14.2-109 provides a listing of the test abstracts. Each abstract identifies each test by title, identifies test objectives, prerequisites, test methods, and acceptance criteria. Detailed preoperational and startup test procedures are developed using these test abstracts.

The test abstracts identify pertinent precautions for individual tests, as necessary (e.g., minimum flow requirements or reactor power level that must be maintained).

Table 14.2-1: Spent Fuel Pool Cooling System Test # 1

<table>
<tr><td colspan="3">Preoperational test is required to be performed once.</td></tr>
<tr><td colspan="3">The spent fuel pool cooling system (SFPCS) is described in Section 9.1.3.2.1. SFPCS functions are not verified by SFPCS tests. SFPCS functions verified by another test is:</td></tr>
<tr><td>System Function</td><td>System Function Categorization</td><td>Function Verified by Test #</td></tr>
<tr><td>The SFPCS supports the pool cleanup system (PCUS) by providing fuel pool water for purification of the ultimate heat sink (UHS).</td><td>nonsafety-related</td><td>Pool Cleanup System Test #2-1</td></tr>
<tr><td colspan="3">Prerequisites
Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.</td></tr>
<tr><td colspan="3">Component Level Tests</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>i. Verify each SFPCS remotely-operated valve can be operated remotely.</td><td>Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).</td><td>MCR display and local, visual observation indicate each valve fully opens and fully closes.</td></tr>
<tr><td>ii. Verify each SFPCS air-operated valve fails to its safe position on loss of air.</td><td>Place each valve in its non-safe position. Isolate and vent air to the valve.</td><td>MCR display and local, visual observation indicate each valve fails to its safe position.</td></tr>
<tr><td>iii. Verify each SFPCS air-operated valve fails to its safe position on loss of electrical power to its solenoid.</td><td>Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.</td><td>MCR display and local, visual observation indicate each valve fails to its safe position.</td></tr>
<tr><td>iv. Verify each SFPCS pump can be started and stopped remotely.</td><td>Align the SFPCS to allow for pump operation.
Stop and start each pump from the MCR.</td><td>MCR display and local, visual observation indicate each pump starts and stops.</td></tr>
<tr><td>v. Verify a local grab sample can be obtained from a SFPCS grab sample device.</td><td>Place the system in service to allow flow through the grab sampling device.</td><td>A local grab sample is successfully obtained.</td></tr>
<tr><td>vi. Verify each SFPC instrument is available on a module control system (MCS) or plant control system (PCS) display.
(Test not required if the instrument calibration verified the MCS or PCS display.)</td><td>Initiate a single real or simulated instrument signal from each SFPCS transmitter.</td><td>The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.</td></tr>
<tr><td colspan="3">System Level Tests</td></tr>
<tr><td colspan="3">None</td></tr>
</table>

Table 14.2-2: Pool Cleanup System Test # 2

Preoperational test is required to be performed once.		
The pool cleanup system (PCUS) is described in Section 9.1.3.2.3 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The spent fuel pool cooling system (SFPCS) supports the PCUS by providing spent fuel pool water for purification of the ultimate heat sink (UHS).	nonsafety-related	Test #2-1
2. The reactor pool cooling system (RPCS) supports the PCUS by providing reactor pool water for purification of the UHS.	nonsafety-related	Test #2-1
3. The pool surge control system (PSCS) supports the PCUS by providing water from the dry dock for UHS inventory control.	nonsafety-related	Component level tests Pool Surge Control System Test #4-1
4. The PCUS supports the PSCS, RPCS, and SFPCS by providing a flowpath to cross-connect the PSCS, RPCS, and SFPCS.	nonsafety-related	Component level tests
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify a pump curve test has been completed and approved for the RPCS pumps. iii. Verify a pump curve test has been completed and approved for the SFPCS pumps. iv. Verify a UHS leakage test has been performed.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each PCUS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each PCUS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each PCUS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify a local grab sample can be obtained from a PCUS grab sample device.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
v. Verify each PCUS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each PCUS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

Table 14.2-2: Pool Cleanup System Test # 2 (Continued)

System Level Test #2-1		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify the SFPCS and the RPCS provide design flow rate to the UHS when aligned for PCUS water cleanup. ii. Verify the SFPCS and the RPCS provide design flow rate to the UHS following a PCUS isolation.	i. Place the SFPCS in service to flow through a pool cleanup filter and a demineralizer and return flow to the spent fuel pool. AND Place the RPCS in service to flow through a different pool cleanup filter and demineralizer and return flow to the reactor pool. ii. Simulate a high water temperature upstream of one of the pool cleanup filters.	i. a. The MCR indication for SFPCS pump flow satisfies the design flow rate specified in Table 9.1.3-1a. b. The MCR indication for RPCS pump flow satisfies the design flow rate specified in Table 9.1.3-1b. ii. a. SFPCS flow and RPCS flow to the pool cleanup filters and demineralizers stop. b. The SFPCS flow is bypassed to the spent fuel pool. c. The RPCS cooling flow is bypassed to the reactor pool. d. The MCR indication for SFPCS pump flow satisfies the design flow rate specified in Table 9.1.3-1a. e. The MCR indication for RPCS pump flow satisfies the design flow rate specified in Table 9.1.3-1b.

Table 14.2-3: Reactor Pool Cooling System Test # 3

Preoperational test is required to be performed once.		
The reactor pool cooling system (RPCS) is described in Section 9.1.3.2.2. RPCS functions are not verified by RPCS tests. RPCS functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The RPCS supports the pool cleanup system (PCUS) by providing reactor pool water for purification of the ultimate heat sink (UHS).	nonsafety-related	Pool Cleanup System Test #2-1
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each RPCS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each RPCS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each RPCS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each RPCS pump can be started and stopped remotely.	Align the RPCS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops.
v. Verify a local grab sample can be obtained from an RPCS grab sample device indicated on the RPCS piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
vi. Verify each RPCS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each RPCS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-4: Pool Surge Control System Test # 4

Preoperational test is required to be performed once.		
The pool surge control system (PSCS) is described in Section 9.1.3.2.4 and the function verified by this test is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The PSCS supports the ultimate heat sink (UHS) by providing surge control for UHS operations.	nonsafety-related	Test #4-1
2. The PSCS supports the UHS by providing a reactor inspection dry dock makeup and drain capability.	nonsafety-related	Test #4-1
3. The PSCS supports the pool cleanup system (PCUS) by providing water from the dry dock for UHS inventory control.	nonsafety-related	Test #4-1 Pool Cleanup System Test #2 Component level tests
4. The PCUS supports the PSCS, RPCS, and spent fuel pool cooling system (SFPCS) by providing a flowpath to cross-connect the PSCS, RPCS, and SFPCS.	nonsafety-related	Component level tests
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. For system level test #4-1, the drydock gate can be open or closed.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each PSCS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each PSCS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each PSCS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each PSCS pump can be started and stopped remotely.	Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops.
v. Verify the PSCS automatically responds to mitigate a release of radioactivity.	Initiate a real or simulated high radiation signal in the PSCS tank vent line.	i. The PSCS tank inlet isolation valve is closed. ii. The PSCS tank outlet isolation valve is closed. [ITAAC 03.09.10]
vi. Verify a local grab sample can be obtained from a PSCS grab sample device indicated on the PSCS piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
vii. Verify each PSCS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each PSCS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

Table 14.2-4: Pool Surge Control System Test # 4 (Continued)

System Level Test #4-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify PSCS automatic control for dry dock fill and drain.	Align the PSCS for fill and drain of the dry dock. Fill the dry dock to a level that allows operation of the reactor inspection dry dock evacuation pump. i. Start a PSCS pump. ii. Simulate the following PSCS conditions: a. Dry dock low level b. PSCS tank high level iii. Open PSCS tank main discharge line isolation valve. iv. Simulate a high dry dock level.	i. Verify inventory addition to the PSCS tank. ii. a. Pump is stopped and return line to pool surge control tank isolation valve is closed. b. Pump is stopped and return line to PSCS tank isolation valve is closed. iii. Verify inventory addition to the dry dock. iv. PSCS tank main discharge line isolation valve is closed.

Table 14.2-5: Ultimate Heat Sink Test # 5

There are no preoperational tests for the UHS.		
The ultimate heat sink (UHS) is described in Section 9.2.5. The only active functions for the UHS are to provide post-accident monitoring (PAM) Type D instrument signals to the safety display and indication system (SDIS). Refer to Table 14.2-66: Safety Display and Indication Test #66 for testing of PAM Type D displays.		
System Function	**System Function Categorization**	**Function Verified by Test #**
The UHS supports the decay heat removal system (DHRS) by accepting the heat from the DHRS heat exchanger.	safety-related	Reactor Trip from 100 Percent Power Test # 104
Prerequisites: N/A		
Component Level Tests		
None		

Table 14.2-6: Pool Leak Detection System Test # 6

<table>
<tr><td colspan="3">There are no preoperational tests for the pool leakage detection system (PLDS).</td></tr>
<tr><td colspan="3">The PLDS is described in Section 9.1.3.2.5. Leakage from the ultimate heat sink liner gravity drains to the radiation waste drain system (RWDS). Test #23-2 tests the main control room alarm when the RWDS sump fill rate exceeds the PLDS leakage rate setpoint.</td></tr>
<tr><td>System Function</td><td>System Function Categorization</td><td>Function Verified by Test #</td></tr>
<tr><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td colspan="3">Prerequisites:
N/A</td></tr>
<tr><td colspan="3">Component Level Tests</td></tr>
<tr><td colspan="3">None</td></tr>
</table>

Table 14.2-7: Reactor Component Cooling Water System Test # 7

Preoperational test is required to be performed once each for the 0A and 0B shared or common components. The module-specific portions of the test must be completed once for each NuScale Power Module.		
The reactor component cooling water system (RCCWS) is described in Section 9.2.2 and 11.5.2.2.12 and the function verified by this test is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The RCCWS supports the following systems by providing cooling water. • control rod drive system (CRDS) • chemical and volume control system (CVCS) • containment evacuation system (CES) • process sampling system (PSS)	nonsafety-related	Test #7-1
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify an RCCWS flow balance has been performed. iii. Verify a pump curve test has been completed for the RCCWS pumps.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each RCCWS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each RCCWS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each RCCWS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each RCCWS pump can be started and stopped remotely.	Align the RCCWS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify the RCCWS standby pump automatically starts to protect plant equipment.	Align the RCCWS to allow for pump operation. Place a pump in service. Initiate a simulated RCCWS pump low header pressure signal.	MCR display and local, visual observation indicate the standby pump starts. Audible and visible water hammer are not observed when the pump starts.
vi. Verify RCCWS demineralized makeup water level control valve automatically operates to maintain RCCWS expansion tank level.	i. Initiate simulated expansion tank high level signal. ii. Initiate a simulated expansion tank low level signal.	MCR display and local, visual observation indicate the following: i. The demineralized makeup water level control valve is fully closed. ii. The demineralized makeup water level control valve is fully open.
vii. Verify a local grab sample can be obtained from an RCCWS grab sample device.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.

Table 14.2-7: Reactor Component Cooling Water System Test # 7 (Continued)

viii. Verify each RCCWS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each RCCWS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #7-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify RCCWS cooling water flow rates satisfy design flow.	**Module 1 Test** i. Align the 0A RCCWS to provide flow to all the Module 1 heat exchangers cooled by RCCWS listed below: **Module 1 Heat Exchangers** Control rod drive mechanism (CRDM) cooling coils CVCS non-regenerative heat exchanger CES vacuum pump CES condenser PSS analyzer cooler PSS temperature control unit ii. Repeat module 1 test for modules 2 through 6. iii. Align the 0B RCCWS to provide flow to all the Module 7 heat exchangers cooled by RCCWS listed below: **Module 7 Heat Exchangers** CRDM cooling coils CVCS non-regenerative heat exchanger CES vacuum pump CES condenser PSS analyzer cooler PSS temperature control unit iv. Repeat Module 7 test for modules 8 through 12.	The RCCWS cooling flow to each heat exchanger under test meets the flow rate acceptance criteria contained in the RCCWS flow balance report.

Table 14.2-8: Chilled Water System Test # 8

<table>
<tr><td colspan="3">Preoperational test is required to be performed once.</td></tr>
<tr><td colspan="3">The chilled water system (CHWS) is described in Section 9.2.8 and the function verified by this test is:</td></tr>
<tr><td>System Function</td><td>System Function Categorization</td><td>Function Verified by Test #</td></tr>
<tr><td>The CHWS supports the following systems by providing cooling water:
• Reactor Building HVAC system (RBVS)
• normal control room HVAC system (CRVS)
• Radioactive Waste Building HVAC system (RWBVS)
• liquid radioactive waste system (LRWS)
• gaseous radioactive waste system (GRWS)</td><td>nonsafety-related</td><td>Test #8-1
Test #8-2</td></tr>
<tr><td colspan="3">Prerequisites
i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.
ii. Verify a CHWS flow balance has been performed.
iii. Verify a pump curve test has been completed for the CHWS pumps.
iv. Chiller performance has been verified by either an Air Conditioning, Heating, and Refrigeration Institute (AHRI) certification or a chiller performance capacity test witnessed at the factory with all test documentation provided.</td></tr>
<tr><td colspan="3">Component Level Tests</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>i. Verify each CHWS remotely-operated valve can be operated remotely.</td><td>Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).</td><td>MCR display and local, visual observation indicate each valve fully opens and fully closes.</td></tr>
<tr><td>ii. Verify each CHWS air-operated valve fails to its safe position on loss of air.</td><td>Place each valve in its non-safe position. Isolate and vent air to the valve.</td><td>MCR display and local, visual observation indicate each valve fails to its safe position.</td></tr>
<tr><td>iii. Verify each CHWS air-operated valve fails to its safe position on loss of electrical power to its solenoid.</td><td>Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.</td><td>MCR display and local, visual observation indicate each valve fails to its safe position.</td></tr>
<tr><td>iv. Verify the speed of each CHWS variable-speed pump can be manually controlled.</td><td>Align the CHWS to provide a flow path to operate a selected pump.
Vary the CHWS pump speed from minimum to maximum from the MCR.</td><td>MCR display indicates the speed of each obtains both minimum and maximum pump speeds.
Audible and visible water hammer are not observed when the pump starts.</td></tr>
<tr><td>v. Verify automatic operation of CHWS pumps and CHWS chiller to protect plant equipment.</td><td>Align the CHWS to allow for chiller operation. Place a pump in service. Initiate a simulated start signal for the following system conditions.
i. Loss of chilled water flow.
ii. Loss of site cooling water system cooling flow to the operating chiller.</td><td>MCR display and local, visual observation indicate the following:
i. a. Operating pump stops
b. Operating chiller stops
ii. Operating chiller stops</td></tr>
<tr><td>vi. Verify each CHWS instrument is available on a module control system (MCS) or plant control system (PCS) display.
(Test not required if the instrument calibration verified the MCS or PCS display.)</td><td>Initiate a single real or simulated instrument signal from each CHWS transmitter.</td><td>The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.</td></tr>
</table>

Table 14.2-8: Chilled Water System Test # 8 (Continued)

System Level Test #8-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify CHWS cooling water flow rates satisfy design.	i. Align the CHWS to provide flow to all heat exchangers cooled by the CHWS chiller: RBVS air handling units RBVS fan coil units CRVS air handling units CRVS fan coil units RWBVS air handling units RWBVS fan coil units LRWS degasifier condenser GRWS gas coolers ii. Open all CHWS flow control valves.	The CHWS cooling flow to each heat exchanger under test meets the minimum flow rate acceptance criteria contained in the CHWS flow balance report.
System Level Test #8-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify CHWS cooling water flow rates satisfy design flow.	i. Align the CHWS to provide flow to the CRVS air handling units and the CRVS fan coil units cooled by the CHWS standby chiller. ii. Open all CHWS flow control valves.	The CRVS standby CHWS cooling flow to each heat exchanger meets the minimum flow rate acceptance criteria contained in the CHWS flow balance report.

Table 14.2-9: Auxiliary Boiler System Test # 9

Preoperational test is required to be performed once.		
The auxiliary boiler system (ABS) is described in Section 10.4.10 and 11.5.2.2.14 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
None		
The ABS functions verified by other tests are:		
The auxiliary boiler supports the condensate polishing system (CPS) by supplying steam for resin regeneration.	nonsafety-related	Condensate Polishing System Test #30-1
The auxiliary boiler supports the turbine generator by supplying gland seal steam.	nonsafety-related	Condenser Air Removal System Test #32-1
The auxiliary boiler supports the feedwater system by supplying steam to the condenser for sparging when necessary.	nonsafety-related	Condenser Air Removal System Test #32-1
The auxiliary boiler supports the module heatup system by supplying steam for heating reactor coolant at startup and shutdown.	nonsafety-related	Turbine Generator System Test #33-1
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each auxiliary boiler remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each auxiliary boiler air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each auxiliary boiler air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each auxiliary boiler low pressure boiler feedwater pump can be started and stopped remotely.	Align the ABS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify each auxiliary boiler high pressure boiler feedwater pump can be started and stopped remotely.	Align the ABS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
vi. Verify the speed of each auxiliary boiler high pressure boiler feedwater pump can be manually controlled.	Align the ABS to provide a flow path to operate a selected ABS variable-speed pump. Vary the auxiliary boiler pump speed from minimum to maximum speed from the MCR.	MCR display indicates the speed of each variable speed pump obtains both minimum and maximum pump speeds.

Table 14.2-9: Auxiliary Boiler System Test # 9 (Continued)

vii. Verify the ABS automatically responds to mitigate a release of radioactivity.	Initiate a real or simulated high radiation signal for the auxiliary boiler flash tank vent.	MCR display verifies the following: i. auxiliary boiler flash tank vent isolation valve is closed. ii. auxiliary boiler high pressure steam supply isolation valves are closed. [ITAAC 03.09.08] (i.and ii.)
viii. Verify the ABS automatically responds to mitigate a release of radioactivity.	Initiate a real or simulated high radiation signal for the auxiliary boiler high pressure to low pressure steam supply.	MCR display verifies the auxiliary boiler high pressure to low pressure steam supply pressure control valve is closed. [ITAAC 03.09.08]
ix. Verify each ABS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each ABS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test		
Test Objective	**Test Method**	**Acceptance Criteria**
None		

Table 14.2-10: Circulating Water System Test # 10

This preoperational test is required to be performed once for each circulating water subsystem.		
The circulating water system (CWS) is described in Section 10.4.5 and the function verified by this test and power ascension testing is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The utility water system (UWS) supports the CWS by providing makeup water to maintain water level in the CWS cooling tower basins.	nonsafety-related	Component-Level Test vi. Ramp Change in Load Demand Test #100
The CWS function verified by another test is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The CWS supports the feedwater system by removing heat from the main condenser.	nonsafety-related	Condenser Air Removal System Test #32-1 Ramp Change in Load Demand Test #100 100 Percent Load Rejection Test #103
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests: NuScale Power Module (NPM) #1 (#7) The minimum inventory of pumps, fans and valves tested for NPM #1 (#7) is that inventory required for 0A (0B) CWS operation to support operation of NPM #1 (#7). The testing will continue until all 0A (0B) CWS equipment is tested.		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each CWS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each CWS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each CWS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each CWS cooling tower fan can be started and stopped remotely	Align the CWS to allow for cooling tower fan operation. Stop and start each cooling tower fan from the MCR.	MCR display and local, visual observation indicate each cooling tower fan starts and stops.
v. Verify each CWS pump can be started and stopped remotely.	Align the CWS to allow for pump operation. Stop and start each pump from the MCR.	i. MCR display and local, visual observation indicate each pump starts and stops. ii. Audible and visible water hammer are not observed when the pump starts. iii. CWS pump cavitation is not observed. iv. Cooling towers do not experience flow surge or overflow.
vi. Verify automatic operation of the CWS cooling tower basin level control valve to maintain CWS cooling tower basin level.	i. Initiate a cooling tower basin low level signal. ii. Initiate a cooling tower basin high level signal.	MCR displays and local, visual observation verifies the following: i. The cooling tower basin level control valve is open. ii. The cooling tower basin level control valve is closed.

Table 14.2-10: Circulating Water System Test # 10 (Continued)

vii. Verify each CWS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each CWS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-11: Site Cooling Water System Test # 11

Preoperational test is required to be performed for each NuScale Power Module (NPM).		
The site cooling water system (SCWS) is described in Section 9.2.7 and 11.5.2.2.13 and the functions verified by this test and power ascension testing are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The SCWS supports the following systems by providing cooling water. • turbine generator system (TGS) • reactor component cooling water system (RCCWS) • condenser air removal system (CARS) • process sampling system (PSS) • chilled water system (CHWS) • instrument air system (IAS) • spent fuel pool cooling system (SFPCS) • reactor pool cooling system (RPCS) • feedwater system (FWS) • balance-of-plant drain system (BPDS)	nonsafety-related	Test #11-1
The utility water system (UWS) supports the SCWS by providing makeup water to maintain water level in the SCWS cooling tower basins.	nonsafety-related	Component-Level Test vii. Ramp Change in Load Demand Test #100
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify an SCWS flow balance has been performed and the system flow balance records have been approved. iii. Verify a pump curve test has been completed and approved for the SCWS pumps.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each SCWS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each SCWS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each SCWS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each SCWS cooling tower fan can be started and stopped remotely.	Align the SCWS to allow for cooling tower fan operation. Stop and start each cooling tower fan from the MCR.	MCR display and local, visual observation indicate each cooling tower fan starts and stops.
v. Verify each SCWS pump can be started and stopped remotely.	Align the SCWS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
vi. Verify the SCWS standby pump automatically starts to protect plant equipment.	Align the SCWS to allow for pump operation. Place a pump in service. Initiate a simulated start signal.	MCR display and local, visual observation indicate the standby pump discharge valve opens to a throttled position, the pump starts, and then the discharge valve fully opens. Audible and visible water hammer are not observed when the pump starts.

Table 14.2-11: Site Cooling Water System Test # 11 (Continued)

vii. Verify a local grab sample can be obtained from a SCWS grab sample device indicated on the SCWS piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
viii. Verify each SCWS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each SCWS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #11-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify SCWS cooling water flow rates satisfy design flow.	i. **NPM 1 Test** Align the SCWS to provide flow to all the Module 1 and common heat exchangers cooled by SCWS listed below: **NPM 1 Heat Exchangers** CARS pumps TGS coolers PSS coolers **Common Heat Exchangers** CHWS chillers IAS coolers PSS chillers RPCS heat exchangers SFPCS heat exchangers RCCWS heat exchangers BPDS aux boiler blowdown coolers The operation of two SCWS pumps may be required to provide sufficient flow to meet acceptance criteria in the SCWS flow balance report. ii. **NPM 2-12 Test** The scope of each subsequent test will include one or more additional modules. The scope will also include previously tested modules to verify that the flow rate still meets the flow rate acceptance criteria contained in the SCWS flow balance report. The testing will continue until all heat exchangers cooled by SCWS have been tested in a single test.	The SCWS cooling flow to each heat exchanger under test meets the minimum flow rate acceptance criteria contained in the SCWS flow balance report.

Table 14.2-12: Potable Water System Test # 12

The potable water system (PWS) is described in Section 9.2.4. The PWS is a site-specific system, and the testing of the PWS is the responsibility of the COL applicant. COL Item 14.2-5: A COL applicant that references the NuScale Power Plant design certification will provide a test abstract for the potable water system pre-operational testing.		
System Function	**System Function Categorization**	**Function Verified by Test #**
As described in Section 9.2.4	nonsafety-related	Provided by COL applicant
Prerequisites Provided by COL applicant		
Component Level Tests		
Provided by COL applicant		
System Level Tests		
Provided by COL applicant		

Table 14.2-13: Utility Water System Test # 13

Preoperational test is required to be performed once.		
The utility water system (UWS) is described in Section 9.2.9 and the functions verified by this test and power ascension testing are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The UWS supports the circulating water system by providing makeup water to maintain water level in the circulating water system cooling tower basins.	nonsafety-related	Circulating Water System Test #10 Component-Level Test vi. Ramp Change in Load Demand Test #100
2. The UWS supports the site cooling water system (SCWS) by providing makeup water to maintain water level in the SCWS cooling tower basins.	nonsafety related	Site Cooling Water System Test #11 Component-Level Test vii. Ramp Change in Load Demand Test #100
3. The UWS supports the following systems by providing makeup water: • demineralized water system • fire protection system • potable water system • chilled water system • Reactor Building • Turbine Generator Building • Radioactive Waste Building • Annex Building • Control Building	nonsafety-related	component-level tests
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify a pump curve test has been completed for the UWS pumps.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each UWS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each UWS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each UWS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each UWS pump can be started and stopped remotely.	Align the UWS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify UWS flow capability by automatic start of a UWS pump while in standby mode.	Align the UWS to allow for pump operation. Place a pump in service. Initiate a simulated pump trip.	MCR display and local, visual observation indicate the standby pump starts. Audible and visible water hammer are not observed when the pump starts.
vi. Verify a local grab sample can be obtained from a UWS grab sample device indicated on the UWS piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is obtained.

Table 14.2-13: Utility Water System Test # 13 (Continued)

vii. Verify each UWS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each UWS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-14: Demineralized Water System Test # 14

Preoperational test is required to be performed once.		
The demineralized water system (DWS) is described in Section 9.2.3 and 11.5.2.2.16 and the function verified by this test is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The DWS supports the following systems by providing makeup water. • chemical and volume control system • boron addition system • liquid radioactive waste system • spent fuel pool cooling system • reactor component cooling water system • process sampling system • feedwater system • auxiliary boiler system • normal control room HVAC system • condenser air removal system • containment evacuation system • Reactor Building HVAC system • Radioactive Waste Building HVAC system • condensate polishing system • pool cleanup system • annex building • balance of plant drains • turbine building HVAC system • annex building HVAC system • reactor building • radioactive waste building • feedwater treatment system	nonsafety-related	component-level tests
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify a pump curve test has been completed for the DWS pumps.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each DWS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control)	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each DWS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each DWS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify the DWS pump can be started and stopped remotely.	Align the DWS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.

Table 14.2-14: Demineralized Water System Test # 14 (Continued)

v. Verify DWS flow capability by automatic start of a DWS pump while in standby mode.	Align the DWS to allow for pump operation. Place a pump in service. Initiate a simulated pump trip.	MCR display and local, visual observation indicate the standby pump starts. Audible and visible water hammer are not observed when the pump starts.
vi. Verify a local grab sample can be obtained from a DWS grab sample device indicated on the DWS piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is obtained.
vii. Verify each DWS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each DWS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-15: Nitrogen Distribution System Test # 15

Preoperational test is required to be performed once.		
The nitrogen distribution system (NDS) is described in Section 9.3.1 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The NDS supports the following systems by providing nitrogen: • chemical and volume control system • containment evacuation system • Reactor Building	nonsafety-related	component-level tests
2. The NDS supports the liquid radioactive waste system (LRWS) by providing nitrogen for purging of the LRWS.	nonsafety-related	component-level tests Liquid Radioactive Waste System Test #35-1
3. The NDS supports the gaseous radioactive waste system (GRWS) by providing nitrogen for purging of the GRWS.	nonsafety-related	component-level tests Gaseous Radioactive Waste System Test #36-1
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each NDS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control)	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each NDS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each NDS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify a local grab sample can be obtained from a NDS grab sample device indicated on the NDS piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
v. Verify each NDS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each NDS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-16: Service Air System Test # 16

Preoperational test is required to be performed once.		
The service air system (SAS) is described in Section 9.3.1 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
Has no specific system function, all functionality is supported through supported systems testing.	N/A	N/A
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each SAS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control)	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each SAS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each SAS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each SAS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each SAS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-17: Instrument Air System Test # 17

Preoperational test is required to be performed once.		
The instrument air system (IAS) is described in Section 9.3.1 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
Has no specific system function, all functionality is supported through supported systems testing.	N/A	N/A
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. Verify performance testing of air compressor skids have been completed by the manufacturer or a site acceptance test has been completed in accordance with manufacturer instructions.		
Component Level Tests: First NuScale Power Module (NPM)		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each IAS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each IAS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each IAS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify the speed of each IAS variable-speed compressor can be manually controlled.	Align the system to provide a flow path to operate a selected compressor. Vary the compressor speed from minimum to maximum from the MCR.	MCR display indicate the speed of each compressor obtains both minimum and maximum pump speeds.
v. Verify each IAS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each IAS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
Component Level Tests: NPMs 2-12		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each IAS remotely-operated valve can be operated remotely.	Operate each valve from the MCR and local control panel (if design has local valve control)	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each IAS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each IAS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each IAS instrument is available on an MCS or PCS display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each IAS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-18: Control Room Habitability System Test # 18

Preoperational test is required to be performed once.		
The control room habitability system (CRHS) is described in Section 6.4 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The CRHS supports the Control Building (CRB) by providing clean breathing air to the control room envelope (CRE) and maintaining a positive control room pressure during high radiation or loss of offsite power conditions.	nonsafety-related	Test #18-1 Test #18-2 Test #18-3
2. The CRHS supports the CRB by providing high pressure, clean breathing air in air bottles for use.	nonsafety-related	Test #18-1 Test #18-2
3. The normal control room HVAC system (CRVS) supports the CRB by providing isolation of the CRE from the surrounding areas and outside environment via isolation dampers.	nonsafety-related	Test #18-1
4. The plant protection system (PPS) supports the CRHS by providing actuation and control signals.	nonsafety-related	Test #18-1
5. The CRVS supports the CRB by providing isolation of the CRE from the surrounding areas and outside environment via isolation dampers.	nonsafety-related	Test #18-1
6. The CRVS supports the PPS by providing instrument information signals relating to isolation of the CRE and activation of the CRH system.	nonsafety-related	Test #18-1
7. The CRVS supports the CRB by isolating the CRVS outside air intake from the environment and operating CRVS in recirculation mode to prevent exposure to smoke and toxic gas, or when radiation is detected downstream of the charcoal filtration unit.	nonsafety-related	Test #18-1 (radiation detection) Normal Control Room HVAC System Test #19-3 (smoke/toxic gas)
8. The PPS supports the CRVS by providing actuation and control signals to the CRE isolation dampers.	nonsafety-related	Test #18-1
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify a CRHS air balance has been performed and the CRHS air balance records have been approved. [This prerequisite is not required for component-level tests.] iii. Verify CRHS air bottlers are pressurized to their design working pressure. [This prerequisite is not required for component-level tests.] iv. Component Level Tests i. and ii. must be performed under preoperational test conditions that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.		

Table 14.2-18: Control Room Habitability System Test # 18 (Continued)

Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each CRHS remotely-operated valve can be operated remotely.	Place the CRHS air bottles in service. Place CRVS in service to supply air to the CRE. Operate each valve from the main control room (MCR).	i. MCR workstation display, safety display instrument display and local, visual observation indicate each valve fully opens and fully closes under preoperational temperature, differential pressure, and flow conditions. [ITAAC 03.01.02]
ii. Verify each CRHS solenoid-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place the CRHS air bottles in service. Place CRVS in service to supply air to the CRE. i. Place each valve in its non-safe position. Isolate electrical power to its solenoid.	i. MCR display, safety display instrument display and local, visual observation indicate each valve fails open under preoperational temperature, differential pressure, and flow conditions. [ITAAC 03.01.03]
iii. Verify each CRHS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each CRHS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #18-1		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify PPS provides actuation signals for CRHS and CRVS. ii. CRHS realigns to provide breathable air to the control room envelope (CRE) under accident conditions. iii. CRVS realigns to isolate outside air dampers and CRE under accident conditions.	Place the CRVS in automatic operation. Place the CRHS air bottles in service. Place CRVS in service to supply air to the CRE. Start CRVS filter unit. Initiate each of the following real or simulated CRHS actuation signals: • High radiation signal downstream of the CRVS filter unit • Loss of AC power.	i. MCR workstation display and local, visual observation indicate the following: a. The CRVS outside air damper closes. b. The CRVS filter unit fan stops. c. The CRVS control room envelope isolation dampers close. d. The CRHS air supply isolation valves open. e. CRHS pressure relief isolation valves open. f. CRVS air handling unit stops. g. CRVS general exhaust fan stops. h. CRVS battery room exhaust fan stops. [ITAAC 03.09.02] (items i.a. through i.e.) ii. PPS generates alarms in the MCR for the following: a. High radiation b. Loss of AC power.

Table 14.2-18: Control Room Habitability System Test # 18 (Continued)

System Level Test #18-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify emergency pressurized air bottles have sufficient volume to provide 72 hours of breathable air through both the main and backup supply flow path to the CRE described in Section 6.4.2.1.	i. Align air bottles for testing. Assume 25 percent of the bottles are unavailable and use 1/6 of the remaining bottles to simulate a test conduct of 72 hours (12 hours/72 hours). Initiate a real or simulated CRHS actuation signal to isolate the CRE. Conduct a CRE test for 12 hours. ii. At the end of 12 hours isolate the main supply flow path and align the manual backup flow path to the CRE. Align air bottles for testing. Assume 25 percent of the bottles are unavailable and use the remaining bottles.	i. a. The CRE described in Section 6.4.2.1 maintains a positive pressure relative to the adjacent areas as specified in Table 6.4-1 as indicated by the CRE differential pressure transmitters. [ITAAC 03.01.05] b. The CRHS minimum flow rate for the main flow path is maintained as specified in Table 6.4-1 for the duration of the test. c. The CRHS flow rate for the manual backup flow path is maintained as specified in Table 6.4-1.
System Level Test #18-3		
Test Objective	**Test Method**	**Acceptance Criteria**
The air exfiltration from the CRE does not exceed the air exfiltration flow rate identified in the CRHS exfiltration/infiltration analysis.	Perform an air exfiltration test of the CRE at 1/8 in. wg. of positive pressure with respect to surrounding areas by performing tracer gas testing in accordance with ASTM E741.	The measured air exfiltration flow rate does not exceed the unfiltered inleakage flow rate assumed in the dose analysis identified in Table 6.4-1. [ITAAC 03.01.01]

Table 14.2-19: Normal Control Room HVAC System Test # 19

Preoperational test is required to be performed once.		
The normal control room HVAC system (CRVS) is described in Sections 6.4.3.2, 9.4.1, and 11.5.2.2.1, and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The CRVS supports the Control Building (CRB) by providing cooling, heating and humidity control to maintain a suitable environment for the safety and comfort of plant personnel.	nonsafety-related	Test #19-1 Test #19-2
2. The CRVS supports the systems located in the CRB by providing cooling, heating and humidity control to maintain a suitable environment for the operation of system components.	nonsafety-related	Test #19-1 Test #19-2
3. The CRVS supports the CRB by isolating the CRVS outside air intake from the environment and operating the CRVS in recirculation mode to prevent exposure to smoke and toxic gas, or when radiation is detected downstream of the charcoal filtration unit.	nonsafety-related	Test #19-3 (smoke/toxic gas) Control Room Habitability System Test #18-1 (radiation)
4. The CRVS supports the CRB by maintaining the CRB at a positive ambient pressure relative to the Reactor Building (RXB) and the outside atmosphere to control the ingress of potentially airborne radioactivity from the RXB or the outside atmosphere to the CRB.	nonsafety-related	Test #19-1 (CRB positive pressure) RBVS Test #20-1 (RXB negative pressure)
5. The plant protection system (PPS) supports the CRVS by providing actuation and control signals to the outside air isolation dampers.	nonsafety-related	Test #19-3
6. The CRVS supports the CRB by protecting personnel from exposure to radiation during a design basis accident, when power is available, by removing radioactive contamination from outside air via charcoal filtration, as required by radiation dose analyses.	nonsafety-related	Test #19-4
The CRVS functions verified by other tests are:		
The CRVS supports the CRB by isolating the CRVS outside air intake when radiation is detected downstream of the charcoal filtration unit.	nonsafety-related	Control Room Habitability System Test #18-1
The CRVS supports the CRB by providing isolation of the control room envelope (CRE) from the surrounding areas and outside environment via isolation dampers.	nonsafety-related	Control Room Habitability System Test #18-1

Table 14.2-19: Normal Control Room HVAC System Test # 19 (Continued)

The CRVS supports the PPS by providing instrument information signals relating to isolation of the CRE and activation of the CRHS.	nonsafety-related	Control Room Habitability System Test #18-1
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify a CRVS air balance has been performed and the CRVS air balance records have been approved. [This prerequisite is not required for component-level tests.] iii. Verify CRVS high-efficiency particulate air (HEPA) and charcoal adsorbers have been installed and tested and the test records have been approved. [This prerequisite is not required for component-level tests.] iv. Verify CRVS control room isolation dampers have been leak tested and the test records have been approved. [This prerequisite is not required for component-level tests.] v. Component Level Tests x. and xi. must be performed under preoperational test conditions that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations. vi. Verify CRVS air filtration unit heater testing specified in RG 1.140 C.4.g has been completed and the test records have been approved.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each CRVS remotely-operated damper can be operated remotely.	Operate each damper from the main control room (MCR) and local control panel (if design has local damper control).	MCR display and local, visual observation indicate each damper fully opens and fully closes.
ii. Verify each CRVS air-operated damper fails to its safe position on loss of air.	Place each damper in its non-safe position. Isolate and vent air to the damper.	MCR display and local, visual observation indicate each damper fails to its safe position.
iii. Verify each CRVS air-operated damper fails to its safe position on loss of electrical power to its solenoid.	Place each damper in its non-safe position. Isolate electrical power to its solenoid.	MCR display and local, visual observation indicate each damper fails to its safe position.
iv. Verify CRVS dampers automatically close on associated smoke or fire signals.	Open each damper actuated by a smoke or fire signal. Initiate an alarm signal for each damper.	MCR display and local, visual observation indicate each damper closes.
v. Verify each required CRVS fan stops on actuation of its associated fire or smoke alarm.	Initiate an alarm signal for each fan.	MCR display and local, visual observation indicate each fan stops.
vi. Verify each CRVS pressurization fan starts automatically on the actuation of its associated fire or smoke alarm.	Initiate an alarm signal for each fan.	MCR display and local, visual observation indicate each pressurization fan starts.
vii. Verify the fan speed of each CRVS variable-speed fan can be manually controlled.	Vary the speed of each fan from the MCR and local control panel (if design has local fan control).	MCR display indicates the speed of each fan varies from minimum to maximum speed.
viii. Verify the standby CRVS main supply air handling unit (AHU) starts automatically on the stop of the operating CRVS main supply AHU.	Place an AHU in service. Place the standby AHU in automatic control. Stop the operating AHU.	MCR display and local, visual observation indicate the standby AHU starts.
ix. Verify each standby CRVS fan coil unit (FCU) starts automatically on the stop of the operating CRVS fan coil unit.	Place an FCU in service. Place the standby FCU in automatic control. Stop the operating FCU.	MCR display and local, visual observation indicate the standby FCU starts.

Table 14.2-19: Normal Control Room HVAC System Test # 19 (Continued)

x. Verify each CRVS control room envelope isolation damper fails to its safe position on loss of air.	Place each damper in its non-safe position. Isolate and vent air to the damper.	Each CRVS control room envelope isolation damper fails to its closed position on loss of air under preoperational temperature, differential pressure, and flow conditions while the CRVS is supplying flow to the CRE. [ITAAC 03.02.01]
xi. Verify each CRVS control room envelope isolation damper fails to its safe position on loss of electrical power to its solenoid.	Place each damper in its non-safe position. Isolate electrical power to its solenoid.	Each CRVS control room envelope isolation damper fails to its closed position on loss of electrical power under preoperational temperature, differential pressure, and flow conditions while the CRVS is supplying flow to the CRE. [ITAAC 03.02.01]
xii. Verify each CRVS remotely operated fan can be operated remotely.	Operate each fan from the MCR and local control panel (if design has local fan control).	MCR display and local, visual observation indicate each fan starts and stops.
xiii. Verify each CRVS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each CRVS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #19-1		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify CRB design temperatures and humidity monitored by the MCR are maintained at design temperature and humidity conditions during normal operation. ii. Verify The CRVS maintains a positive pressure in the CRB relative to the outside environment while the CRVS is operating in normal alignment. iii. Verify the CRVS maintains the air flow to the battery rooms to maintain hydrogen concentration to less than 1 percent by volume.	Place the CRVS in automatic operation. i. Record the CRB temperatures and humidity indications monitored by the MCR. ii. Measure the CRB pressure relative the outside environment. iii. Measure the air flow rate to the battery rooms.	i. The temperature and humidity of rooms and areas monitored by the MCR satisfy the design temperature and humidity requirements contained in Table 9.4.1-2. ii. The CRVS maintains a positive pressure of greater than or equal to 0.125 inches water gauge in the CRB relative to the outside environment, while operating in the normal operating alignment. [ITAAC 03.02.02] iii. Measured flow to the battery rooms is equal to or greater than the flow specified by the air flow balance. [ITAAC 03.02.03]
System Level Test #19-2		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify CRB design temperatures and humidity monitored by the MCR are maintained at design temperature and humidity conditions while cooling to the CRV main supply AHU is supplied by the chilled water system (CHWS) standby chiller.	Align the CHWS standby chiller to cool each CRVS main supply AHU. Place the CRVS in automatic operation.	The temperature and humidity of rooms and areas monitored by the MCR satisfy the design temperature and humidity requirements contained in Table 9.4.1-2.

Table 14.2-19: Normal Control Room HVAC System Test # 19 (Continued)

System Level Test #19-3		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify PPS actuates CRVS outside air dampers when toxic gas or smoke is detected in the makeup air ductwork.	Place the CRVS in automatic operation. i. Initiate a simulated high smoke or toxic gas signal for the makeup air ductwork upstream of the CRVS filter unit.	Outside air dampers close to isolate makeup air.
System Level Test #19-4		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the CRVS automatically responds to mitigate the consequences of high radiation in the outside air.	Place the CRVS in automatic operation. Initiate a real or simulated high radiation signal for the outside air ductwork upstream of the CRVS filter unit.	i. Outside air is diverted through the CRVS filter unit by closing the CRVS filter unit bypass dampers and opening the CRVS filter unit isolation dampers. ii. The CRVS filter unit fan starts. [ITAAC 03.09.01] (items i. and ii.)

Table 14.2-20: Reactor Building HVAC System Test # 20

Preoperational test is required to be performed once.		
The Reactor Building HVAC system (RBVS) is described in Section 9.4.2 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The RBVS supports the Reactor Building (RXB) by providing cooling, heating and humidity control to maintain a suitable environment for the safety and comfort of plant personnel.	nonsafety-related	Test #20-1 Test #20-2 Reactor Building Ventilation System Capability Test # 96
2. The RBVS supports the systems located in the RXB by providing cooling, heating and humidity control to maintain a suitable environment for the operation of system components.	nonsafety-related	Test #20-1 Test #20-2 Reactor Building Ventilation System Capability Test # 96
3. The RBVS supports the RXB by maintaining the RXB at a negative ambient pressure relative to the outside atmosphere to control the movement of potentially airborne radioactivity from the RXB to the environment.	nonsafety-related	Test #20-1 Test #20-3
4. The normal control room HVAC system (CRVS) supports the Control Building (CRB) by maintaining the CRB at a positive ambient 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.	nonsafety-related	Test #20-1 (RXB negative pressure) Normal Control Room HVAC System Test #19-1 (CRB positive pressure)
5. The Radioactive Waste Building HVAC system (RWBVS) supports the Radioactive Waste Building (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.	nonsafety-related	Test #20-3 (off-normal RBVS exhaust alignment) Radioactive Waste Building HVAC System Test #21-1 (normal RBVS exhaust alignment)
Prerequisites		
i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
ii. Verify an RBVS air balance has been performed and the RBVS air balance records have been approved. [This prerequisite is not required for component-level tests.] (Note: The RBVS is designed to move air from areas that are not contaminated or are expected to have low levels of contamination to areas that are likely to be more contaminated.)		
iii. RBVS high-efficiency particulate air and charcoal adsorbers have been installed and tested. [This prerequisite is not required for component-level tests.]		
iv. Verify spent fuel pool exhaust charcoal and HEPA filter unit heater bank testing specified in RG 1.140 C.4.g has been completed and the test records have been approved.		

Table 14.2-20: Reactor Building HVAC System Test # 20 (Continued)

Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each RBVS remotely-operated damper can be operated remotely.	Operate each damper from the main control room (MCR) and local control panel (if design has local damper control).	MCR display and local, visual observation indicate each damper fully opens and fully closes.
ii. Verify each RBVS air-operated damper fails to its safe position on loss of air.	Place each damper in its non-safe position. Isolate and vent air to the damper.	MCR display and local, visual observation indicate each damper fails to its safe position.
iii. Verify each RBVS air-operated damper fails to its safe position on loss of electrical power to its solenoid.	Place each damper in its non-safe position. Isolate electrical power to its solenoid.	MCR display and local, visual observation indicate each damper fails to its safe position.
iv. Verify RBVS dampers automatically close on associated smoke or fire signals.	Open each damper actuated by a smoke or fire signal. Initiate an alarm signal for each damper.	MCR display and local, visual observation indicate each damper closes.
v. Verify each required RBVS fan stops on actuation of its associated fire or smoke alarm.	Initiate an alarm signal for each fan.	MCR display and local, visual observation indicate each fan stops.
vi. Verify each RBVS pressurization fan starts automatically on the actuation of its associated fire or smoke alarm.	Initiate an alarm signal for each fan.	MCR display and local, visual observation indicate each RBVS pressurization fan starts.
vii. Verify the fan speed of each RBVS variable-speed fan can be manually controlled.	Vary the speed of each fan from the MCR and local control panel (if design has local fan control).	MCR display indicates the speed of each fan varies from minimum to maximum speed.
viii. Verify each standby RBVS air handling unit starts automatically on the stop of the operating RBVS air handling unit (AHU).	Place an AHU in service. Place the standby AHU in automatic control. Stop the operating AHU.	MCR display and local, visual observation indicate the standby AHU starts.
ix. Verify each standby RBVS fan coil unit (FCU) starts automatically on the stop of the operating RBVS fan coil unit.	Place an FCU in service. Place the standby FCU in automatic control. Stop the operating FCU.	MCR display and local, visual observation indicate the standby FCU starts.
x. Verify each RBVS remotely-operated fan can be started and stopped remotely.	Start and stop each fan from the MCR and local panel (if design has local fan control).	MCR display and local, visual observation indicate each fan started and stopped.
xi. Verify each RBVS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each RBVS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

Table 14.2-20: Reactor Building HVAC System Test # 20 (Continued)

System Level Test #20-1		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify RXB design temperatures and humidity monitored by the MCR are maintained at design temperature and humidity conditions during normal operation. ii. Verify The RBVS maintains a negative pressure in the RXB relative to the outside environment while the RBVS is operating in normal alignment. iii. Verify The RBVS maintains a negative pressure in the RWB relative to the outside environment while the RBVS is operating in normal alignment. iv. Verify the RBVS maintains the air flow to the battery rooms to maintain hydrogen concentration to less than 1 percent by volume.	Place the RBVS supply, general area exhaust and spent fuel pool exhaust in automatic operation. Place the RWBVS in automatic operation. i. Record the RXB temperatures and humidity indications monitored by the MCR. ii. Measure the RXB pressure relative the outside environment. iii. Measure the RWB pressure relative the outside environment. iv. Measure the air flow rate to the battery rooms.	i. The temperature and humidity of rooms and areas monitored by the MCR satisfy the design temperature and humidity requirements contained in Table 9.4.2-2. ii. MCR display indicates the RBVS maintains a negative pressure in the RXB relative to the outside environment while operating in the normal operating alignment. [ITAAC 03.03.01] iii. MCR display indicates the RBVS maintains a negative pressure in the RWB relative to the outside environment while operating in the normal operating alignment. [ITAAC 03.03.02] iv. Measured flow to the battery rooms is equal to or greater than the flow specified by the air flow balance. [ITAAC 03.03.03]
System Level Test #20-2		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify design temperatures of the following rooms can be controlled using AHUs with installed direct expansion coils. a. remote shutdown station b. module protection system equipment rooms c. battery rooms d. battery charger rooms	Place the RBVS air handling units with installed direct expansion coils in automatic operation.	i. The temperature and humidity of rooms and areas monitored by the MCR satisfy the design temperature and humidity requirements contained in Table 9.4.2-2.

Table 14.2-20: Reactor Building HVAC System Test # 20 (Continued)

System Level Test #20-3		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify RBVS automatic alignment on a simulated spent fuel pool hi-hi radiation level. ii. Verify The RBVS maintains a negative pressure in the RXB relative to the outside environment while the RBVS is operating in accident alignment. iii. Verify The RWBVS maintains a negative pressure in the RWB relative to the outside environment while the RBVS is operating in accident alignment.	Place the RBVS general area exhaust, RBVS spent fuel pool exhaust, RWBVS exhaust and Annex Building exhaust in automatic operation. Place the RBVS supply in automatic operation. Place the RWBVS supply system in automatic operation. Simulate a Hi-Hi radiation signal in the spent fuel pool exhaust upstream of the spent fuel pool charcoal filter units.	i. The RBVS general area exhaust isolation damper for the spent fuel pool and dry dock area is closed to isolate the spent fuel pool area exhaust flow from the RBVS general exhaust. ii. The RBVS diverts spent fuel pool exhaust flow to charcoal adsorbers and additional HEPAs in the spent fuel pool charcoal filter units. iii. Flow from the RBVS supply fans is reduced to maintain the design negative pressure in the RXB and RWB relative to the outside environment while the RBVS is operating in the off- normal alignment. [ITAAC 03.09.03] (items i thru iii)

Table 14.2-21: Radioactive Waste Building HVAC System Test # 21

Preoperational test is required to be performed once.		
The Radioactive Waste Building HVAC system (RWBVS) is described in Section 9.4.3 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The RWBVS supports the Radioactive Waste Building (RWB) by providing cooling, heating and humidity control to maintain a suitable environment for the safety and comfort of plant personnel.	nonsafety-related	Test #21-1
2. The RWBVS supports the systems located in the RWB by providing cooling, heating and humidity control to maintain a suitable environment for the operation of system components.	nonsafety-related	Test #21-1
3. The RWBVS 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.	nonsafety-related	Test #21-1 (normal Reactor Building HVAC system (RBVS) exhaust alignment) Reactor Building HVAC System Test #20-3 (off-normal RBVS exhaust alignment)
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify an RWBVS air balance has been performed and the RWBVS air balance records have been approved. [This prerequisite is not required for component-level tests.]		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each RWBVS remotely-operated damper can be operated remotely.	Operate each damper from the main control room (MCR) and local control panel (if design has local damper control).	MCR display and local, visual observation indicate each damper fully opens and fully closes.
ii. Verify each RWBVS air-operated damper fails to its safe position on loss of air.	Place each damper in its non-safe position. Isolate and vent air to the damper.	MCR display and local, visual observation indicate each damper fails to its safe position.
iii. Verify each RWBVS air-operated damper fails to its safe position on loss of electrical power to its solenoid.	Place each damper in its non-safe position. Isolate electrical power to its solenoid.	MCR display and local, visual observation indicate each damper fails to its safe position.
iv. Verify RWBVS dampers automatically close on associated smoke or fire signals.	Open each damper actuated by a smoke or fire signal. Initiate an alarm signal for each damper.	MCR display and local, visual observation indicate each damper closes.
v. Verify each required RWBVS fan stops on actuation of its associated fire or smoke alarm.	Initiate an alarm signal for each fan.	MCR display and local, visual observation indicate each fan stops.
vi. Verify each RWBVS pressurization fan starts automatically on the actuation of its associated fire or smoke alarm.	Initiate an alarm signal for each fan.	MCR display and local, visual observation indicate each pressurization fan starts.
vii. Verify the fan speed of each RWBVS variable-speed fan can be manually controlled.	Vary the speed of each fan from the MCR and local control panel (if design has local fan control).	MCR display indicates the speed of each fan varies from minimum to maximum speed.

Table 14.2-21: Radioactive Waste Building HVAC System Test # 21 (Continued)

viii. Verify the standby RWBVS main supply air handling unit (AHU) starts automatically on the stop of the operating RWBVS main supply AHU.	Place an AHU in service. Place the standby AHU in automatic control. Stop the operating recirculation AHU.	MCR display and local, visual observation indicate the standby AHU starts.
ix. Verify each standby RWBVS fan coil unit (FCU) starts automatically on the stop of the operating RWBVS FCU.	Place an FCU in service. Place the standby FCU in automatic control. Stop the operating FCU.	MCR display and local, visual observation indicate the standby FCU starts.
x. Verify each RWBVS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each RWBVS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #21-1		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify the RWB design temperatures and humidity monitored by the MCR are maintained at design temperature and humidity conditions during normal operation. ii. Verify the RWBVS maintains a negative pressure in the RWB relative to the outside environment while the RWBVS is operating in normal alignment.	Place the RWBVS in automatic operation. Place the RBVS in automatic operation.	i. The temperature and humidity of rooms and areas monitored by the MCR satisfy the design temperature and humidity requirements contained in Table 9.4.1-2. ii. MCR display indicates the RWBVS maintains a negative pressure in the RWB relative to the outside environment while operating in the normal operating alignment.

Table 14.2-22: Turbine Building HVAC System Test # 22

Preoperational test is required to be performed once.		
The Turbine Building HVAC system (TBVS) is described in Section 9.4.4 and the function verified by this test is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The TBVS supports the systems located in the Turbine Generator Building (TGB) by providing cooling, heating and humidity control to maintain a suitable environment for the operation of system components.	nonsafety-related	Test #22-1
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each air conditioning unit and condensing unit operates to maintain room temperature.	Start air conditioning and condensing unit.	Air conditioning unit and condensing unit operates to maintain room temperature.
ii. Verify TBVS dampers automatically close on associated smoke or fire signals.	Open each damper actuated by a smoke or fire signal. Initiate an alarm signal for each damper.	Main control room (MCR) display and local, visual observation indicate each damper closes.
iii. Verify each required TBVS fan stops on actuation of its associated fire or smoke alarm.	Initiate an alarm signal for each fan.	MCR display and local, visual observation indicate each fan stops.
iv. Verify the fan speed of each TBVS variable-speed fan can be manually controlled.	Vary the speed of each fan from the MCR and local control panel (if design has local fan control).	MCR display indicates the speed of each fan varies from minimum to maximum speed.
v. Verify each TBVS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each TBVS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #22-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the TGB battery and battery charger room design temperatures are maintained at design temperature and humidity conditions during normal operation.	Place the turbine bypass system battery and battery charger room ventilation units in automatic operation.	The temperature and humidity of TGB battery and battery charger rooms satisfy the temperature and humidity requirements.

Table 14.2-23: Radioactive Waste Drain System Test # 23

Preoperational test is required to be performed once.		
The radioactive waste drain system (RWDS) is described in Section 9.3.3 and the functions verified by this test or another preoperational test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The RWDS supports the Radioactive Waste Building (RWB) by collecting radioactive waste in drain sumps and tanks and transfers it to the liquid radioactive waste system (LRWS) for processing.	nonsafety-related	Test #23-1
2. The RWDS supports the Reactor Building (RXB) by collecting radioactive waste in drain sumps and tanks and transfers it to the LRWS for processing.	nonsafety-related	Test #23-1
3. The RWDS supports the Annex Building (ANB) by collecting radioactive waste in drain sumps and tanks and transfers it to the LRWS for processing.	nonsafety-related	Test #23-1
4. The RWDS supports the ultimate heat sink (UHS) by providing detection and monitoring of leakage through the UHS liner and the dry dock liner.	nonsafety-related	Test #23-2
5. The LRWS supports the RWDS by receiving and processing the effluent from the RWB radioactive waste drain sumps.	nonsafety-related	Test #23-1 Liquid Radioactive Waste System Test #35-2
6. The LRWS supports the RWDS by receiving and processing the effluent from the RXB radioactive waste drain sumps.	nonsafety-related	Test #23-1 Liquid Radioactive Waste System Test #35-2
7. The LRWS supports the RWDS by receiving and processing the effluent from the ANB radioactive waste drain sumps.	nonsafety-related	Test #23-1 Liquid Radioactive Waste System Test #35-2
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each RWDS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each RWDS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each RWDS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each RWDS pump can be started and stopped remotely.	Align the RWDS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops.

Table 14.2-23: Radioactive Waste Drain System Test # 23 (Continued)

v. Verify a local grab sample can be obtained from an RWDS grab sample device indicated on the RWDS piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
vi. Verify each RWDS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each RWDS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #23-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify RWDS pumps start and stop automatically and transfer liquid waste to its design location in the LRWS.	Align each RWDS sump or tank to allow water in a selected sump or tank to be pumped to its design location in the LRWS (as indicated by the RWDS piping and instrumentation diagrams). i. Fill the selected sump or tank until a HI water level is obtained to start the first (primary) pump. ii. Continue filling the sump or tank until a HI-HI level starts the second (alternate) pump. iii. Stop filling the sump or tank to allow the primary and alternate pumps to stop on low level. iv. Refill the sump or tank until the alternate pump starts on HI level.	MCR displays and local, visual observation verifies the following: i. The first pump starts on HI level and transfers water to its design location in the LRWS. ii. The second (alternate) pump starts on HI-HI level. iii. Both primary and alternate pumps stop on LO level. iv. The alternate pump starts on HI level.
System Level Test #23-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify each RWDS equipment drain sump alarms on a fill rate that exceeds the pool leakage detection system (PLDS) leakage rate setpoint.	Fill the selected sump at a rate that exceeds the PLDS leakage rate setpoint.	PCS data indicates the sump fill rate alarmed at the PLDS leakage rate setpoint.

Table 14.2-24: Balance-of-Plant Drain System Test # 24

Preoperational test is required to be performed to support sequence of construction turnover of the balance-of-plant drain system (BPDS) system.		
BPDS system is described in Section 9.3.3 and 11.5.2.2.15 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The BPDS supports the condensate polisher demineralizers, the three cooling tower chemical addition systems, and the demineralized water system reverse osmosis units by providing a means to collect and transfer chemical wastes to either the liquid radioactive waste system (LRWS) or to the utility water system (UWS).	nonsafety-related	Test #24-1 Test #24-7
2. The BPDS supports the two Turbine Buildings (TGBs), the two diesel generators, the auxiliary boiler, the combustion turbine, the Central Utility Building, and the diesel driven firewater pump by providing a means to collect, treat, and transfer the waste water to the either the LRWS or to the UWS.	nonsafety-related	Test #24-1 Test #24-7
3. The BPDS supports the Control Building floor drains by providing a means to collect, treat, and transfer the waste water to the UWS.	nonsafety-related	Test #24-1 Test #24-7
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each BPDS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each BPDS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each BPDS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each BPDS pump can be started and stopped remotely.	Align the BPDS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify the pump speed of each BPDS variable-speed pump can be manually controlled.	Vary the speed of each pump from the MCR and local control panel (if design has local pump control).	MCR display indicates the speed of each pump varies from minimum to maximum speed.

Table 14.2-24: Balance-of-Plant Drain System Test # 24 (Continued)

vi. Verify each BPDS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each BPDS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #24-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify BPDS automatically controlled pumps in sumps and tanks with a fire water removal pump, start and stop automatically and transfer liquid waste to its design location.	Align each BPDS sump or tank to allow water in a selected sump or tank to be pumped to its design location. If the sump fill rate in the following test method is insufficient for automatic start of the alternate pump or fire pump, the primary pump or alternate pump may be temporarily removed from service to allow an increase in the sump level. i. Verify that Pump #1 is set to the primary pump and Pump #2 is set to alternate. Fill the selected sump or tank until a HI water level is obtained to start the primary pump. ii. Continue filling the sump or tank until a HI-HI level starts the alternate pump. iii. Fill the sump or tank until a HI-HI-HI level starts the fire water removal pump. iv. Stop filling the sump or tank to allow the fire water removal pump to stop on HI-HI level. v. Continue (or start) sump or tank dewatering to allow the primary and alternate pumps to stop on LO level. vi. Change pump controls to make Pump #2 the primary pump and Pump #1 the alternate pump, and refill the sump or tank until the primary pump starts on HI level. vii. Continue filling the sump or tank until a HI-HI level starts the alternate pump. Note: Pump #1 and Pump #2 are not the actual names of the pumps, these names are used to differentiate between the two pumps.	MCR displays and local, visual observation verifies the following: i. The primary pump starts on HI level and transfers water to its design location in the LRWS or UWS. ii. The alternate pump starts on HI-HI level. iii. The fire water removal pump starts on HI-HI-HI level. iv. The fire water removal pump stops on HI-HI level. v. Both primary and alternate pumps stop on LO level. vi. The primary pump starts on HI level. vii. The alternate pump starts on HI-HI level.

Table 14.2-24: Balance-of-Plant Drain System Test # 24 (Continued)

System Level Test #24-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the BPDS automatically responds to mitigate a release of radioactivity.	Place a north chemical waste water sump pump in operation. Initiate a real or simulated high radiation signal on the 0A condensate polishing system (CPS) regeneration skid waste effluent. Repeat the test for each pump.	i. The north chemical waste water sump pump stops. ii. North chemical waste collection sump to BPDS collection tank isolation valve is closed. iii. North chemical waste collection sump to LRWS high conductivity waste tank isolation valve is closed. [ITAAC 03.17.02] (i through iii)
System Level Test #24-3		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the BPDS automatically responds to mitigate a release of radioactivity.	Place a south chemical waste water sump pump in operation. Initiate a real or simulated high radiation signal on the 0B CPS regeneration skid waste effluent. Repeat the test for each pump.	i. The pump stops. ii. South chemical waste collection sump to BPDS collection tank isolation valve is closed. iii. South chemical waste collection sump to LRWS high conductivity waste tank isolation valve is closed. [ITAAC 03.18.02] (i through iii)
System Level Test #24-4		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the BPDS automatically responds to mitigate a release of radioactivity.	Place a north waste water sump pump in operation. Initiate a real or simulated high radiation signal in the BPDS north TGB floor drains. Repeat the test for each pump.	i. The north waste water sump pump stops. ii. North waste water sump discharge to BPDS collection tank isolation valve is closed. iii. North waste water sump discharge to LRWS high conductivity waste tank isolation valve is closed. [ITAAC 03.17.02] (i thorugh iii)
System Level Test #24-5		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the BPDS automatically responds to mitigate a release of radioactivity.	Place a south waste water sump pump in operation. Initiate a real or simulated high radiation signal in the BPDS south TGB floor drains. Repeat the test for each pump.	i. The south waste water sump pump stops. ii. South waste water sump discharge to BPDS collection tank isolation valve is closed. iii. South waste water sump discharge to LRWS high conductivity waste tank isolation valve is closed. [ITAAC 03.18.02] (i through iii)

Table 14.2-24: Balance-of-Plant Drain System Test # 24 (Continued)

System Level Test #24-6		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the BPDS automatically responds to mitigate a release of radioactivity.	Place a north waste water sump pump in operation. Initiate a real or simulated high radiation signal in the BPDS auxiliary blowdown cooler condensate. Repeat the test for each pump.	i. The north chemical waste water sump pump stops. ii. North chemical waste collection sump to BPDS collection tank isolation valve is closed. iii. North chemical waste collection sump to LRWS high conductivity waste tank isolation valve is closed. [ITAAC 03.17.02] (i through iii)
System Level Test #24-7		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify BPDS automatically controlled pumps, in sumps and tanks without a fire water removal pump, start and stop automatically and transfer liquid waste to its design location.	Align each BPDS sump or tank to allow water in a selected sump or tank to be pumped to its design location. If the sump fill rate in the following test method is insufficient for automatic start of the alternate pump, the primary pump may be temporarily removed from service to allow an increase in the sump level. i. Verify that Pump #1 is set to the primary pump and Pump #2 is set to alternate. Fill the selected sump or tank until a HI water level is obtained to start the primary pump. ii. Continue filling the sump or tank until a HI-HI level starts the alternate pump. iii. Stop filling the sump or tank to allow the primary and alternate pumps to stop on LO level. iv. Change pump controls to make Pump #2 the primary pump and Pump #1 the alternate pump, and refill the sump or tank until the primary pump starts on HI level. v. Continue filling the sump or tank until a HI-HI level starts the alternate pump. Note: Pump #1 and Pump #2 are not the actual names of the pumps; these names are used to differentiate between the two pumps.	MCR displays and local, visual observation verifies the following: i. The primary pump starts on HI level and transfers water to its design location in the LRWS or UWS. ii. The alternate pump starts on HI-HI level. iii. Both primary and alternate pumps stop on LO level. iv. The primary pump starts on HI level. v. The alternate pump starts on HI-HI level.

Table 14.2-25: Fire Protection System Test # 25

<table>
<tr><td colspan="3">Preoperational test is required to be performed once, and is conducted in accordance with the applicable criteria in codes and standards listed in Table 9.5.1-1.</td></tr>
<tr><td colspan="3">The fire protection system (FPS) is described in Section 9.5.1 and the functions verified by this test are:</td></tr>
<tr><td>System Function</td><td>System Function Categorization</td><td>Function Verified by Test #</td></tr>
<tr><td>1. The FPS supports the following buildings and systems by providing fire prevention, detection, and suppression.
• Reactor Building
• Turbine Building
• Radioactive Waste Building
• Security Buildings
• Annex Building
• Diesel Generator Building
• Administration and Training Building
• Warehouse Building
• Fire Water Building
• switchyard
• site plant cooling structures
• Central Utility Building
• Control Building (CRB)
• 13.8 KV and SWYD system
• medium voltage AC electrical distribution system
• low voltage AC electrical distribution system
• Radioactive Waste Building HVAC system
• normal control room HVAC system
• Reactor Building HVAC system</td><td>nonsafety-related</td><td>Component-level tests</td></tr>
<tr><td>2. The FPS supports the CRB by providing audible and visual alarms to alert operators in the main control room (MCR).</td><td>nonsafety-related</td><td>Component-level test vii</td></tr>
<tr><td colspan="3">Prerequisites
i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.
ii. Verify a pump curve test has been completed for the fire protection pumps.</td></tr>
<tr><td colspan="3">Component Level Tests</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>i. Verify position indication for each FPS manual valve with remote position indication.</td><td>Operate each valve manually.</td><td>MCR display and local, visual observation indicate each valve fully opens and fully closes.</td></tr>
<tr><td>ii. Verify each FPS air-operated valve fails to its safe position on loss of air.</td><td>Place each valve in its non-safe position. Isolate and vent air to the valve.</td><td>MCR display and local, visual observation indicate each valve fails to its safe position.</td></tr>
<tr><td>iii. Verify each FPS air-operated valve fails to its safe position on loss of electrical power to its solenoid.</td><td>Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.</td><td>MCR display and local, visual observation indicate each valve fails to its safe position.</td></tr>
</table>

Table 14.2-25: Fire Protection System Test # 25 (Continued)

iv. Verify each FPS pump can be started and stopped.	Align the FPS to allow for pump operation. i. Start each pump locally. ii. Stop each pump locally.	i. MCR display and local, visual observation indicate each pump starts. Audible and visible water hammer are not observed when the pump starts. ii. MCR display and local, visual observation indicate each pump stops.
v. Verify automatic operation of FPS pumps.	i. Align the FPS and place the FPS pumps in automatic operation to pressurize the system. ii. Stop the jockey pump and simulate a low FPS header pressure to start the electric fire pump. iii. Stop the electric fire pump and simulate a low FPS header pressure to start the diesel fire pump.	Any MCR display or the local, visual observation indicate the following: i. The jockey pump maintains the FPS header greater than or equal to 10 psig above the pressure setting for the automatic start of the electric fire pump. ii. The electric fire pump starts. Audible and visible water hammer are not observed when the pump starts. iii. The diesel pump starts. Audible and visible water hammer are not observed when the pump starts.
vi. Verify each valve with a tamper switch alarms when partially closed.	Partially close each FPS manual valve with a tamper switch to its alarm position (approximately 20 per cent of its total travel distance).	An alarm is received in the MCR when each valve is partially closed.
vii. Verify each smoke and fire detector provides audible and visual alarms and annunciation in the MCR.	Isolate the water supply to each preaction or deluge sprinkler before perming this test to prevent wetting equipment. Simulate a smoke or fire signal to each detector.	The MCR receives an alarm and indication from each smoke and fire detector.
viii. Verify fire pump flow meets its fire protection volumetric flow rate.	Align the FPS for pump operation through the recirculation line. i. Start the electric fire pump. ii. Start the diesel fire pump.	i. The electric fire pump meets its design volumetric flow rate. [ITAAC 03.07.02] ii. The diesel fire pump meets its design volumetric flow rate. [ITAAC 03.07.02]
ix. Verify each suppression system actuation valve opens and alarms in the MCR when signal is received.	Isolate fluid source and simulate actuation signal.	Valve opens and send alarm to MCR.
x. Verify each suppression system flow switch alarms in the MCR.	Isolate fluid source and simulate flow signal.	Flow alarm sent to MCR.
xi. Verify each FPS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each FPS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

Table 14.2-25: Fire Protection System Test # 25 (Continued)

System Level Tests
None

Table 14.2-26: Fire Detection System Test # 26

Preoperational test is required to be performed once, and is conducted in accordance with the applicable criteria in codes and standards listed in Table 9.5.1-1.		
The fire detection system (FDS) is described in Section 9.5.1 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
As described in Test Abstract Table 14.2-25	nonsafety-related	As described in Test Abstract Table 14.2-25
Prerequisites		
As described in Test Abstract Table 14.2-25		
Component Level Tests		
As described in Test Abstract Table 14.2-25		
System Level Tests		
As described in Test Abstract Table 14.2-25		

Table 14.2-27: Main Steam System Test # 27

Preoperational test is required to be performed for each NuScale Power Module.		
The main steam system (MSS) is described in Section 10.3. MSS functions are not verified by this test. The MSS functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The MSS supports the turbine generator system (TGS) by providing steam to the TGS.	nonsafety-related	Turbine Generator System Test #33-2 Ramp Change in Load Test #100
2. The MSS supports the containment system by providing secondary isolation of the main steam lines.	nonsafety-related	Module Protection System Test #63-6
3. The MSS supports the decay heat removal system (DHRS) by providing a backup means for required boundary conditions for DHRS operation.	nonsafety-related	Module Protection System Test #63-6
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each MSS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control)	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each MSS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each MSS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify automatic operation of MSS extraction steam to protect the main turbine.	Initiate a simulated signal for the following system conditions. i. feedwater heater high level ii. turbine trip	Any remote display or local verification indicates the following: i. extraction steam block valve closes ii. extraction steam non-return check valve closes
v. Verify each MSS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each MSS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-28: Feedwater System Test # 28

Preoperational test is required to be performed for each NuScale Power Module.		
The feedwater system (FWS) is described in Section 10.4.7; Section 9.2.6 (condensate storage tank); Section 10.4.1 (condenser); FWS functions are not verified by FWS tests. FWS functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The FWS supports the condensate polishing system (CPS) by providing water for CPS rinse and CPS resin transfer.	nonsafety-related	Condensate Polishing System Test #30-1
2. The FWS supports the TGS by cooling superheated steam in the gland steam desuperheater prior to the steam entering the gland seals.	nonsafety-related	Turbine Generator System Test #33-1 Ramp Change in Load Test #100
3. The FWS supports the containment system (CNTS) by supplying feedwater to the steam generators.	nonsafety-related	Turbine Generator System Test #33-1 Ramp Change in Load Test #100
4. The FWS supports the TGS by cooling superheated turbine bypass steam in the turbine bypass desuperheater prior to the steam entering the main condenser.	nonsafety-related	Turbine Generator System Test #33-1 100 Percent Load Rejection Test #103
5. The FWS supports the TGS by accepting turbine bypass steam into the main condenser.	nonsafety-related	Turbine Generator System Test #33-1 100 Percent Load Rejection Test #103
6. The FWS supports the CNTS by providing secondary isolation of the feedwater lines.	nonsafety-related	Module Protection System Test #63-6
7. The FWS supports the decay heat removal system (DHRS) by providing secondary isolation of the feedwater lines, ensuring required boundary conditions for DHRS operation.	nonsafety-related	Module Protection System Test #63-6
Prerequisites i.Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii.Verify a pump curve test has been completed for the FWS pumps.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each FWS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each FWS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each FWS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each FWS condensate pump can be started and stopped remotely.	Align the FWS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.

Table 14.2-28: Feedwater System Test # 28 (Continued)

v. Verify the pump speed of each FWS variable-speed pump can be manually controlled.	Align the FWS to provide a flow path to operate a selected FWS variable-speed pump. Vary the FWS pump speed from minimum to maximum speed from the MCR.	MCR display indicates the speed of each variable speed pump obtains both minimum and maximum pump speeds. Audible and visible water hammer are not observed when the pump starts.
vi. Verify the condensate standby pump automatically starts to protect plant equipment.	Align the FWS to allow for pump operation. Place a pump in service. Initiate a simulated low pump header pressure low signal.	MCR display and local, visual observation indicate the standby pump starts. Audible and visible water hammer are not observed when the pump starts.
vii. Verify the feedwater standby pump automatically starts to protect plant equipment.	Align the FWS to allow for pump operation. Place a pump in service. Initiate a simulated low pump header pressure low signal.	MCR display and local, visual observation indicate the standby pump starts. Audible and visible water hammer are not observed when the pump starts.
viii. Verify condensate pump low flow protection and short cycle automatic operation.	i. Align the FWS for automatic short cycle cleanup. Place a condensate pump in operation. ii. Manually throttle a valve in the pump flow path until the flow rate reaches the pump minimum flow setpoint. iii. Open the throttled valve.	MCR displays and local, visual observation verifies the following: i. The short cycle flow is automatically maintained by the short cycle cleanup flow control valve. ii. The condensate pump minimum flow valve is open. iii. The condensate pump minimum flow valve is closed.
ix. Verify feedwater pump low flow protection.	i. Align the FWS for automatic long cycle cleanup. Place a condensate pump in operation. ii. Manually throttle a valve in the pump flow path until the flow rate reaches the feedwater pump minimum flow setpoint. iii. Open the throttled valve.	MCR displays and local, visual observation verifies the following: i. The long cycle flow is automatically maintained by the long cycle cleanup flow control valve. ii. The feedwater pump minimum flow valve is open. iii. The feedwater pump minimum flow valve is closed.
x. Verify a local grab sample can be obtained from an FWS grab sample device.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
xi. Verify each FWS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each FWS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-29: Feedwater Treatment System Test # 29

Preoperational test is required to be performed for the 0A NuScale Power Modules (NPMs) and for the 0B NPMs.		
The feedwater treatment system (FWTS) is described in Section 10.4.11 and the function verified by this test and power ascension testing is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The FWTS supports the feedwater system by controlling and maintaining feedwater chemistry.	nonsafety-related	Component-level tests Primary and Secondary Chemistry Test #79
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each FWTS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each FWTS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each FWTS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each FWTS pump can be started and stopped remotely and locally (if designed).	Align the FWTS to allow for pump operation. Stop and start each remotely-controlled pump from the MCR. Stop and start each locally-controlled pump locally.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify the speed of each FWTS variable-speed pump can be manually controlled.	Vary the speed of each pump from the MCR and local control panel (if design has local pump control).	MCR display indicates pump speed varies from minimum to maximum speed.
vi. Verify each FWTS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each FWTS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-30: Condensate Polishing System Test # 30

Preoperational test is required to be performed once.		
The condensate polishing system (CPS) is described in Section 10.4.6. The CPS and other system functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The CPS supports the feedwater system (FWS) by regenerating the resin that purifies the condensate.	nonsafety-related	Test #30-1
2. The FWS supports the CPS by providing water for CPS rinse and CPS resin transfer.	nonsafety-related	Test #30-1
3. The auxiliary boiler system (ABS) supports the CPS by supplying steam for resin regeneration.	nonsafety-related	Test #30-1
Prerequisites: Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each CPS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each CPS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each CPS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each CPS pump can be started and stopped remotely and locally (if designed).	Align the CPS to allow for pump operation. Stop and start each remotely-controlled pump from the MCR. Stop and start each locally-controlled pump locally.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify the speed of each CPS variable-speed pump can be manually controlled.	Vary the speed of each pump from the MCR and local control panel (if design has local pump control)	MCR display indicates pump speed varies from minimum to maximum speed.
vi. Verify a local grab sample can be obtained from a CPS grab sample device.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
vii. Verify each CPS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each CPS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

Table 14.2-30: Condensate Polishing System Test # 30 (Continued)

System Level Test #30-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the CPS automatically completes resin regeneration.	Align the FWS to support CPS resin regeneration. Align the ABS to support CPS resin regeneration. i. Automatically transfer the test resin bed from a condensate polisher to the CPS regeneration skid. ii. Initiate an automatic regeneration of the resin. iii. Automatically transfer the test resin bed from the CPS regeneration skid to a condensate polisher.	i. The resin transferred to the regeneration skid. ii. The CPS regeneration cycle completed successfully. iii. The resin transferred to a condensate polisher. iv. ABS steam maintains hot water heater outlet temperature at design setpoint during resin regeneration.

Table 14.2-31: Feedwater Heater Vents and Drains System Test # 31

Preoperational test is required to be performed for each NuScale Power Module.		
The feedwater heater vents and drains system (HVDS) is described in Section 10.4.7. and the functions verified by this test and power ascension testing are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The HVDS supports the feedwater system (FWS) by venting the feedwater heaters.	nonsafety-related	Component level tests
2. The HVDS supports the FWS by controlling level in the shell sides feedwater heaters.	nonsafety-related	Component level tests Ramp Change in Load Demand Test #100
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each HVDS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each HVDS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each HVDS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify automatic operation of HVDS valves to protect the turbine on turbine trip.	Initiate a simulated turbine trip.	Any remote display or local verification indicates the following: i. Low, intermediate and high pressure feedwater heater extraction steam supply valves are closed. ii. Low, intermediate and high pressure feedwater heater air assisted check valves are closed. iii. Low, intermediate and high pressure feedwater heater extraction steam dump valves are open.
v. Verify automatic operation of HVDS valves to protect turbine on high feedwater heater level.	Initiate a simulated signal for the following system conditions. i. Low pressure feedwater heater high level. ii. Intermediate pressure feedwater heater high level. iii. High pressure feedwater heater high level	Any remote display or local verification indicates the following: i. Low pressure feedwater heater extraction steam supply valve and low pressure feedwater heater extraction steam dump valve are open. ii. Intermediate pressure feedwater heater extraction steam supply valve and intermediate pressure feedwater heater extraction steam dump valve are open. iii. High pressure feedwater heater extraction steam supply valve and high pressure feedwater heater extraction steam dump valve are open.

Table 14.2-31: Feedwater Heater Vents and Drains System Test # 31 (Continued)

vi. Verify each HVDS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each HVDS system transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-32: Condenser Air Removal System Test # 32

Preoperational test is required to be performed for each NuScale Power Module.		
The condenser air removal system (CARS) is described in Section 10.4.2 and the functions verified by this test and power ascension testing are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The CARS supports the feedwater system (FWS) by removing air and non-condensable gases from the main condenser.	nonsafety-related	Test #32-1
2. The circulating water system (CWS) supports the FWS by removing heat from the main condenser.	nonsafety related	Test #32-1 Ramp Change in Load Demand Test #100
3. The auxiliary boiler system (ABS) supports the turbine generator by supplying gland seal steam.	nonsafety-related	Test #32-1
4. The ABS supports the FWS by supplying steam to the condenser for sparging when necessary.	nonsafety-related	Test #32-1
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each CARS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each CARS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each CARS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each CARS pump can be started and stopped remotely.	Align the CARS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify a CARS standby pump automatically starts to protect plant equipment.	Align the CARS to allow for pump operation. Place a pump in service. Initiate a simulated main condenser high pressure.	MCR display and local, visual observation indicate the standby pump starts. Audible and visible water hammer are not observed when the pump starts.
vi. Verify a local integrated grab sample can be obtained from the CARS grab sample device.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is obtained.
vii. Verify each CARS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each CARS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

Table 14.2-32: Condenser Air Removal System Test # 32 (Continued)

System Level Test #32-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the CARS can maintain main condenser vacuum pressure.	Place the ABS in automatic control to supply gland seal steam. Place the FWS in automatic control to condense the gland seal steam in the gland exhaust condenser. Place the CWS in automatic control to provide cooling to the main condenser. i. Place the CARS in service to establish vacuum in the main condenser. ii. Open the feedwater sparge isolation valves to provide steam sparging to the main condenser.	i. Maintain main condenser design vacuum pressure. ii. The ABS is capable of providing sparging steam to the main condenser as indicated by steam flow. iii. The ABS is capable of supplying gland seal steam to the turbine generator at design pressures.

Table 14.2-33: Turbine Generator System Test # 33

Preoperational test is required to be performed for each NuScale Power Module (NPM).		
The turbine generator system (TGS) is described in Sections 10.2, 10.4.3, and 10.4.4. The TGS and other functions verified by this test and power ascension testing are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The TGS supports the main steam system (MSS) by providing steam bypass from the MSS to the main condenser.	nonsafety-related	Test #33-1 100 Percent Load Rejection Test #103
2. The module heatup system (MHS) supports the chemical and volume control system (CVCS) by adding heat to primary coolant.	nonsafety-related	Test #33-1
3. The CVCS supports the reactor coolant system (RCS) by heating primary coolant.	nonsafety-related	Test #33-1
4. The auxiliary boiler system (ABS) supports the MHS by supplying steam for heating reactor coolant at startup and shutdown.	nonsafety-related	Test #33-1
5. The feedwater system (FWS) supports the containment system by supplying feedwater to the steam generators (SGs).	nonsafety-related	Test #33-1 Ramp Change in Load Test #99
6. The FWS supports the TGS by cooling superheated turbine bypass steam in the turbine bypass desuperheater prior to the steam entering the main condenser.	nonsafety-related	Test #33-1 100 Percent Load Rejection Test #103
7. The FWS supports the TGS by accepting turbine bypass steam into the main condenser.	nonsafety-related	Test #33-1 100 Percent Load Rejection Test #103
8. The FWS supports the TGS by cooling superheated steam in the gland steam desuperheater prior to the steam entering the gland seals.	nonsafety-related	Test #33-1 Ramp Change in Load Test #100
9. The CVCS supports emergency core cooling system (ECCS) valves by providing water to reset the ECCS valves.	nonsafety-related	Test #33-1
10. The MSS supports the TGS by providing steam to the TGS.	nonsafety-related	Test #33-2 Ramp Change in Load Test #100
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. The following prerequisites are not required for component testing: ii. Verify Test #32-1 has been completed to verify the condenser air removal system (CARS) can maintain main condenser vacuum pressure (reference test 14.2-32). iii. The SG feedwater flush is complete. iv. The CARS is automatically maintaining main condenser vacuum. v. Initial RCS temperature must be approximately 200°F to allow for hot functional testing to obtain data at an RCS temperature of 200°F and above. vi. The NPM and supporting systems are aligned to increase RCS temperature and pressure.		

Table 14.2-33: Turbine Generator System Test # 33 (Continued)

Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each TGS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each TGS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each TGS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each TGS lube oil pump can be started and stopped remotely.	Align the TGS to allow for main lube oil, auxiliary lube oil, and emergency pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify the TGS exhaust hood is protected against high temperature.	Initiate a simulated high exhaust hood temperature.	Any remote display or the local, visual observation indicates the exhaust hood spray valve is open.
vi. Verify TGS lubricating oil flow capability by automatic start of the auxiliary lube oil pump.	Align the TGS to allow for main lube oil and auxiliary lube oil pump operation. Place the TGS main oil pump in normal service. Place the auxiliary oil pump in standby. Simulate a TGS auxiliary oil pump start.	MCR display and local, visual observation indicate the auxiliary oil pump starts. Audible and visible water hammer are not observed when the pump starts.
vii. Verify TGS lubricating oil flow capability by automatic start of the emergency DC lube oil pump.	Align the TGS to allow for auxiliary lube oil pump and emergency lube oil pump operation. Place the turbine generator auxiliary oil pump in normal service. Simulate a turbine generator emergency oil pump start or simulate a loss of AC power.	MCR displays and local, visual observation indicate the TGS emergency oil pump starts. Audible and visible water hammer are not observed when the pump starts.
viii. Verify the turbine stop valve and turbine control valves close on turbine overspeed.	i. a. Simulate an overspeed trip signal from the turbine overspeed emergency trip system. b. Record the stroke times of the turbine stop valve and the turbine control valves. ii. a. Simulate an overspeed trip signal from the governor overspeed detection circuit. b. Record the stroke times of the turbine stop valve and the turbine control valves.	i. a. The turbine stop valve and turbine control valves close. b. Each turbine stop valve and turbine control valve close stroke time is within design limits. ii. a. The turbine stop valve and turbine control valves close. b. Each turbine stop valve and turbine control valve close stroke time is within design limits.
ix. Verify a local grab sample can be obtained from the gland seal exhauster discharge grab sample device.	Place the system in service to allow flow through the grab sample device.	A local grab sample is obtained.
x. Verify the turbine can be manually tripped.	Manually trip the turbine from an operator workstation in the MCR.	The turbine stop valve and turbine control valves close.

Table 14.2-33: Turbine Generator System Test # 33 (Continued)

xi. Verify each TGS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each TGS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #33-1		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify the CVCS is capable of supplying water at sufficient pressure to close the ECCS valves. ii. Verify the MHS is capable of heating the RCS to a temperature sufficient to obtain criticality. iii. Verify the MHS is capable of heating the RCS to establish natural circulation flow sufficient to obtain criticality. iv. Verify the TGS automatically controls turbine bypass flow to the main condenser. v. Verify the FWS automatically controls flow to the SGs to maintain SG inventory. vi. Verify the FWS automatically cools the TGS bypass steam in the main steam desuperheater. vii. Verify a local grab sample can be obtained from an MHS system grab sample device. viii. Verify the FWS automatically cools the TGS gland steam in the gland steam desuperheater. ix. Verify hotwell level is automatically controlled while receiving bypass steam.	Close the ECCS valves. Align the plant to cool the RCS via the TGS bypass system. Warm main steam lines. Place the TGS steam bypass valve in automatic control. Place the feedwater regulating valve in steam generator inventory control. Place the MHS and the CVCS in automatic control to heat the RCS. Place the ABS high-pressure system in automatic control to heat the MHS heat exchanger from RCS ambient temperature to the highest temperature achievable by MHS heating. Align the FWS to cool the gland seal steam desuperheater.	i. CVCS pressure is sufficient as indicated by closure of the ECCS valves. ii. a. CVCS supply remains in a sub-cooled state while heating the RCS using the module heatup system as verified by CVCS temperature and pressure. b. RCS temperature is sufficient to obtain criticality. iii. RCS natural circulation flow is sufficient to obtain criticality. iv. The TGS bypass flow maintains steam pressure at setpoint. v. The feedwater flow to the steam generator is maintained at setpoint. vi. The cooled TGS bypass temperature is maintained at setpoint. vii. A local grab sample is successfully obtained at RCS normal operating temperature and pressure. viii. The cooled gland seal steam temperature is maintained at setpoint. ix. Hotwell level is maintained at setpoint while receiving bypass steam.
System Level Test #33-2 This test may be performed after the completion of Test 33-1 when the RCS is at normal operating pressure and the RCS has achieved the maximum temperature achievable by warming the RCS using MHS heating.		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the maximum main turbine speed that can be obtained using the MHS to heat the RCS.	Place the main turbine in service as follows: i. Ensure the RCS is at normal operating pressure and at maximum temperature achievable by warming the RCS using MHS heating. ii. Place turbine on turning gear with seal steam in service. iii. Warm up turbine to required temperature. iv. Increase main turbine speed.	The maximum main turbine speed is obtained.

Table 14.2-34: Turbine Lube Oil Storage System Test # 34

Preoperational test is required to be performed once for the 0A NuScale Power Modules (NPMs) and once for and the 0B NPMs.		
The turbine lube oil storage system (TLOSS) is described in Section 10.2.2.1.3 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The TLOSS supports the turbine generator system (TGS) by supplying lube oil storage for clean and dirty lube oil for protection of plant assets	nonsafety-related	Component-level tests
2. The TLOSS supports the TGS by receiving and purifying lube oil from the turbine generator lube oil reservoir for protection of plant assets.	nonsafety-related	Component-level tests
3. The TLOSS supports the TGS by providing clean lube oil makeup to the turbine generator lube oil reservoir.	nonsafety-related	Component-level tests
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each TLOSS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each TLOSS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each TLOSS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each TLOSS pump can be started and stopped remotely and locally (if designed).	Align the TLOSS system to allow for pump operation. Stop and start each remotely-controlled pump from the MCR. Stop and start each locally-controlled pump locally.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify each TLOSS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each TLOSS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-35: Liquid Radioactive Waste System Test # 35

Preoperational test is required to be performed once.		
The liquid radioactive waste system (LRWS) is described in Section 11.2 and 11.5.2.1.5 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The LRWS supports the solid radioactive waste system (SRWS) by receiving and processing liquid radioactive waste from the SRWS dewatering skid.	nonsafety-related	Test #35-2 Component-level test xi Solid Radioactive Waste System Test #37-7
2. The LRWS supports the spent fuel pool cooling system (SFPCS) by receiving contaminated pool water to aid in the removal of titrated water or boron. Treated liquid radwaste has the option to return to the pool as makeup.	nonsafety-related	Test #35-2 Component-level tests
3. The LRWS supports the chemical and volume control system (CVCS) by receiving and processing primary coolant from CVCS letdown.	nonsafety-related	Test #35-2 Chemical and Volume Control System Test #38-1
4. The LRWS supports the radioactive waste drain system (RWDS) by receiving and processing the effluent from the Radioactive Waste Building radioactive waste drain sumps.	nonsafety-related	Test #35-2 Radioactive Waste Drain System Test #23-1
5. The LRWS supports the RWDS by receiving and processing the effluent from the Reactor Building radioactive waste drain sumps.	nonsafety-related	Test #35-2 Radioactive Waste Drain System Test #23-1
6. The LRWS supports the RWDS by receiving and processing the effluent from the Annex Building radioactive waste drain sumps.	nonsafety-related	Test #35-2 Radioactive Waste Drain System Test #23-1
7. LRWS supports the CVCS by receiving and processing the noncondensable gases and vapor from the pressurizer.	nonsafety-related	Test #35-1
8. LRWS supports the pool surge control system (PSCS) by processing any fluid collected in the drain sump of the PSCS dike.	nonsafety-related	Test #35-2 Component-level tests
9. The nitrogen distribution system supports the LRWS by providing nitrogen for purging of the LRWS.	nonsafety-related	Test #35-1 Nitrogen Distribution System Test #15 component-level tests
Prerequisites i. Required ANSI/ANS-55.6 construction testing has been completed. ii. Verify an instrument calibration has been completed, with approved records and within calibration due dates, for instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each LRWS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control)	MCR display and local, visual observation indicate each valve fully opens and fully closes.

Table 14.2-35: Liquid Radioactive Waste System Test # 35 (Continued)

ii. Verify each LRWS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each LRWS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each LRWS pump can be started and stopped remotely.	Align the LRWS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify the speed of each LRWS variable-speed pump can be manually controlled.	Align the LRWS to provide a flow path to operate a selected pump. Vary the LRWS pump speed from minimum to maximum from the MCR.	MCR display indicates the speed of each obtains both minimum and maximum pump speeds.
vi. Verify radiation isolation on discharge to the utility water discharge basin high radiation, low dilution flow or underground pipe break.	Initiate the following a real or simulated signals: i. LRWS discharge to the utility water discharge basin high radiation signal. ii. LRWS discharge to the utility water discharge basin low dilution flow signal. iii. LRWS discharge to the utility water discharge basin low guard pipe pressure signal.	MCR display and local, visual observation indicate the following: i. The LRWS discharge to the utility water discharge basin isolation valves close. ii. The LRWS discharge to the utility water discharge basin isolation valves close. iii. The LRWS discharge to the utility water discharge basin isolation valves close. [ITAAC 03.09.07] (items i through iii)
vii. Verify tank valves operate to ensure uninterrupted waste receiving.	Simulate an inservice tank high level signal for each of the following tanks: low-conductivity waste (LCW) collection tank A and B high-conductivity waste (HCW) collection tank A and B LCW sample tank A and B HCW sample tank A and B	MCR display and local, visual observation indicate the inservice tank fill valve is closed and the standby tank fill valve is open.
viii. Verify degasifier valves operate to ensure uninterrupted waste receiving.	i. Initiate a simulated high degasifier level signal. ii. Initiate a simulated high degasifier pressure signal.	i and ii. MCR display and local, visual observation indicate the inservice degasifier fill valve is closed and the standby degasifier fill valve is open.

Table 14.2-35: Liquid Radioactive Waste System Test # 35 (Continued)

ix. Verify LRWS pumps automatically operate to prevent tank overflow.	Align the LRWS to allow each of the following LRWS transfer pumps to automatically transfer effluent to one of its design locations. Degasifier transfer pump A and B LCW collection tank transfer pump A and B HCW collection tank transfer pump A and B LCW sample tank transfer pump A and B HCW sample tank transfer pump A and B Detergent waste collection tank transfer pump Demineralized water break tank transfer pump i. Simulate a HI HI level signal in each of the above tanks. ii. Simulate a low level signal in each of the above tanks.	MCR displays and local, visual observation indicate the following: i. The transfer pump starts and transfers effluent to its design location. ii. The transfer pump stops.
x. Verify a local grab sample can be obtained from a LRWS grab sample device indicated on the LRWS piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
xi. Verify SRWS dewatering skid effluent can be transferred to LRWS HCW collection tanks.	Align SRWS dewatering skid discharge to one of the LRWS high-conductivity waste collection tanks. Fill the SRWS dewatering skid high integrity container to above the low level pump stop setpoint. Start the SRWS dewatering skid diaphragm pump.	SRWS dewatering skid effluent is transferred to the LRWS high-conductivity waste collection tank. The SRWS dewatering skid diaphragm pump is stopped.
xii. Verify each LRWS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each LRWS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #35-1 This test should be performed after the completion of Test 33-1 when the reactor coolant system (RCS) is at normal operating pressure and the RCS has achieved the maximum temperature achievable by warming the RCS using module heatup system heating.		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify LRWS can process a gaseous waste stream.	Align LRWS to receive pressurizer gaseous waste from the pressurizer during hot functional testing. Process the pressurizer gaseous waste through the LRWS degasifier. Purge the degasifier with nitrogen following operation.	i. The LRWS degasifier removes condensable gases and vents waste to the Reactor Building HVAC system or gaseous radioactive waste system. ii. The LRWS degasifier liquid transfer pumps transfer the liquid condensate waste to the low conductivity waste collection tanks. iii. LRWS degasifier is purged with nitrogen.

Table 14.2-35: Liquid Radioactive Waste System Test # 35 (Continued)

System Level Test #35-2		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify LRWS can process a liquid waste stream.	Align LRWS to receive liquid waste from a liquid waste stream. i. Process the liquid waste stream through the LCW waste process. ii. Process the liquid waste stream through the HCW process.	The waste treatment streams are successfully processed through the following processes: • filtration • tubular filtration skid • LCW processing skid • HCW processing skid • drum dryer skid • demineralization • transfer to LCW or HCW sample tanks • transfer from LCW or HCW sample tanks to the utility water system discharge basin.

Table 14.2-36: Gaseous Radioactive Waste System Test # 36

Preoperational test is required to be performed once.		
The gaseous radioactive waste system (GRWS) is described in Section 11.3 and 11.5.2.2.6 and the functions verified by this test or another preoperational test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The GRWS supports the liquid radioactive waste system (LRWS) by receiving and / or collecting potentially radioactive and hydrogen-bearing waste gases which require processing prior to release to the environment.	nonsafety-related	Test #36-1
2. The GRWS supports the containment evacuation system (CES) by receiving and / or collecting potentially radioactive and hydrogen-bearing waste gases which require processing prior to release to the environment.	nonsafety-related	Test #36-1 Containment Evacuation System Test #41-2
3. The nitrogen distribution system (NDS) supports the GRWS by providing nitrogen for purging of the GRWS.	nonsafety-related	Test #36-1 Nitrogen Distribution System Test #15 component-level tests
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each GRWS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each GRWS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each GRWS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify GRWS valves automatically operate to maintain vessel volume.	i. Initiate a real or simulated high GRWS moisture separator level. ii. Initiate a real or simulated low GRWS moisture separator level.	MCR display and local, visual observation indicate the following: i. The moisture separator drain valve is open. ii. The moisture separator drain valve is closed.
v. Verify GRWS inlet isolation valves automatically close and nitrogen purge valve opens on high inlet stream oxygen concentration.	Simulate a GRWS inlet stream oxygen concentration high signal.	MCR display and local, visual observation indicate the following: i. The inlet stream isolation valves are closed. ii. The nitrogen purge valve is open.
vi. Verify GRWS isolates upon loss of Radioactive Waste Building HVAC system (RWBVS) exhaust flow.	Simulate a loss of RWBVS exhaust flow.	MCR display and local, visual observation indicate the GRWS isolation valves are closed.

Table 14.2-36: Gaseous Radioactive Waste System Test # 36 (Continued)

vii. Verify radiation isolation of GRWS charcoal decay beds upon detection of decay bed discharge flow high radiation level.	i. Initiate a real or simulated GRWS train A decay bed discharge flow high radiation signal. ii. Initiate a real or simulated GRWS train B decay bed discharge flow high radiation signal.	MCR display and local, visual observation indicate the following: i. GRWS train A charcoal decay bed discharge isolation valve is closed. [ITAAC 03.09.04] ii. GRWS train B charcoal decay bed discharge isolation valve is closed. [ITAAC 03.09.04]
viii. Verify radiation isolation of GRWS discharge to the Reactor Building HVAC system (RBVS) exhaust upon detection of a high radiation level.	Initiate a real or simulated GRWS discharge to the RBVS exhaust high radiation signal.	MCR display and local, visual observation indicate the GRWS discharge to the RBVS exhaust isolation valves are closed. [ITAAC 03.09.04]
ix. Verify a local grab sample can be obtained from a GRWS grab sample device indicated on the GRWS piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
x. Verify each GRWS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each GRWS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #36-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify GRWS can process a gaseous waste stream and nitrogen stream.	i. Align GRWS to receive gaseous waste from a gaseous waste stream. Process the gaseous waste stream through the gaseous waste process. ii. Align GRWS charcoal drying heater to receive nitrogen from NDS. Process nitrogen through the charcoal drying process.	i. The gaseous waste stream is successfully processed through the following processes: • gas cooler • moisture separator • charcoal guard bed • charcoal decay beds • Radioactive Waste Building exhaust ii. Nitrogen is successfully processed through the charcoal drying heater.

Table 14.2-37: Solid Radioactive Waste System Test # 37

Preoperational test is required to be performed once.		
The solid radioactive waste system (SRWS) is described in Section 11.4 and the functions verified by this test or another preoperational test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The SRWS supports the liquid radioactive waste system (LRWS) by receiving spent resin and carbon bed from LRWS processing skids.	nonsafety-related	Test #37-1 Test #37-4 Test #37-6
2. The SRWS supports the chemical and volume control system (CVCS) by receiving spent resin from CVCS ion exchange vessels.	nonsafety-related	Test #37-2 Test #37-5
3. The SRWS supports the pool cleanup system (PCUS) by receiving spent resin and sludge from PCUS ion exchange vessels.	nonsafety-related	Test #37-3 Test #37-5
4. The SRWS supports the normal control room HVAC system (CRVS) by receiving exhausted HEPA filters to be compacted and shipped off site.	nonsafety-related	Test #37-8
5. The SRWS supports the Radioactive Waste Building HVAC system by receiving exhausted HEPA filters to be compacted and shipped off site.	nonsafety-related	Test #37-8
6. The SRWS supports the Reactor Building HVAC system (RBVS) by receiving exhausted HEPA filters and charcoal bed from RBVS and CRVS, to be compacted and shipped off site.	nonsafety-related	Test #37-8
7. The SRWS supports the gaseous radioactive waste system by receiving contaminated or exhausted charcoal beds, packaging the waste in approved containers and shipping it to a licensed facility.	nonsafety-related	Test #37-8
8. The SRWS supports portions of the Annex Building HVAC system by receiving, compacting, packaging, and storing exhausted HEPA and charcoal filters for storage and shipment offsite.	nonsafety-related	Test #37-8
9. The LRWS supports the SRWS by receiving and processing liquid radioactive waste from the SRWS dewatering skid.	nonsafety-related	Test #37-7 Liquid Radioactive Waste System Test #35-2
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each SRWS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.

Table 14.2-37: Solid Radioactive Waste System Test # 37 (Continued)

ii. Verify each SRWS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each SRWS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each SRWS pump can be started and stopped remotely.	Align the SRWS to allow for pump operation. Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify the speed of each SRWS variable-speed pump can be manually controlled.	Align the SRWS to provide a flow path to operate a selected pump. Vary the SRWS pump speed from minimum to maximum from the MCR.	MCR display indicates the speed of each obtains both minimum and maximum pump speeds.
vi. Verify each SRWS transfer pump automatically stops to protect the pump.	Align the SRWS to allow for transfer pump operation. Place a transfer pump in service. Initiate a simulated tank low level signal.	MCR display and local, visual observation indicate each transfer pump stopped.
vii. Verify a local grab sample can be obtained from a SRWS grab sample device indicated on the SRWS piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
viii. Verify each SRWS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each SRWS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #37-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify spent resin from the LRWS demineralizers can be transferred to the SRWS phase separator tanks.	Align the LRWS and SRWSs to transfer LRWS demineralizer resin to a SRWS phase separator tank. Start a phase separator transfer pump.	The waste management control room (WMCR) displays and local, visual observation verifies LRWS demineralizer resins transferred to a SRWS phase separator tank.
System Level Test #37-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify spent resin from the CVCS ion exchangers can be transferred to the SRWS spent resin storage tanks.	Align the CVCS and SRWSs to transfer CVCS ion exchanger resin to a SRWS spent resin storage tank. Start a SRWS spent resin storage tank transfer pump.	WMCR displays and local, visual observation verifies CVCS ion exchanger resin transferred to a SRWS spent resin storage tank.
System Level Test #37-3		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify spent resin from the PCUS demineralizers can be transferred to the SRWS spent resin storage tanks.	Align the PCUS and SRWSs to transfer PCUS demineralizer resin to a SRWS spent resin storage tank. Start a SRWS spent resin storage tank transfer pump.	WMCR displays and local, visual observation verifies PCUS demineralizer resins transferred to a SRWS spent resin storage tank.

Table 14.2-37: Solid Radioactive Waste System Test # 37 (Continued)

System Level Test #37-4		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify spent resin from the SRWS phase separator tanks can be transferred to a dewatering station high integrity container (HIC).	Align a SRWS phase separator tank and the SRWS dewatering station to transfer spent resin to the dewatering station HIC using the service air system (SAS). Open SAS isolation valve to the SRWS phase separator tank.	WMCR displays and local, visual observation verifies phase separator tank resins are transferred to a dewatering station HIC.
System Level Test #37-5		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify spent resin from the SRWS spent resin storage tanks can be transferred to a dewatering station HIC.	Align an SRWS spent resin storage tank and the SRWS dewatering station to transfer spent resin to the dewatering station HIC using SAS air. Open SAS isolation valve to the spent resin storage tank.	WMCR displays and local, visual observation verifies spent resin storage tank resins are transferred to a dewatering station HIC.
System Level Test #37-6		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify granulated activated charcoal from the LRWS granulated activated charcoal filter can be transferred to a dewatering station HIC.	Align a LRWS and SRWS to granulated activated charcoal to the dewatering station HIC using the clean in place system.	WMCR displays and local, visual observation verifies spent resin storage tank resins are transferred to a dewatering station HIC.
System Level Test #37-7		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the dewatering skid pump removes standing water in the HIC with spent resin in the dewatering station HIC.	Align the dewatering skid pump to a LRWS high conductivity waste tank and start the dewatering skid pump.	Free-standing water in the HIC has been removed.
System Level Test #37-8		
Test Method	**Acceptance Criteria**	
Verify the SRWS waste compactor compacts solid radioactive waste.	Place solid radioactive waste in compactor and start compactor.	The waste has been compacted.

Table 14.2-38: Chemical and Volume Control System Test # 38

Preoperational test is required to be performed for each NuScale Power Module.		
The chemical and volume control system (CVCS) is described in Section 9.3.4 and 11.5.2.2.11 and the functions verified by this test, other preoperational tests and power ascension testing are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The CVCS supports the reactor coolant system (RCS) by providing primary coolant makeup.	nonsafety-related	Test #38-1 Ramp Change in Load Demand Test #100
2. The CVCS supports the RCS by providing primary coolant letdown.	nonsafety-related	Test #38-1 Ramp Change in Load Demand Test #100
3. The CVCS supports the RCS by providing pressurizer spray flow for RCS pressure control.	nonsafety-related	Test #38-2 Ramp Change in Load Demand Test #100
4. The CVCS supports the RCS by changing the boron concentration of the primary coolant.	nonsafety-related	Test #38-3
5. The boron addition system (BAS) supports the CVCS by providing uniformly mixed borated water on demand.	nonsafety-related	Test #38-3
6. The liquid radioactive waste system (LRWS) supports the CVCS by receiving and processing primary coolant from CVCS letdown.	nonsafety-related	Test #38-1 Liquid Radioactive Waste System Test #35-2
The CVCS functions verified by other tests are:		
The CVCS supports emergency core cooling system (ECCS) valves by providing water to reset the ECCS valves.	nonsafety-related	Turbine Generator System Test #33-1
The CVCS supports the RCS by heating primary coolant.	nonsafety-related	Turbine Generator System Test #33-1
The CVCS supports the RCS by isolating dilution sources.	safety-related	Module Protection System Test #63-6
The CVCS supports the RCS by providing primary coolant makeup in beyond design basis events.	nonsafety-related	Module Protection System Test #63-11
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify a pump curve test has been completed and approved for the CVCS pumps. iii. Component Level Tests iv., v., and vi. must be performed under preoperational test conditions that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each CVCS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each CVCS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each CVCS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position

Table 14.2-38: Chemical and Volume Control System Test # 38 (Continued)

iv. Verify each CVCS ASME Code Class 3 air-operated valve changes position under preoperational temperature, differential pressure, and flow conditions.	Operate each valve from the MCR.	MCR display verifies the valve opens and closes under preoperational temperature, differential pressure, and flow conditions. [ITAAC 02.02.03]
v. Verify each CVCS ASME Code Class 3 air-operated valve fails to its safe position on loss of air under preoperational temperature, differential pressure, and flow conditions.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position under preoperational temperature, differential pressure, and flow conditions. [ITAAC 02.02.05]
vi. Verify each CVCS ASME Code Class 3 air-operated valve fails to its safe position on loss of electrical power to its solenoid under preoperational temperature, differential pressure, and flow conditions.	Place each valve in its non-safe position. Isolate electrical power to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position under preoperational temperature, differential pressure, and flow conditions. [ITAAC 02.02.05]
vii. Verify each CVCS pump can be started and stopped remotely.	Align the CVCS to allow for pump operation. Start and stop each CVCS pump form the MCR.	MCR display and local, visual observation indicate the pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
viii. Verify the speed of each CVCS variable-speed pump can be manually controlled.	Align the CVCS to provide a flow path to operate a selected pump. Vary the CVCS pump speed from minimum to maximum from the MCR.	MCR display indicates the speed of each obtains both minimum and maximum pump speeds.
ix. Verify each CVCS operating makeup pump automatically stops to protect the pump and the standby pump starts.	Align the CVCS to allow for pump operation. Place a makeup pump in service. Initiate a simulated makeup pump trip.	MCR display and local, visual observation indicate the operating pump stops and the standby pump starts. Audible and visible water hammer are not observed when the pump starts.
x. Verify each CVCS recirculation pump automatically stops to protect the pump and the standby pump starts.	Align the CVCS to allow for pump operation. Place a recirculation pump in service. Initiate a simulated recirculation pump trip.	MCR display and local, visual observation indicate the operating pump stops and the standby pump starts.
xi. Verify CVCS letdown flow isolates on high flow to protect plant equipment.	Initiate a simulated CVCS high letdown flow signal.	MCR display and local, visual observation indicate the following: LRWS letdown flow control valve and LRWS letdown isolation valves (3) are closed.
xii. Verify CVCS hydrogen injection isolates on low injection pressure to protect plant equipment.	Initiate a simulated CVCS low hydrogen injection pressure signal.	MCR display and local, visual observation indicate the following: CVCS hydrogen injection pressure regulating valve and hydrogen injection isolation valve are closed.

Table 14.2-38: Chemical and Volume Control System Test # 38 (Continued)

xiii. Verify ion exchanger isolation on non-regenerative heat exchanger high outlet temperature to protect plant equipment.	Initiate a simulated high non-regenerative heat exchanger outlet temperature signal.	MCR display and local, visual observation indicate the following: i. CVCS purification bypass diverting valve is in the bypass position. ii. Mixed bed ion exchanger A inlet isolation valves (2) are closed. iii. Auxiliary ion exchanger inlet isolation valve is closed. iv. Cation exchanger inlet isolation valve is closed.
xiv. Verify the CVCS automatically responds to mitigate a release of radioactivity.	Initiate a real or simulated high radiation signal for the auxiliary boiler steam flow to the 0A module heatup system (MHS) heat exchanger.	MCR display verifies the following: i. CVCS MHS 0A heat exchanger inlet and outlet isolation valves are closed. [This component-level test is required to be performed once for each CVCS associated with the MHS 0A heat exchanger.] [ITAAC 02.07.02]
xv. Verify the CVCS automatically responds to mitigate a release of radioactivity.	Initiate a real or simulated high radiation signal for the auxiliary boiler steam flow to the 0B MHS heat exchanger.	MCR display verifies the following: i. CVCS MHS 0B heat exchanger inlet and outlet isolation valves are closed. [This component-level test is required to be performed once for each CVCS associated with the MHS 0B heat exchanger.] [ITAAC 02.07.02]
xvi. Verify the CVCS automatically responds to mitigate a release of radioactivity.	Initiate a real or simulated high radiation signal for the RCS discharge flow to the regenerative heat exchanger.	MCR display verifies the following: i. chemical and volume control RCS discharge to process sampling isolation valve closed. [This component-level test is required to be performed once for each CVCS.] [ITAAC 02.07.02]
xvii. Verify each CVCS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each CVCS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

Table 14.2-38: Chemical and Volume Control System Test # 38 (Continued)

System Level Test #38-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify proper operation of the automatic pressurizer level control.	This test will be performed in conjunction with Turbine Generator System Test #33-1, which heats the RCS from ambient conditions to no less than 420°F but as high as reasonably achievable. i. Place pressurizer level control in automatic operation during RCS heatup to demonstrate automatic letdown. Use the MCS data historian to review pressurizer level at maximum-obtained RCS temperature. ii. To raise pressurizer level, use MCS automation and operator permission to increase to a target pressurizer level. Note: Pressurizer letdown level control is automatic; however pressurizer makeup level control is automatic with consent of operator.	i. MCS data indicates that automatic pressurizer letdown maintained pressurizer level at setpoint as described in Section 9.3.4.5. ii. MCS data indicates that the pressurizer level control results in CVCS makeup to the RCS to increase pressurizer level to the target setpoint as described in Section 9.3.4.5.
System Level Test #38-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify proper operation of the automatic pressurizer pressure control.	This test will be performed in conjunction with Turbine Generator System Test #33-1 which heats the RCS from ambient conditions to no less than 420°F but as high as reasonably achievable. Place pressurizer pressure control in automatic and raise pressure setpoint to the normal operating band. Raise pressurizer pressure to the pressurizer spray valve open setpoint. Use the MCS data historian to review pressurizer pressure at maximum-obtained RCS temperature.	i. MCS data indicates automatic pressurizer heater operation raised pressurizer pressure to the setpoint as described in Section 9.3.4.5. ii. MCS data indicates automatic pressurizer spray valve operation lowered pressurizer pressure to the spray valve closure setpoint as described in Section 9.3.4.5.

Table 14.2-38: Chemical and Volume Control System Test # 38 (Continued)

System Level Test #38-3		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify proper operation of CVCS automatic dilution and boration control.	This test will be performed in conjunction with Turbine Generator System Test #33-1 which heats the RCS from ambient conditions to no less than 420°F but as high as reasonably achievable. Ensure that RCS low flow rate alarm is clear to ensure adequate mixing for dilution and boration. i. Place the BAS storage tank on recirculation and sample boron concentration. ii. Use the MCS automation and operator permission to decrease to a target RCS boron concentration. iii. Use the MCS and operator permission to increase to a target RCS boron concentration.	i. BAS storage tank sample boron concentration is within specifications (as described in Section 9.3.4.2.1). ii. MCS data indicates that the dilution of the RCS results in a decreased boron concentration within acceptable limits of the target concentration as described in Section 9.3.4.5. iii. MCS data indicates that the boration of the RCS results in a increased boron concentration within acceptable limits of the target concentration as described in Section 9.3.4.5.

Table 14.2-39: Boron Addition System Test # 39

Preoperational test is required to be performed for each NPM.		
The boron addition system (BAS) is described in Section 9.3.4. The BAS function verified by this test is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The BAS supports the spent fuel pool cooling system (SFPCS) by providing borated water to the Reactor Building (RXB) pools.	nonsafety-related	Test #39-1
The BAS function verified by other test is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The BAS supports the chemical and volume control system by providing uniformly mixed borated water on demand.	nonsafety-related	Chemical and Volume Control System Test #38-3
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify a pump curve test has been completed and approved for the BAS pumps.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each BAS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each BAS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each BAS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify the BAS transfer pump can be started and stopped remotely.	Align the BAS to allow for pump operation. Stop and start the transfer pump from the MCR.	MCR display and local, visual observation indicate the pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
v. Verify the BAS supply pump can be started and stopped remotely.	Align the BAS to allow for pump operation. Start and stop the supply pump from the MCR.	MCR display and local, visual observation indicate the pump starts and stops. Audible and visible water hammer are not observed when the pump starts.
vi. Verify the speed of the BAS variable-speed pumps can be manually controlled.	Align the BAS to provide a flow path to operate a selected pump. Vary the BAS pump speed from minimum to maximum from the MCR.	MCR display indicates the speed of each pump obtains both minimum and maximum pump speeds. Audible and visible water hammer are not observed when the pump starts.
vii. Verify a local grab sample can be obtained from a BAS grab sample device.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
viii. Verify each BAS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each BAS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

Table 14.2-39: Boron Addition System Test # 39 (Continued)

System Level Tests		
System Level Test #39-1		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify the BAS automatically adds a specified quantity of borated water from the BAS batch tank to the RXB pools.	i. Verify the BAS batch tank contains a sufficient volume of water to conduct this test. ii. Align the BAS and the SFPCS to supply water from the BAS to the SFPCS pump suction. iii. Enter a BAS batch tank target level to terminate batch operation to the spent fuel pool.	MCR displays and local, visual observation verifies the following: i. The BAS to SFPCS valve initially opens to supply water from the BAS to the SFPCS pump suction. ii. The BAS to SFPCS valve automatically closes when the BAS batch tank obtains the target level.

Table 14.2-40: Module Heatup System Test # 40

Preoperational test is required to be performed for each NuScale Power Module.		
The module heatup system (MHS) is described in Section 9.3.4.2. MHS functions are not verified by MHS tests. MHS functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The MHS supports the chemical and volume control system by adding heat to primary coolant.	nonsafety-related	Turbine Generator System Test #33-1
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
i. Verify a local grab sample can be obtained from a MHS grab sample device.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is obtained.
ii. Verify each MHS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each MHS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-41: Containment Evacuation System Test # 41

<table>
<tr><td colspan="3">Preoperational test is required to be performed for each NuScale Power Module (NPM).</td></tr>
<tr><td colspan="3">The containment evacuation system (CES) is described in Sections 9.3.6, 11.5.2.2.7 and 5.2.5 and the functions verified by this test or another preoperational test are:</td></tr>
<tr><td>System Function</td><td>System Function Categorization</td><td>Function Verified by Test #</td></tr>
<tr><td>1. The CES supports the containment system (CNTS) by removing water vapor from the containment vessel (CNV).</td><td>nonsafety-related</td><td>Test #41-1
Test #41-2
Test #41-3</td></tr>
<tr><td>2. The CES supports the CNTS by condensing water vapor removed from the CNV in the containment evacuation condenser.</td><td>nonsafety-related</td><td>Test #41-1
Test #41-2
Test #41-3</td></tr>
<tr><td>3. The CES supports the CNTS by removing non-condensable gases from the CNV.</td><td>nonsafety-related</td><td>Test #41-1
Test #41-2
Test #41-3</td></tr>
<tr><td>4. The CES supports CNTS by providing leak-before-break leak detection monitoring capability.</td><td>nonsafety-related</td><td>Test #41-4</td></tr>
<tr><td>5. The CES supports the reactor coolant system (RCS) by providing RCS leak detection monitoring capability.</td><td>nonsafety-related</td><td>Test #41-3</td></tr>
<tr><td>6. The gaseous radioactive waste system (GRWS) supports the CES by receiving and collecting potentially radioactive and hydrogen-bearing waste gases which require processing prior to release to the environment.</td><td>nonsafety-related</td><td>Test #41-2
Gaseous Radioactive Waste System Test #36-1</td></tr>
<tr><td colspan="3">Prerequisites
Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.</td></tr>
<tr><td colspan="3">Component Level Tests</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>i. Verify each CES remotely-operated valve can be operated remotely.</td><td>Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).</td><td>MCR display and local, visual observation indicate each valve fully opens and fully closes.</td></tr>
<tr><td>ii. Verify each CES air-operated valve fails to its safe position on loss of air.</td><td>Place each CES valve in its non-safe position. Isolate and vent air to the valve.</td><td>MCR display and local, visual observation indicate each valve fails to its safe position.</td></tr>
<tr><td>iii. Verify each CES air-operated valve fails to its safe position on loss of electrical power to its solenoid.</td><td>Place each CES valve in its non-safe position. Isolate electrical power to each CES air-operated valve.</td><td>MCR display and local, visual observation indicate each valve fails to its safe position.</td></tr>
<tr><td>iv. Verify each CES pump can be started and stopped remotely.</td><td>Stop and start each pump from the MCR.</td><td>MCR display and local, visual observation indicate each pump starts and stops.</td></tr>
<tr><td>v. Verify the speed of each CES variable-speed pump can be manually controlled.</td><td>Vary the speed of each pump from the MCR and local control panel (if design has local pump control).</td><td>MCR display indicates pump speed varies from minimum to maximum speed.</td></tr>
<tr><td>vi. Verify each CES pump automatically stops to protect plant equipment.</td><td>Place a pump in operation. Initiate a real or simulated signal for each pump trip condition.</td><td>MCR displays and local, visual observation verifies the pump stops.</td></tr>
</table>

Table 14.2-41: Containment Evacuation System Test # 41 (Continued)

vii. Verify each CES pump suction and discharge valve automatically closes to protect the CES equipment.	Open the pump suction and discharge valves. Initiate a real or simulated signal for each valve close conditions.	Each pump suction and discharge valve closes on each real or simulated valve close condition.
viii. Verify a local grab sample can be obtained from a CES grab sample device indicated on the CES piping and instrumentation diagram.	Place the system in service to allow flow through the grab sampling device.	A local grab sample is successfully obtained.
ix. Verify each CES instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each CES transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #41-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the automatic operation of the CES to establish and maintain design vacuum for the CNV.	After the containment flooding and drain system (CFDS) completes draindown of the CNV and the NPM is in hot functional testing, place the CES in automatic operation.	The automated control establishes and maintains vacuum in the CNV within design limit per Section 6.2.2.
System Level Test #41-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify radiation isolation and flow diversion on high radiation level in the CES.	The NPM is in hot functional testing with the RCS at normal operating pressure. The CES is operating in automatic control with a CNV steady-state vacuum pressure indicating the noncondensable gases have been removed from the CNV. Initiate a real or simulated high radiation signal for the CES vacuum pump discharge.	i. The CES effluent flow path to the Reactor Building HVAC system is isolated and diverted to GRWS. [ITAAC 02.07.01] ii. The CES effluent to process sample panel isolation valve is closed. [ITAAC 02.07.01] iii. The CES purge air solenoid valves to the vacuum pumps are closed. [ITAAC 02.07.01] iv. The automated control maintains vacuum in the CNV.

Table 14.2-41: Containment Evacuation System Test # 41 (Continued)

System Level Test #41-3		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify the CES level instrumentation supports RCS leakage detection. ii. Verify the CES pressure instrumentation supports RCS leakage detection.	i. The NPM is in hot functional testing with the RCS at normal operating pressure and the maximum operating temperature achievable by heating the RCS with the module heatup system (MHS). ii. The CES is operating in automatic control with a CNV steady-state vacuum pressure indicating the noncondensable gasses have been removed from the CNV. iii. Record the MCS baseline leakage rate into the CNV. iv. Isolate the CFDS to CNTS spool piece to allow test equipment to be connected to the spool piece. v. Inject water at a flow rate less than or equal to one gpm. This test may be done in conjunction Test #41-2.	i. The CES detects a level increase in the CES sample tank, which correlates to a detection of an unidentified RCS leakage rate of one gpm within one hour, by providing an alarm signal to the MCR within one hour of the start of water injection into the CNV indicating the baseline leakage rate has been exceeded. [ITAAC 02.03.01] ii. The CES detects a pressure increase in the inlet pressure instrumentation that correlates to a detection of an unidentified RCS leakage rate of one gpm within one hour, by providing an alarm signal to the MCR within one hour of the start of water injection into the CNV indicating the baseline leakage rate has been exceeded. [ITAAC 02.03.02]
System Level Test #41-4		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify the CES level instrumentation supports leak-before-break leakage detection. ii. Verify the CES pressure instrumentation supports leak-before-break leakage detection.	i. The NPM is in hot functional testing with the RCS at normal operating pressure and the maximum operating temperature achievable by heating the RCS with the MHS. ii. The CES is operating in automatic control with a CNV steady-state vacuum pressure indicating the noncondensable gases have been removed from the CNV. iii. Record the MCS baseline leakage rate into the CNV. iv. Isolate the CFDS to CNTS spool piece to allow test equipment to be connected to the spool piece. v. Inject water at a flow rate less than or equal to one gpm. This test may be done in conjunction Test #41-2.	i. The CES detects a level increase in the CES sample tank, which correlates to a detection of an unidentified leakage rate of one gpm within one hour, by providing an alarm signal to the MCR within one hour of the start of water injection into the CNV indicating the baseline leakage rate has been exceeded. ii. The CES detects a pressure increase in the inlet pressure instrumentation that correlates to a detection of an unidentified leakage rate of one gpm within one hour, by providing an alarm signal to the MCR within one hour of the start of water injection into the CNV indicating the baseline leakage rate has been exceeded.

Table 14.2-42: Containment Flooding and Drain System Test # 42

Preoperational test component level testing is required to be performed for the 0A containment flooding and drain system (CFDS) and for the 0B CFDS. System level testing is required to be performed as indicated for each system level test.		
The CFDS is described in Section 9.3.6 and 11.5.2.2.9 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The CFDS supports the containment system (CNTS) by flooding the containment vessel (CNV) in preparation for refueling operations.	nonsafety-related	Test #42-2
2. The CFDS supports the CNTS by draining the CNV in preparation for startup operations.	nonsafety-related	Test #42-1
The CFDS function verified by another test is:		
1. The CFDS supports the reactor coolant system (RCS) by providing borated coolant inventory for the removal of core heat during a beyond design basis accident.	nonsafety-related	Module Protection System Test #63-11
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each CFDS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each CFDS air-operated valve fails to its safe position on loss of air.	Place each CFDS valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each CFDS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each CFDS valve in its non-safe position. Isolate electrical power to each CFDS air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each CFDS pump can be started and stopped remotely.	Stop and start each pump from the MCR.	MCR display and local, visual observation indicate each pump starts and stops.
v. Verify each CFDS pump automatically stops to protect plant equipment.	Place a pump in operation. Initiate a real or simulated signal for each pump trip condition.	MCR displays and local, visual observation verifies the pump stops.
vi. Verify each CFDS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each CFDS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #42-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the CFDS can automatically drain the CNTS.	Drain the CNTS using CFDS automatic operation and designed manual operation.	The CNTS is drained using CFDS automatic controls. (This test is required to be performed for each NuScale Power Module (NPM).)

Table 14.2-42: Containment Flooding and Drain System Test # 42 (Continued)

System Level Test #42-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the CFDS can automatically flood the CNTS.	Flood the CNTS using CFDS automatic operation and designed manual operation.	The CNTS is flooded using CFDS automatic controls. (This test is required to be performed for each NPM.)
System Level Test #42-3		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the 0A CFDS automatically responds to mitigate a release of radioactivity.	While the 0A CFDS is draining the CNTS initiate a real or simulated high radiation signal on the gaseous effluent of the 0A CFDS containment drain separator tank.	The 0A CFDS containment drain separator gaseous discharge to Reactor Building HVAC system (RBVS) isolation valve is closed. [ITAAC 03.17.01]
System Level Test #42-4		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the 0B CFDS automatically responds to mitigate a release of radioactivity.	While the 0B CFDS is draining the CNTS initiate a real or simulated high radiation signal on the gaseous effluent of the 0B CFDS containment drain separator tank.	The 0B CFDS containment drain separator gaseous discharge to RBVS isolation valve is closed. [ITAAC 03.18.01]

Table 14.2-43: Containment System Test # 43

Preoperational test is required to be performed for each NuScale Power Module.		
The containment system (CNTS) is described in Section 6.2 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The CNTS supports the Reactor Building (RXB) by providing a barrier to contain mass, energy, and fission product release from a degradation of the reactor coolant pressure boundary.	safety-related	Test #43-1 Test #43-2
2. The CNTS supports the emergency core cooling system (ECCS) operations by providing a sealed containment.	safety-related	Test #43-1
3. The ECCS supports CNTS by providing a portion of the containment boundary for maintaining containment integrity.	safety-related	Test #43-1
The CNTS functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The CNTS supports the decay heat removal system (DHRS) by closing containment isolation valves (CIVs) for the main steam and feedwater systems when actuated by the module protection system (MPS) for DHRS operation.	safety-related	Module Protection System Test #63-6
The CNTS supports the reactor coolant system (RCS) by closing the CIVs for pressurizer spray, RCS injection, RCS discharge, and reactor pressure vessel (RPV) high point degasification when actuated by the MPS for RCS isolation.	safety-related	Module Protection System Test #63-6
The CNTS supports the RXB by providing a barrier to contain mass, energy, and fission product release by closure of the CIVs upon a containment isolation signal.	safety-related	Module Protection System Test #63-6
The CNTS supports the Reactor Building crane (RBC) by providing lifting attachment points that the RBC can connect to so that the module can be lifted.	nonsafety-related, risk-significant	Reactor Building Crane Test #52-1 Reactor Building Crane Test #52-2
The CNTS supports the MPS by providing post-accident monitoring nonsafety-related information signals	nonsafety-related	Safety Display and Indication System Test #66-2
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each hydraulic skid supplies sufficient pressure for valve operation.	i. Verify each hydraulic skid supplies sufficient pressure for valve operation.	i. Pump maintains required system pressure.

Table 14.2-43: Containment System Test # 43 (Continued)

ii. Verify each CNTS instrument is available on an MCS or PCS display. (Test not required if the instrument calibration verified the MCS or PCS display.)	ii. Initiate a single real or simulated instrument signal from each CNTS transmitter.	ii. The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #43-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the leaktightness of the containment system.	Perform 10 CFR Part 50, Appendix J local leak rate tests (Type B and Type C tests) of the CNTS in accordance with the guidance provided in ANSI/ANS 56.8,RG 1.163, and NEI 94-01.	Local leak rate tests are completed on containment penetrations listed in Table 6.2-4 which require Appendix J, Type B or C testing. [ITAAC 02.01.07]
System Level Test #43-2 Test 43-2 is performed at hot functional conditions.		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the CNTS safety-related check valves change position under design temperature, differential pressure, and flow.	The check valves are tested in accordance with the requirements of ASME OM code, ISTC-5220, check valves.	Each CNTS safety-related check valve strokes fully open and closed under forward and reverse flow conditions, respectively. [ITAAC 02.01.21]

Table 14.2-44: Not Used

Table 14.2-45: Not Used

Table 14.2-46: Reactor Coolant System Test # 46

Preoperational test is required to be performed for each NuScale Power Module.		
The reactor coolant system (RCS) is described in Section 5.4 and the RCS functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The RCS supports the module protection system (MPS) by providing instrument information signals for MPS actuation.	safety-related, risk-significant	Module Protection System Test #63-1
2. The RCS supports the MPS by providing instrument information signals for low temperature overpressure protection actuation.	safety-related	Module Protection System Test #63-1
3. The RCS supports the MPS by providing post-accident monitoring instrument information signals.	nonsafety-related	Safety Display and Indication System Test #66-2
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each RCS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each RCS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-47: Emergency Core Cooling System Test # 47

Preoperational test is required to be performed for each NuScale Power Module. System Level Test #47-1 is only required to be performed once for NPM #1. This test supports first-of-a-kind (FOAK) testing as described in Section 14.2.3.3.		
The emergency core cooling system (ECCS) is described in Section 6.3, and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The ECCS supports the reactor coolant system (RCS) by opening the ECCS reactor vent valves and reactor recirculation valves when their respective trip valve is actuated by the module protection system (MPS).	safety-related	Test #47-1 Module Protection System Test #63-6
2. The ECCS supports the RCS by providing recirculated coolant from the containment to the reactor pressure vessel (RPV) for the removal of core heat.	safety-related	Test #47-1 Module Protection System Test #63-6
The ECCS functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The ECCS supports the RCS by providing low temperature overpressure protection (LTOP) for maintaining the reactor coolant pressure boundary.	safety-related	Module Protection System Test #63-6
2. The ECCS supports the containment system (CNTS) by providing a portion of the containment boundary for maintaining containment integrity.	safety-related	Containment System Test #43-1
3. The ECCS supports MPS by providing post accident monitoring instrument information signals.	nonsafety related	Safety Display and Indication System Test #66-2
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
None		
System Level Test #47-1 **Test #47-1 is performed at hot functional testing to allow ECCS actuation at elevated RCS pressure and temperature conditions, without engaging the inadvertent actuation block (IAB).**		
The RCS is heated to the highest temperature achievable by module heatup system (MHS) heating. These hot functional testing conditions provide the highest temperature conditions that can be achieved prior to fuel load. The RCS level is within the expected range of module operation, near the low end of the normal operating range for hot zero power conditions.		

Table 14.2-47: Emergency Core Cooling System Test # 47 (Continued)

Test Objective	Test Method	Acceptance Criteria
i. Verify the RPV liquid level remains above the top of the core during and following ECCS actuation. ii. Verify the heat removal capacity of the ECCS, operating with the containment vessel (CNV), is consistent with the design basis.	i. Ensure RCS pressure is as close to, but below, the IAB RCS pressure threshold as practical. ii. Ensure RCS temperature is at the maximum temperature achievable by heating the RCS using MHS heating. iii. Ensure RCS level is as low in the normal operating band as is practically achievable for the established plant conditions. iv. Manually initiate ECCS from the main control room. v. Allow RPV riser level and CNV level to become relatively stable.	i. RPV riser level remains above the top of the core. ii. CNV pressure remains within upper and lower bounds calculated using safety analysis methods, while accounting for test initial conditions and instrumentation uncertainty.

Table 14.2-48: Decay Heat Removal System Test # 48

Preoperational test is required to be performed for each NuScale Power Module (NPM). System Test #48-1 is required to be performed once for NPM #1. This test supports first-of-a-kind (FOAK) testing described in Section 14.2.3.3.		
The decay heat removal system (DHRS) is described in Section 6.3. FOAK Test #48-1 is described in Section 5.4.3.4. DHRS functions are not verified by DHRS tests. DHRS functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The DHRS supports the reactor coolant system (RCS) by opening the DHRS actuation valves for DHRS operation.	safety-related	Module Protection System Test #63-6 Reactor Trip from 100 Percent Power Test # 104
2. The DHRS supports the MPS by providing MPS actuation instrument information signals.	safety-related	Module Protection System Test #63-1
3. The DHRS supports the MPS by providing post-accident monitoring instrument information signals.	nonsafety-related	Safety Display and Indication System Test #66-2
4. The ultimate heat sink supports the DHRS by accepting the heat from the DHRS heat exchanger.	safety-related	Reactor Trip from 100 Percent Power Test # 104
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each DHRS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each DHRS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Test #48-1 **RCS is at normal operating pressure and the RCS has achieved the maximum temperature achievable by warming the RCS using module heatup system heating.**		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify DHRS removes heat from the RCS.	i. Verify RCS is at normal operating pressure and the RCS has achieved the maximum temperature achievable by warming the RCS using module heatup system heating. ii. Open DHRS actuation valves and close containment isolation valves by initiating a containment isolation via MPS. iii. Allow the RCS to cool down less than 420 degrees. iv. Compare RCS cooldown rate to test analysis conducted using the code of record as described in Section 5.4.3.4.	DHRS cooldown of RCS meets design basis requirements.

Table 14.2-49: In-core Instrumentation System Test # 49

<table>
<tr><td colspan="3">Preoperational test is required to be performed for each NuScale Power Module.</td></tr>
<tr><td colspan="3">The in-core instrumentation system (ICIS) is described in Section 7.0.4.7 and the function verified by this test and power ascension testing is:</td></tr>
<tr><td>System Function</td><td>System Function Categorization</td><td>Function Verified by Test #</td></tr>
<tr><td>1. The ICIS supports the module protection system (MPS) by providing reactor core (RXC) temperature information.</td><td>nonsafety-related</td><td>Test #49-1
Reactor Coolant System Temperature Instrument Calibration Test #93</td></tr>
<tr><td colspan="3">The ICIS functions verified by another test is:</td></tr>
<tr><td>System Function</td><td>System Function Categorization</td><td>Function Verified by Test #</td></tr>
<tr><td>The ICIS supports the MPS by providing RXC temperature information.</td><td>nonsafety-related</td><td>Safety Display and Indication System Test #66-2</td></tr>
<tr><td colspan="3">Prerequisites
i. The ICIS instrument strings are inserted into the core.
ii. Verify an instrument calibration has been performed on all ICIS thermocouples by cross-calibrating the thermocouple to the reactor coolant system (RCS) narrow range resistance temperature detectors (RTDs) prior to RCS heatup.</td></tr>
<tr><td colspan="3">Component Level Tests</td></tr>
<tr><td colspan="3">None</td></tr>
<tr><td colspan="3">System Level Test #49-1</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>Verify proper temperature indication is obtained from the ICIS thermocouples.</td><td>Heat the RCS from ambient conditions to the maximum RCS temperature that can be obtained by the module heatup system.
Use the module control system (MCS) data historian to cross-check the ICIS thermocouples to each other and the RCS narrow-range and wide range RTDs.</td><td>MCS data indicates that the ICIS thermocouples respond properly.</td></tr>
</table>

Table 14.2-50: Module Assembly Equipment Test # 50

There are no preoperational tests for module assembly equipment (MAE).		
The MAE consists of module import trolley, the upender, and the inspection rack.		
System Function	**System Function Categorization**	**Function Verified by Test #**
MAE supports the NuScale Power Module actively by providing material handling to allow its transport in the horizontal orientation to travel from outside the Reactor Building to its interior and to rotate it to operational orientation.	nonsafety-related	component-level tests
Prerequisites An MAE factory acceptance test has been successfully completed and approved, if required.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify the operation of MAE controls that limit motion and speed. (This test may be performed as part of site acceptance testing.)	Actuate or simulate actuation of the interlocks.	The MAE equipment controls limit motion and speed per design.
System-Level Tests		
None		

Table 14.2-51: Fuel Handling Equipment Test # 51

<table>
<tr><td colspan="3">Preoperational test is required to be performed once.</td></tr>
<tr><td colspan="3">The fuel handling equipment (FHE) system is described in Section 9.1.4 and the functions verified by this test are:</td></tr>
<tr><td>System Function</td><td>System Function Categorization</td><td>Function Verified by Test #</td></tr>
<tr><td>1. The FHE system supports new fuel by providing ability to visually inspect fuel.</td><td>nonsafety-related</td><td>Test #51-1
Test #51-2</td></tr>
<tr><td>2. The FHE system supports the reactor core (RXC) by moving fuel within the core.</td><td>nonsafety-related</td><td>Test #51-3
Test #51-4</td></tr>
<tr><td>3. The FHE system supports the spent fuel storage system by moving fuel into the spent fuel storage system.</td><td>nonsafety-related</td><td>Test #51-4</td></tr>
<tr><td colspan="3">Prerequisites
i. An FHE system factory acceptance test has been successfully completed and approved.
ii. A rated-load test has been successfully completed and approved on the FHE system on the following equipment in accordance with ASME NOG-1 paragraph 7423.
a. Fuel handling machine (FHM) main hoist
b. FHM auxiliary hoists
iii. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.</td></tr>
<tr><td colspan="3">Component Level Tests</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>i. Verify the operation of FHE controls that limit motion and speed.</td><td>Actuate or simulate actuation of the interlocks contained in Table 14.2-51a.</td><td>The FHE equipment controls limit motion and speed per design.</td></tr>
<tr><td>ii. Verify each FHE instrument is available on a module control system (MCS) or plant control system (PCS) display if the FHE instrument is designed to be displayed on a main control room workstation.
(Test not required if the instrument calibration verified the MCS or PCS display.)</td><td>Initiate a single real or simulated instrument signal from each FHE system transmitter.</td><td>The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.</td></tr>
<tr><td colspan="3">System Level Test #51-1</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>Verify the proper operation of the new fuel jib crane.</td><td>Transfer a dummy fuel assembly from its receipt shipping container to the new fuel inspection stand and from the new fuel inspection stand to the new fuel elevator.</td><td>i. A dummy fuel assembly is successfully transferred to the new fuel inspection stand.
ii. A dummy fuel assembly is successfully transferred to the new fuel elevator.</td></tr>
</table>

Table 14.2-51: Fuel Handling Equipment Test # 51 (Continued)

System Level Test #51-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the proper operation of the new fuel elevator.	Lower a dummy fuel assembly in the new fuel elevator.	A dummy fuel assembly is successfully lowered to the position where it can be retrieved by the FHM mast.
System Level Test #51-3		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the proper operation of the FHM.	i. Transfer the dummy fuel assembly from the new fuel elevator to the FHM mast. ii. Transfer the dummy fuel assembly from the new fuel elevator location to a designated RXC location. iii. Seat the dummy fuel assembly.	i. The dummy fuel assembly is successfully transferred to the FHM mast. ii. The dummy fuel assembly is successfully transferred to its designated core location and partially inserted. iii. The dummy fuel assembly is fully seated.
System Level Test #51-4		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the proper operation of the FHM.	i. Withdraw the dummy fuel assembly to a position where the FHM can automatically transfer the assembly. Transfer the dummy fuel assembly from the RXC to a designated spent fuel storage location. (Manual operation of the fuel assembly is required for final fuel insertion.) ii. Seat the dummy fuel assembly.	i. The dummy fuel assembly is successfully transferred to its designated storage location and partially inserted. ii. The dummy fuel assembly is fully seated.
System Level Test #51-5		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify 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.	Perform a test of the FHM mast mechanical stop limit switch.	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. [ITAAC 03.04.05]
System Level Test #51-6		
Test Objective	**Test Method**	**Acceptance Criteria**
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.	Using the new fuel jib crane hook attempt to transfer a dummy fuel assembly or new fuel assembly over the fuel storage racks in the spent fuel pool.	The new fuel jib crane interlocks prevent the crane from carrying a fuel assembly over the spent fuel racks. [ITAAC 03.04.06]

Table 14.2-51a: FHE Interlock Testing

Equipment	Emergency Stop	Bridge and Trolley End of Travel	Crane Zone Limits	Hoist Underload (Underweight / Slack Rope) (Lower)	Hoist Overload (Overweight) (Raise)	Hoist Up-Position (Upper Travel Limit)	Hoist Down-Position (Lower Travel Limit)	Rotation and Gripper	Hoist, Bridge, Trolley Slow Zones	Overspeed Limit	Mis-reeve Limit Switch
FHM bridge	X	X	X	---	---	---	---		X	---	---
FHM trolley	X	X	X	---	---	---	---		X	---	---
FHM main hoist/mast	X	---	---	X	X	X	X	X	X	X	X
FHM auxiliary hoist	X	---	---	---	X	X	X		---	---	X
New fuel jib crane trolley	X	X	---	---	---	---	---	X		---	---
New fuel jib crane hoist	X	---	---	---	X	X	X			X	X
New fuel elevator	X	---	---	---	X	X	X				X

Table 14.2-52: Reactor Building Cranes Test # 52

<table>
<tr><td colspan="3">Preoperational test is required to be performed once unless otherwise noted in the test.</td></tr>
<tr><td colspan="3">The Reactor Building crane (RBC) system is described in Section 9.1.5 and the functions verified by this test are:</td></tr>
<tr><td>System Function</td><td>System Function Categorization</td><td>Function Verified by Test #</td></tr>
<tr><td>1. The RBC supports the NuScale Power Module (NPM) by providing structural support and mobility while moving from refueling, inspection and operating bay.</td><td>nonsafety-related, risk-significant</td><td>Test #52-1
Test #52-2</td></tr>
<tr><td>2. Module assembly equipment (MAE) bolting supports the containment system (CNTS) by providing material handling to allow for disassembly and reassembly of the containment vessel (CNV) lower flange.</td><td>nonsafety-related</td><td>Test #52-2</td></tr>
<tr><td>3. MAE bolting supports the reactor pressure vessel (RPV) actively by providing material handling to allow for disassembly and reassembly of the RPV lower flange.</td><td>nonsafety-related</td><td>Test #52-2</td></tr>
<tr><td>4. The CNTS supports the RBC by providing lifting attachment points that the RBC can connect to so that the module can be lifted.</td><td>nonsafety-related, risk-significant</td><td>Test #52-1
Test #52-2</td></tr>
<tr><td colspan="3">Prerequisites
i. An RBC site acceptance test has been completed and approved.
ii. A rated-load test has been completed and approved on the RBC on the following equipment in accordance with ASME NOG-1 paragraph 7423.
a. RBC main hoist
b. RBC auxiliary hoists
c. RBC wet hoist
iii. A rated-load test has been completed and approved on the following equipment in accordance with ANSI N14.6.
a. Module lifting adapter (MLA)
b. NPM lifting fixture
iv. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.</td></tr>
</table>

Table 14.2-52: Reactor Building Cranes Test # 52 (Continued)

Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify RBC controls that limit RBC motion and speed.	Actuate or simulate actuation of the RBC interlocks contained in Table 14.2-52a.	Local visual observation indicates that the interlocks limit RBC motion and speed.
ii. Verify RBC remains in current position on loss of control or power or seismic event.	Initiate the following real or simulated signals: i. Loss of control. ii. Loss of power. iii. Seismic switch actuation.	Local visual observation indicates that the bridge, trolley, main hoist, wet hoist, auxiliary hoist trolley and auxiliary hoist brakes are set.
iii. Verify each RBC instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each RBC system transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

Table 14.2-52: Reactor Building Cranes Test # 52 (Continued)

System Level Test #52-1		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify RBC load path and removal of an NPM from a reactor bay. ii. Verify RBC load path and installation of an NPM in a reactor bay.	Place the module lifting adaptor on the RBC. Lift an NPM and move the RBC with the attached NPM to its design home location. i. Use the RBC semi-automatic programmed controls to install the NPM in the lead NPM bay location and return the RBC to the design home location ii. Use the RBC semi-automatic programmed controls to retrieve the NPM from the lead NPM bay location and return the RBC with attached module to the design home location. Repeat this sequence for each NPM installation.	i. The bridge and trolley speeds do not exceed maximum design speeds. ii. The bridge and trolley does not move at the same time. iii. The bridge and trolley maximum allowable speed is toggled from full-speed to microspeed when the RBC hook gets within the design distance of a predefined reference location. iv. The main hoist only moves within the predefined elevation zones. v. The NPM is positioned at the design rotation at predefined reference locations. vi. The NPM is fully seated in the reactor bay receiver. (Acceptance Criteria i through iv only need to be satisfied for the first performance of the test. Acceptance Criteria v and vi need to be satisfied for each NPM)

Table 14.2-52: Reactor Building Cranes Test # 52 (Continued)

System Level Test #52-2		
Test Objective	**Test Method**	**Acceptance Criteria**
i. a. Verify the NPM can be disassembled using the CNV support stand and the RPV support stand and associated tooling. b. Verify the RBC semi-automatic controls can be used to transport the NPM through the disassembly process. ii. a. Verify the NPM can be assembled using the CNV support stand and the RPV support stand and associated tooling. b. Verify the RBC semi-automatic controls can be used to transport the NPM through the assembly process.	The RBC is at the design home location with an NPM attached to the module lifting adaptor (MLA). i. Use the RBC semi-automatic programmed controls to move the NPM from the design home location to the CNV support stand and seat the NPM lower CNV in the CNV support stand. De-tension and remove the lower CNV closure bolts. Use the RBC semi-automatic programmed controls to move the NPM from the CNV support stand to the RPV support stand and seat the NPM in the RPV support stand. De-tension and remove the lower RPV closure bolts. Use the RBC semi-automatic programmed controls to move the upper NPM from the RPV support stand to the module inspection rack and seat the upper NPM on the module inspection rack support lug receiving pockets. Use the RBC semi-automatic programmed controls to disengage the MLA from the upper NPM and move the RBC and MLA from the module inspection rack to the design home location.	i. a. The NPM is disassembled using the CNV support stand and the RPV support stand and associated tooling. b. The RBC semi-automatic controls are used to transport the NPM through the disassembly process. ii. a. The NPM is assembled using the CNV support stand and the RPV support stand and associated tooling. b. The RBC semi-automatic controls are used to transport the NPM through the assembly process.

Table 14.2-52: Reactor Building Cranes Test # 52 (Continued)

	ii. Use the RBC semi-automatic programmed controls to move the NPM and MLA from the design home location to the module inspection rack and attach the upper NPM to the MLA. Use the RBC semi-automatic programmed controls to move the upper NPM from the module inspection rack to the RPV support stand and seat the upper NPM on the lower RPV and RPV support stand. Install and tension the lower RPV closure bolts. Use the RBC semi-automatic programmed controls to move the upper NPM from the RPV support stand to the CNV support stand and seat the upper NPM on the lower CNV and CNV support stand. Install and tension the lower CNV closure bolts. Use the RBC semi-automatic programmed controls move the RBC and NPM from the CNV support stand to the design home location.	

Table 14.2-52a: RBC System Interlock Testing

Equipment	Emergency Stop	Bridge and Trolley End of Travel	Crane Zone Limits	Hoist Underload (Underweight / Slack Rope) (Lower)	Hoist Overload (Overweight) (Raise)	Hoist Up-Position (Upper Travel Limit)	Hoist Down-Position (Lower Travel Limit)	Overspeed Limit	Mis-reeve Limit Switch	Unbalanced Load	Two-Blocking
RBC trolley	X	X	X								
RBC bridge	X	X	X								
RBC main hoist	X			X	X	X	X	X	X	X	X
RBC aux hoist trolley 1 & 2	X	X									
RBC aux hoist 1 & 2	X			X	X	X	X	X	X	X	
Wet hoist	X			X	X	X	X	X	X	X	X

Table 14.2-53: Process Sampling System Test # 53

Preoperational test is required to be performed for each NuScale Power Module (NPM).		
The process sampling system (PSS) is described in Section 9.3.2 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The PSS supports the reactor coolant system (RCS) during normal operations by providing sampling and analysis of reactor coolant discharge (letdown) liquid.	nonsafety-related	Test #53-1
2. The PSS supports the chemical and volume control system (CVCS) by providing sampling of reactor coolant at process points in the CVCS.	nonsafety-related	Test #53-1
3. The PSS supports the containment system (CNTS) during normal operations by providing sampling of containment gas and analysis of hydrogen and oxygen concentration in containment.	nonsafety-related	Test #53-2
4. The PSS supports the feedwater system (FWS) by providing sampling and analysis of condensate and feedwater.	nonsafety-related	Test #53-3
5. The PSS supports the main steam system (MSS) by providing sampling and analysis of main steam.	nonsafety-related	Test #53-3
6. The PSS system supports the auxiliary boiler system (ABS) by providing sampling and analysis of the auxiliary boiler steam and feedwater.	nonsafety-related	Test #53-3
7. PSS supports the CNTS during accident condition by providing containment atmosphere monitoring and analysis of hydrogen and oxygen concentration to respond to emergencies.	nonsafety-related	Test #53-2
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		

Table 14.2-53: Process Sampling System Test # 53 (Continued)

Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each PSS remotely-operated valve can be operated remotely.	Operate each valve from the main control room (MCR) and local control panel (if design has local valve control).	MCR display and local, visual observation indicate each valve fully opens and fully closes.
ii. Verify each PSS air-operated valve fails to its safe position on loss of air.	Place each valve in its non-safe position. Isolate and vent air to the valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iii. Verify each PSS air-operated valve fails to its safe position on loss of electrical power to its solenoid.	Place each valve in its non-safe position. Isolate electrical power to each air-operated valve.	MCR display and local, visual observation indicate each valve fails to its safe position.
iv. Verify each PSS return pump to CVCS can be started and stopped locally.	Align the PSS and CVCS to allow for return pump operation. Stop and start each return pump locally.	Local display and local, visual observation indicate each return pump starts and stops. Audible and visible water hammer are not observed when each return pump starts.
v. Verify each PSS return compressor to containment evacuation system (CES) can be started and stopped locally.	Align the PSS and CES to allow for compressor operation. Stop and start each return compressor locally.	Local display and local, visual observation indicate each return compressor starts and stops.
vi. Verify each PSS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each PSS transmitter.	The instrument signal is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.

Table 14.2-53: Process Sampling System Test # 53 (Continued)

<table>
<tr><td colspan="3">System Level Test #53-1</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>Verify sampling capability of the primary sampling points.</td><td>i. The NPM is in hot functional testing with the RCS at normal operating pressure and the maximum operating temperature achievable by heating the RCS with the module heatup system (MHS).
The RCS supply and discharge flow is in service.
Align the CVCS and PSS to provide continuous sampling flow to the PSS analysis panel.
ii. The RCS discharge line is in service.
Align the RCS and PSS to provide sampling flow to the primary sampling ion chromatography units.
iii. Open the PSS grab sample panel manual valve to obtain an RCS injection flow pressurized grab sample.
iv. Open the PSS grab sample panel manual valve to obtain an RCS discharge flow pressurized grab sample.
v. Open the PSS grab sample panel manual valve to obtain a CVCS demineralizer discharge flow pressurized grab sample.</td><td>i. The PSS analysis panel instruments provide indication of the water analysis.
ii. The primary sampling ion chromatography unit monitors for the programmed ion.
iii. An RCS injection flow grab sample is successfully obtained.
iv. An RCS discharge flow grab sample is successfully obtained.
v. A CVCS demineralizer discharge flow grab sample is successfully obtained.</td></tr>
<tr><td colspan="3">System Level Test #53-2</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>Verify sampling capability of the containment sampling points.</td><td>The NPM is in hot functional testing with the RCS at normal operating pressure and the maximum operating temperature achievable by heating the RCS with the MHS.
The CES is in service.
Align the CES and PSS to provide continuous sampling flow to the PSS containment gas sample panel.</td><td>The PSS containment gas sample panel instruments provide indication of the gas analysis.</td></tr>
</table>

Table 14.2-53: Process Sampling System Test # 53 (Continued)

System Level Test #53-3		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify sampling capability of the secondary sampling points.	i. The NPM is in hot functional testing with the RCS at normal operating pressure and the maximum operating temperature achievable by heating the RCS with the MHS. The FWS and MSS are in service. Align the FWS, MSS, PSS, and ABS to provide continuous sampling flow to the PSS secondary sampling system feedwater/main steam sample panel. ii. Open the manual feedwater/main steam ion chromatography analysis panel valve to obtain a feedwater to steam generator (SG) sample. iii. Open the manual feedwater/main steam ion chromatography analysis panel valve to obtain an SG #1 steam sample. iv. Open the manual feedwater/main steam ion chromatography analysis panel valve to obtain a SG #2 steam sample. v. Open the manual feedwater/main steam ion chromatography analysis panel valve to obtain a condensate pump discharge sample.	i. The PSS secondary sampling system feedwater/main steam sample panel instruments provide indication of the water and steam analysis. ii. The feedwater/main steam ion chromatography analysis panel monitors the programmed ion. iii. The feedwater/main steam ion chromatography analysis panel monitors the programmed ion. iv. The feedwater/main steam ion chromatography analysis panel monitors the programmed ion. v. The feedwater/main steam ion chromatography analysis panel monitors the programmed ion.

Table 14.2-54: 13.8kV and Switchyard System Test # 54

Preoperational test is required to be performed once for the 0A 13.8 kV and switchyard system (EHVS) and once for the 0B EHVS.		
The EHVS is described in Sections 8.1.2.1 and 8.3.1.1, and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The EHVS supports the medium voltage AC electrical distribution system by providing electrical power.	nonsafety-related	Component level tests
2. The EHVS supports the turbine generator system by providing electrical protection and control.	nonsafety-related	Component level tests
3. The EHVS supports the backup power supply system by providing electrical protection and control to the auxiliary AC power source.	nonsafety-related	Component level tests
4. The EHVS supports the offsite transmission system by providing electrical power during normal operation and configuration management of utility.	nonsafety-related	Component level tests
Prerequisites i. Verify an instrument calibration has been performed on all EHVS instruments that provide information signals to the plant control system (PCS) for the bus and main power transformer under test. ii. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. iii. Verify all protective devices associated with the EHVS bus and main power transformer under test are tested before that bus is energized, and approved test records indicate each protective device has been calibrated within its required test interval.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each EHVS breaker can be operated locally.	Operate each breaker from the local control panel while the breaker is in the test position.	Main control room (MCR) display and local, visual observation indicate each breaker opens and closes.
ii. Verify each EHVS breaker can be operated remotely.	Operate each breaker from the MCR while the breaker is in the test position.	MCR display and local, visual observation indicate each breaker opens and closes.
iii. Verify each EHVS breaker trips on its fault conditions.	Simulate each fault condition for a breaker when the breaker is in the test position.	MCR display and local, visual observation indicate each breaker opens on each fault condition.
iv. Verify each EHVS bus can be powered by offsite power via its main power transformer. (Test not required if an offsite power system is not provided.)	Energize each EHVS bus from its main power transformer.	Bus voltage is within design limits.
v. Verify each EHVS instrument is available on a module control system (MCS) or PCS display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each EHVS transmitter.	The instrument signal is displayed on an MCS or PCS display, or recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-55: Medium Voltage AC Electrical Distribution System Test # 55

Preoperational test is required to be performed once for the 0A medium voltage AC electrical distribution system (EMVS) and once for the 0B EMVS. The testing of each EMVS bus which provides power to 00 loads (common system loads) is performed with the EMVS loads of the first NuScale Power Module in power operation.		
The EMVS is described in Sections 8.1.2.1 and 8.3.1.1 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The EMVS supports the low voltage AC electrical distribution system by providing electrical power.	nonsafety-related	component-level tests
2. The EMVS supports the circulating water system by providing electrical power to loads.	nonsafety-related	component-level tests
3. The EMVS supports the chilled water system by providing electrical power to loads.	nonsafety-related	component-level tests
4. The EMVS supports the site cooling water system by providing electrical power to loads.	nonsafety-related	component-level tests
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify all protective devices associated with the EMVS bus and unit auxiliary transformer under test are tested before that bus is energized. Approved test records indicate each protective device has been calibrated within its required test interval.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each EMVS breaker can be operated locally.	Operate each breaker from the local control panel while the breaker is in the test position.	MCR display and local, visual observation indicate each breaker opens and closes.
ii. Verify each EMVS breaker can be operated remotely.	Operate each breaker from the MCR while the breaker is in the test position.	MCR display and local, visual observation indicate each breaker opens and closes.
iii. Verify each EMVS breaker trips on its fault conditions.	Simulate each fault condition for a breaker when the breaker is in the test position.	MCR display and local, visual observation indicate each breaker opens on each fault condition.
iv. a. Verify each EMVS bus can be powered via its unit auxiliary transformer. b.Verify each EMVS bus can be powered via an adjacent bus. (Test not required if an offsite power system is not provided.)	a. Energize each EMVS bus from its unit auxiliary transformer. b. Energize each EMVS bus from an adjacent EMVS bus.	Bus voltage is within design limits.
v. Verify each EMVS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each EMVS transmitter.	The instrument signal is displayed on an MCS or PCS display, or recorded by the applicable control system historian.

Table 14.2-55: Medium Voltage AC Electrical Distribution System Test # 55 (Continued)

vi. Verify the automatic transfer of each EMVS bus to each adjacent EMVS bus.	Simulate all conditions that require an automatic bus transfer to an adjacent bus. Repeat for each adjacent EMVS bus. This test may be performed with the EMVS bus energized or deenergized.	MCR display and local, visual observation indicate the required tie breaker from the adjacent bus closes.
System Level Tests		
None		

Table 14.2-56: Low Voltage AC Electrical Distribution System Test # 56

Preoperational test is required to be performed in support of the testing of each NuScale Power Module (NPM). The testing of each low voltage AC electrical distribution system (ELVS) bus which provides power to common system loads is performed with the first NPM tested. The testing of each ELVS bus which provides power to 0A loads is performed with the first 0A NPM tested. The testing of each ELVS bus which provides power to 0B loads is performed with the first 0B NPM tested.		
The ELVS is described in Section 8.1.2.1 and 8.3.1.1, and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The ELVS provides AC power to system loads via ELVS buses.	nonsafety-related	component-level tests
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify all protective devices associated with the ELVS bus and station service transformer under test are tested before that bus is energized.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each ELVS breaker can be operated locally.	Operate each breaker from the local control panel while the breaker is in the test position.	Main control room (MCR) display and local, visual observation indicate each breaker opens and closes.
ii. Verify each ELVS breaker can be operated remotely.	Operate each breaker from the MCR while the breaker is in the test position.	MCR display and local, visual observation indicate each breaker opens and closes.
iii. Verify each ELVS breaker trips on its fault conditions.	Simulate each fault condition for a breaker when the breaker is in the test position.	MCR display and local, visual observation indicate each breaker opens on each fault condition.
iv. Verify each ELVS bus can be powered by offsite power via its station service transformer. (Test not required if an offsite power system is not provided.)	Energize each ELVS bus from its station service transformer.	Bus voltage is within design limits.
v. Verify automatic bus transfer of each ELVS bus.	Perform the following test for each of the ELVS buses. Open the ELVS supply breaker to a given ELVS bus.	The associated ELVS bus tie breaker closes.
vi. Verify each ELVS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each ELVS transmitter.	The instrument signal is displayed on an MCS or PCS display, or recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-57: Highly Reliable DC Power System Test # 57

Component level tests are required to be performed for each NuScale Power Module (NPM), and once for the highly reliable DC power system (EDSS) common channels. **System Level Test #57-1 and Test #57-2 are required to be performed once.** **System Level Test #57-1 and Test #57-2 may be performed concurrently.** **System Level Test #57-3 is required to be performed once for each NPM.**		
The EDSS is described in Sections 8.1.2.2, 8.1.4.2 and 8.3.2.1.1, and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The EDSS supports the following systems by providing DC electrical power. • module protection system (MPS) • neutron monitoring system • fixed area radiation monitoring system • plant lighting system (PLS) • plant protection system (PPS) • safety display information system (SDIS) • normal control room HVAC system	nonsafety-related	All functions are verified by component-level tests. System level tests provide additional verification as follows: Plant Lighting System - Test #57-1 Safety Display and Indication System - Test #57-2 Module Protection System - Test #57-3
EDSS system functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. EDSS supports the MPS by providing EDSS module-specific operating parameter information signals.	nonsafety-related	SDIS Test #66, Component-level test: Module-Specific test iii.
2. EDSS supports the PPS by providing EDSS common operating parameter information signals.	nonsafety-related	SDIS Test #66, Component-level test: Common test iii.
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify a valve-regulated lead-acid battery acceptance tests has been performed on all EDSS batteries to confirm battery capacity in accordance with IEEE Standard 1188 Sections 6 and 7. iii. Verify battery charger performance testing has been completed by the manufacturer or a site acceptance test has been completed in accordance with manufacturer instructions.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each EDSS bus can be powered by its associated batteries.	Configure the EDSS batteries and battery chargers associated with an EDSS bus such that one of the batteries is the only source of power to the bus. Repeat the test using the other battery associated with the EDSS bus under test as the only power source. Repeat the test for the remaining EDSS channels.	EDSS bus voltage is within design limits.

Table 14.2-57: Highly Reliable DC Power System Test # 57 (Continued)

ii. Verify each EDSS bus can be powered by its associated battery chargers. (Test may be performed as part of site acceptance testing.)	Configure the EDSS batteries and battery chargers associated with an EDSS bus such that one of the battery chargers is the only source of power to the bus. Repeat the test using the other battery charger associated with the EDSS bus under test as the only power source. Repeat the test for the remaining EDSS channels.	EDSS bus voltage is within design limits.
iii. Verify each EDSS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each EDSS transmitter.	The instrument signal is displayed on an MCS or PCS display, or recorded by the applicable control system historian.
System Level Tests		
System Level Test #57-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the EDSS common buses provide independent power to the main control room (MCR) emergency lighting. (RG 1.41 Independence Test)	i. With both EDSS common buses energized and providing power to MCR emergency lighting, de-energize the EDSS Division I common bus. ii. With both EDSS common buses energized and providing power to MCR emergency lighting, de-energize the EDSS Division II common bus.	i. The MCR lighting designed to be powered by the EDSS Division I common bus is de-energized, and the MCR emergency lighting designed to be powered by the EDSS Division II common bus is energized. ii. The MCR emergency lighting designed to be powered by the EDSS Division II common bus is de-energized, and the MCR emergency lighting designed to be powered by the EDSS Division I common bus is energized.
System Level Test #57-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the EDSS common buses provide independent power to all SDIS MCR displays. (RG 1.41 Independence Test)	i. With EDSS Division I and Division II common buses energized verify power is available in the MCR for all SDIS displays. ii. De-energize the EDSS Division I common bus. iii. Re-energize the EDSS Division I common bus and de-energize the EDSS Division II common bus.	i. Power is available in the MCR for SDIS displays. ii.a. Power is not available in the MCR for SDIS Division I displays. ii.b. Power is available in the MCR for SDIS Division II displays. iii.a. Power is not available in the MCR for SDIS Division II displays. iii.b. Power is available in the MCR for SDIS Division I displays.

Table 14.2-57: Highly Reliable DC Power System Test # 57 (Continued)

System Level Test #57-3		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify EDSS module-specific channels provide independent and redundant power to the emergency core cooling system (ECCS) trip valve solenoids and post-accident monitoring (PAM) Type B and C variables. (RG 1.41 Independence Test)	i. With all EDSS module-specific channels de-energized for the NPM under test, energize EDSS module-specific channel A. ii. With all EDSS module-specific channels de-energized for the NPM under test, energize EDSS module-specific channel C. iii. With all EDSS module-specific channels de-energized for the NPM under test, energize EDSS module-specific channel B. iv. With all EDSS module-specific channels de-energized for the NPM under test, energize EDSS module-specific channel D.	i. Power is available to the Division I ECCS trip valve solenoids. ii.a. Power is available to the Division I ECCS trip valve solenoids. ii.b. All PAM Type B and C variables shown on Figure 7.1-2 are displayed on an SDIS display for the NPM under test. iii.a. Power is available to the Division II ECCS trip valve solenoids. iii.b. All PAM Type B and C variables shown on Figure 7.1-2 are displayed on an SDIS display for the NPM under test. iv. Power is available to the Division II ECCS trip valve solenoids.

Table 14.2-58: Normal DC Power System Test # 58

Component-level tests i.-xii. are required to be performed for the first NuScale Power Module (NPM). Component-level tests xiii.-xviii. are required to be performed once for the normal DC power system (EDNS) 0A Turbine Generator Building (TGB) subsystem and the first 0A NPM. Component-level tests xiii.-xxi, xix., and xx. are required to be performed once for the EDNS 0B TGB subsystem and the first 0B NPM. Component-level test xxi. is required to be performed once per EDNS subsystem. Component-level battery, battery charger, and inverter tests may be completed as part of site acceptance testing.		
EDNS is described in Sections 8.1.2.2, 8.1.4.2 and 8.3.2.1.2 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The EDNS supports the following systems by providing DC electrical power. • 13.8KV and switchyard system • medium voltage AC electrical distribution system • low voltage AC electrical distribution system • turbine generator system	nonsafety-related	Functions verified by prerequisite and component level tests.
The EDNS supports the following systems by providing AC electrical power. • communication system • control rod drive system • fire detection system • fire protection system • fixed area radiation monitoring system • meteorological and environmental monitoring system • module control system (MCS) • plant control system (PCS) • plant-wide video monitoring system • Reactor Building HVAC system • seismic monitoring system • Turbine Building HVAC system	nonsafety-related	Functions verified by prerequisite and component-level tests.
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii Verify a valve-regulated lead-acid battery acceptance tests has been performed on all EDNS batteries to confirm battery capacity in accordance with IEEE Standard 1188 Sections 6 and 7. iii. Verify battery charger performance testing has been completed by the manufacturer or a site acceptance test has been completed in accordance with manufacturer instructions. iv. Verify inverter performance testing has been completed by the manufacturer or a site acceptance test has been completed in accordance with manufacturer instructions.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each EDNS Reactor Building (RXB) subsystem DC bus can be powered by its associated battery.	Configure the battery and battery charger associated with one of the EDNS RXB subsystems such that the battery is the only source of power to its associated DC bus. Repeat the test for the other EDNS RXB subsystem.	EDNS DC bus voltage is within design limits.

Table 14.2-58: Normal DC Power System Test # 58 (Continued)

ii. Verify each EDNS RXB subsystem DC bus can be powered by its associated battery charger.	Configure the battery and battery charger associated with one of the EDNS RXB subsystems such that the battery charger is the only source of power to its associated DC bus. Repeat the test for the other EDNS RXB subsystem.	EDNS DC bus voltage is within design limits.
iii. Verify each EDNS RXB subsystem AC bus can be powered by its associated inverters.	Energize the AC bus of one of the EDNS RXB subsystems from the inverter source of one of its associated inverter. Repeat the test for the other inverter in the EDNS RXB subsystem under test. Repeat the test for the other EDNS RXB subsystem.	EDNS AC bus voltage is within design limits.
iv. Verify each EDNS RXB subsystem AC bus can be powered by its associated voltage regulating transformer.	Energize the AC bus of one of the EDNS RXB subsystems from the voltage regulating transformer source of one of its associated inverters. Repeat the test using the other inverter in the EDNS RXB subsystem under test. Repeat the test for the other EDNS RXB subsystem.	EDNS AC bus voltage is within design limits.
v. Verify the EDNS Control Building (CRB) subsystem DC bus can be powered by its associated battery.	Configure the battery and battery charger associated with the EDNS CRB subsystems such that the battery is the only source of power to its associated DC bus.	EDNS DC bus voltage is within design limits.
vi. Verify the EDNS CRB subsystem DC bus can be powered by its associated battery charger.	Configure the battery and battery charger associated with the EDNS CRB subsystem such that the battery charger is the only source of power to its associated DC bus.	EDNS DC bus voltage is within design limits.
vii. Verify each EDNS CRB subsystem AC bus can be powered by its associated inverter.	Energize the EDNS CRB subsystem AC bus from the inverter source of its associated inverter	EDNS AC bus voltage is within design limits.
viii. Verify each EDNS CRB subsystem AC bus can be powered by its associated voltage regulating transformer.	Energize the EDNS CRB subsystem AC bus from the voltage regulating transformer source of its associated inverter.	EDNS AC bus voltage is within design limits.
ix. Verify the EDNS Radioactive Waste Building (RWB) subsystem DC bus can be powered by its associated battery.	Configure the battery and battery charger associated with the EDNS RWB subsystems such that the battery is the only source of power to its associated DC bus.	EDNS DC bus voltage is within design limits.
x. Verify the EDNS RWB subsystem DC bus can be powered by its associated battery charger.	Configure the battery and battery charger associated with the EDNS RWB subsystem such that the battery charger is the only source of power to its associated DC bus.	EDNS DC bus voltage is within design limits.

Table 14.2-58: Normal DC Power System Test # 58 (Continued)

xi. Verify each EDNS Radioactive Waste Building (RWB) subsystem AC bus can be powered by its associated inverter.	Energize the EDNS RWB subsystem AC bus from the inverter source of its associated inverter.	EDNS AC bus voltage is within design limits.
xii. Verify each EDNS RWB subsystem AC bus can be powered by its associated voltage regulating transformer.	Energize the EDNS RWB subsystem AC bus from the voltage regulating transformer source of its associated inverter.	EDNS AC bus voltage is within design limits.
xiii. Verify the EDNS Turbine Generator Building (TGB) subsystem DC bus can be powered by its associated battery.	Configure the battery and battery charger associated with the EDNS TGB subsystems such that the battery is the only source of power to its associated DC bus.	EDNS DC bus voltage is within design limits.
xiv. Verify the EDNS TGB subsystem DC bus can be powered by its associated battery charger.	Configure the battery and battery charger associated with the EDNS TGB subsystem such that the battery charger is the only source of power to its associated DC bus.	EDNS DC bus voltage is within design limits.
xv. Verify each EDNS TGB subsystem AC bus can be powered by its associated inverter.	Energize the EDNS TGB subsystem AC bus from the inverter source of its associated inverter.	EDNS AC bus voltage is within design limits.
xvi. Verify each EDNS TGB subsystem AC bus can be powered by its associated voltage regulating transformer.	Energize the EDNS TGB subsystem AC bus from the voltage regulating transformer source of its associated inverter.	EDNS AC bus voltage is within design limits.
xvii. Verify the EDNS 0A power distribution center (PDC) subsystem DC buses can be powered by their associated batteries.	Configure the battery and battery chargers associated with the EDNS PDC subsystem #1 such that the battery is the only source of power to its associated DC bus. Repeat the test for EDNS PDC subsystems #3, #5, and #7.	EDNS DC bus voltage is within design limits.
xviii. Verify the EDNS 0A PDC subsystem DC buses can be powered by their associated battery chargers.	Configure the battery and battery chargers associated with EDNS PDC subsystem #1 such that one battery charger is the only source of power to its associated DC bus. Repeat the test for the other EDNS PDC subsystem #1 battery charger. Repeat the tests for EDNS PDC subsystems #3, #5, and #7.	EDNS DC bus voltage is within design limits.
xix. Verify the EDNS 0B PDC subsystem DC buses can be powered by their associated batteries.	Configure the battery and battery chargers associated with the EDNS PDC subsystem #2 such that the battery is the only source of power to its associated DC bus. Repeat the test for EDNS PDC subsystems #4, #6, and #8.	EDNS DC bus voltage is within design limits.

Table 14.2-58: Normal DC Power System Test # 58 (Continued)

xx. Verify the EDNS 0B PDC subsystem DC buses can be powered by their associated battery chargers.	Configure the battery and battery chargers associated with EDNS PDC subsystem #2 such that one battery charger is the only source of power to its associated DC bus. Repeat the test for the other EDNS PDC subsystem #2 battery charger. Repeat the tests for EDNS PDC subsystems #4, #6, and #8.	EDNS DC bus voltage is within design limits.
xxi. Verify each EDNS instrument is available on an MCS or PCS display. (Test not required if the instrument calibration verified the MCS or PCS display)	Initiate a single real or simulated instrument signal from each EDNS transmitter.	The instrument signal is displayed on an MCS or PCS display, or recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-59: Backup Power Supply System Test # 59

Preoperational test is required to be performed once.		
The backup power supply system (BPSS) is described in Sections 8.1.2.1 and 8.1.2.2 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The BPSS supports low voltage AC electrical distribution system (ELVS) by providing diesel generator backup electrical power to 480V B-6000 motor control centers.	nonsafety-related	Test #59-1
2. The BPSS supports ELVS by providing diesel generator backup electrical power to the operation of selected Reactor Building HVAC system components.	nonsafety-related	Test #59-1
3. The BPSS supports ELVS by providing diesel generator backup electrical power to the operation of selected normal DC power system components.	nonsafety-related	Test #59-1
4. The BPSS supports ELVS by providing diesel generator backup electrical power to the operation of selected normal control room HVAC system components.	nonsafety-related	Test #59-1
5. The BPSS supports 13.8 kV and switchyard system by providing auxiliary AC power source (AAPS) to the 13.8 kV bus.	nonsafety-related	Test #59-2
6. The BPSS supports the Diesel Generator Building HVAC system by providing diesel generator backup electrical power to the air handling units.	nonsafety-related	Test #59-1
Prerequisites i. Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. ii. Verify all protective devices associated with the BPSS diesel generators have been tested before performing this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each BPSS breaker can be operated locally.	Operate each breaker from the local control panel while the breaker is in the test position.	Main control room (MCR) display and local, visual observation indicate each breaker opens and closes.
ii. Verify each BPSS breaker can be operated remotely.	Operate each breaker from the MCR while the breaker is in the test position.	MCR display and local, visual observation indicate each breaker opens and closes.
iii. Verify the BPSS diesel generators can be started and stopped locally and remotely.	Align the BPSS to allow for diesel generator operation. i. Start and stop the diesel generator from the MCR. ii. Start and stop the diesel generator locally. Repeat the test for the other diesel generator.	i. and ii. MCR display and local, visual observation indicate the diesel generator started and stopped.

Table 14.2-59: Backup Power Supply System Test # 59 (Continued)

iv. Verify the BPSS diesel generator fuel oil transfer pump automatically maintains day tank level.	Align the fuel oil transfer pump to provide oil to the day tank. Simulate a low level in the day tank.	MCR display and local, visual observation indicate the transfer pump starts.
v. Verify the BPSS AAPS can be started and stopped locally and remotely.	Align the BPSS to allow for AAPS operation. i. Start and stop the AAPS from the MCR. ii. Start and stop the AAPS locally.	i. and ii. MCR display and local, visual observation indicate the AAPS started and stopped.
vi. Verify each BPSS instrument is available on a module control system (MCS) or plant control system (PCS) display. (Test not required if the instrument calibration verified the MCS or PCS display.)	Initiate a single real or simulated instrument signal from each BPSS transmitter.	The instrument signal is displayed on an MCS or PCS display, or recorded by the applicable control system historian.
System Level Test #59-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify BPSS diesel generator automatically starts and achieves rated voltage and frequency.	Align the BPSS to allow for diesel generator operation. Initiate a real or simulated loss of power signal.	MCR display and local, visual observation indicate the diesel generator started and achieved rated voltage and frequency.
System Level Test #59-2		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify BPSS AAPS automatically starts and achieves rated voltage and frequency.	Align the BPSS to allow for AAPS operation. Initiate a real or simulated loss of power signal.	MCR display and local, visual observation indicate the AAPS started and achieved rated voltage and frequency.

Table 14.2-60: Plant Lighting System Test # 60

Preoperational test is required to be performed once.		
The plant lighting system (PLS) is described in Section 9.5.3 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. PLS supports the Control Building (CRB) by providing normal lighting.	nonsafety-related	component-level test i.
2. The PLS supports the CRB by providing emergency lighting in the main control room (MCR).	nonsafety-related	component-level test ii.
3. The PLS supports the Reactor Building (RXB) by providing normal lighting.	nonsafety-related	component-level test i.
4. The PLS supports the RXB by providing emergency lighting for the remote shutdown station (RSS).	nonsafety-related	component-level test ii.
5. The PLS supports the RXB by providing emergency lighting for post-fire safe-shutdown activities outside of the MCR and RSS.	nonsafety-related	component-level test iii.
Prerequisites N/A (Note: Component level test iii. supports ITAAC and the requirements of NFPA 804.)		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify the PLS provides normal illumination of the MCR and RSS operator workstations, and the MCR safety display information panel.	With normal MCR and RSS lighting in service, measure the light at each MCR and RSS workstation.	i. a. The PLS provides at least 100 foot-candles illumination at the MCR operator workstations and at least 50 foot-candles at the MCR auxiliary panels. [ITAAC 03.08.01] i. b. The PLS provides at least 100 foot-candles illumination at the RSS operator workstations.
ii. The PLS provides emergency illumination of the MCR and RSS operator workstations and the MCR safety display information panel.	With MCR and RSS emergency illumination in service, measure the light at each MCR and RSS workstation and MCR safety display information panel.	ii. a. The PLS provides at least 10 foot-candles of illumination at the MCR operator workstations and the MCR safety display information panel. [ITAAC 03.08.02] ii. b. The PLS provides at least 10 foot-candles at the RSS operator workstations.
iii. Verify the eight-hour battery pack emergency lighting fixtures provide illumination for post-fire safe-shutdown activities performed by operators outside the MCR and RSS.	With no AC power available, measure the light at each eight-hour battery pack emergency lighting fixture target area.	iii. The required target areas are illuminated to provide at least one foot-candle illumination in the areas outside the MCR or RSS where post-fire safe-shutdown activities are performed. [ITAAC 03.08.03]
System Level Tests		
None		

Table 14.2-61: Module Control System Test # 61

Preoperational test is required to be performed as indicated by tests for module control system (MCS)-controlled systems and systems providing data to the MCS.		
The MCS is described in Section 7.0.4.5. The remote shutdown station (RSS) is described in Section 7.1.1.2.3.		
On-site testing of the system is performed by MCS site acceptance testing (SAT). The MCS is a distributed control system which allows monitoring and control of NuScale Power Module (NPM)-specific plant components. The MCS includes all manual controls and visual display units necessary to provide operator interaction with the process control mechanism. The boundary of the MCS is at the terminations on the MCS hardware. The MCS supplies nonsafety inputs to the human-system interfaces (HSIs) for nonsafety displays in the main control room (MCR), the RSS, and other locations where MCS HSIs are necessary. There are two boundaries between MCS and module protection system (MPS), the fiber-optic isolated portion and the hard-wired module boundary. The MCS has a direct, bi-directional interface with the plant control system (PCS). A complete staging and testing of system hardware and software configurations will be conducted. This factory acceptance testing (FAT) will be conducted in accordance with a written test procedure for testing the software and hardware of the MCS prior to installation in the plant. Following installation, SAT shall be completed in accordance with developed procedures to ensure the MCS is installed and fully functional as designed. To ensure the MCS communicates with module-specific plant components, component-level testing is performed on all systems controlled by MCS to manually operate the associated components from the main control room (MCR). By design, operation of components from the MCR also verifies the ability to operate the components from the RSS. These component-level tests are described in the test abstracts of the systems that contain the actuated components. In addition, it is verified that each instrument supplying data to the MCS is component tested in preoperational test abstracts to ensure the signal is available on an MCS or PCS display. These component-level tests are described in the test abstracts of the systems that contain the instrument. The RSS provides an alternate location to monitor the NPM status and operate the MCS and PCS during a MCR evacuation. The ability to activate the nonsafety MCS and PCS displays and controls at the RSS will be verified during SAT. The ability to isolate the safety-related MCR module protection system (MPS) manual switches using the MCR isolation switches in the RSS as described in Section 7.2.12 will be verified during MPS FAT and SAT.		
System Function	**System Function Categorization**	**Function Verified by Test #**
The MCS supports the Reactor Building by providing indication and operator display to the RSS, which resides in the Reactor Building.	nonsafety-related	FAT and SAT as described above.
Prerequisites		
Prerequisites associated with MCS testing are identified in the test abstracts that contain module-specific components that ensure communication with the MCS.		
Component Level Tests		
None		
System Level Tests		
None		

Table 14.2-62: Plant Control System Test # 62

Preoperational test is required to be performed as indicated by tests for plant control system (PCS)-controlled systems and systems providing data to the PCS.		
The PCS is described in Section 7.0.4.6. The Remote Shutdown Station (RSS) is described in Section 7.1.1.2.3.		
On-site testing of the system is performed by PCS site acceptance testing (SAT). The PCS is a distributed control system which allows monitoring and control of virtually all module-specific plant components. The PCS includes all manual controls and video display units (VDUs) necessary to provide operator interaction with the process control mechanism. The boundary of the PCS is at the terminations on the PCS hardware. The PCS supplies nonsafety inputs to the VDUs for nonsafety displays in the main control room (MCR), the RSS, and other locations where PCS video display units are necessary. The boundary between the plant protection system (PPS) and PCS is at the output connection of the safety-related optical isolators in the PPS, and on the terminals of the equipment interface module for each input from the PCS to the PPS. The PCS has a direct, bidirectional interface with the module control system (MCS). The network interface devices for the PCS domain controller/historian provide the interface between the human machine interface network layer and the control network layer. A complete staging and testing of system hardware and software configurations will be conducted. This factory acceptance testing (FAT) will be conducted in accordance with a written test procedure for testing the software and hardware of the PCS prior to installation in the plant. Following installation, SAT shall be completed in accordance with developed procedures to ensure the PCS is installed and fully functional as designed. To ensure the PCS communicates with module-specific plant components, component-level testing is performed on all systems controlled by PCS to manually operate the associated components from the MCR. By design, operation of components from the MCR also verifies the ability to operate the components from the RSS. These component-level tests are described in the test abstracts of the systems that contain the actuated components. In addition, it is verified that each instrument supplying data to the PCS is component tested in preoperational test abstracts to ensure the signal is available on an MCS or PCS display. These component-level tests are described in the test abstracts of the systems that contain the instrument. The RSS provides an alternate location to monitor the NuScale Power Module status and operate the MCS and PCS during a MCR evacuation. The ability to activate the nonsafety MCS and PCS displays and controls at the RSS will be verified during SAT. The ability to isolate the safety-related MCR module protection system (MPS) manual switches using the MCR isolation switches in the RSS as described in Section 7.2.12 will be verified during MPS FAT and SAT.		
System Function	**System Function Categorization**	**Function Verified by Test #**
The PCS supports the Reactor Building by providing indication and operator display to the RSS, which resides in the Reactor Building.	nonsafety-related	FAT and SAT as described above.
Prerequisites		
Prerequisites associated with PCS testing are identified in the test abstracts that contain module-specific components that ensure communication with or are controlled by the PCS.		
Component Level Tests		
None		
System Level Tests		
None		

Table 14.2-63: Module Protection System Test #63

<table>
<tr><td colspan="3">Preoperational test is required to be performed for each NuScale Power Module (NPM).</td></tr>
<tr><td colspan="3">The module protection system (MPS) is described in Sections 7.0, 7.1, and 7.2 and the functions verified by this test and power ascension testing are:</td></tr>
<tr><td>System Function</td><td>System Function Categorization</td><td>Function Verified by Test #</td></tr>
<tr><td>1. The MPS supports the containment system (CNTS) by removing electrical power to the trip solenoids of the following containment isolation valves (CIVs) on a CNTS isolation actuation signal:
• Reactor coolant system (RCS) injection CIVs
• RCS discharge CIVs
• Pressurizer (PZR) spray CIVs
• Reactor pressure vessel (RPV) high point degasification CIVs
• Feedwater isolation valves (FWIVs)
• Main steam isolation valves (MSIVs)
• Main steam isolation bypass valves (MSIBVs)
• Containment evacuation system CIVs
• Reactor component cooling water system CIVs RVVs
• Containment flooding and drain system (CFDS) CIVs</td><td>safety-related</td><td>Test #63-6</td></tr>
<tr><td>2. The MPS supports the CNTS by removing electrical power to the trip solenoids of the following valves on a decay heat removal system (DHRS) actuation signal.
• MSIVs
• MSIBVs
• FWIVs</td><td>safety-related</td><td>Test #63-6</td></tr>
<tr><td>3. The MPS supports the emergency core cooling system (ECCS) by removing electrical power to the trip solenoids of the following valves on an ECCS actuation signal.
• Reactor vent valves (RVVs)
• Reactor recirculation valves (RRVs)</td><td>safety-related</td><td>Test #63-6</td></tr>
</table>

Table 14.2-63: Module Protection System Test #63 (Continued)

4. The MPS supports the CNTS by removing electrical power to the trip solenoids of the following CIVs on a chemical and volume control system (CVCS) isolation actuation signal: • RCS injection CIVs • RCS discharge CIVs • PZR spray CIVs • RPV high point degasification CIVs	safety-related	Test #63-6
5. The MPS supports the CVCS by removing electrical power to the trip solenoids of the demineralized water system (DWS) supply isolation valves on a DWS isolation actuation signal.	safety-related	Test #63-6
6. The MPS supports the ECCS by removing electrical power to the trip solenoids of the reactor vent valves on an low temperature overpressure protection (LTOP) actuation signal.	safety-related	Test #63-6
7. The MPS supports the low voltage AC electrical distribution system by removing electrical power to the PZR heaters on a PZR heater trip actuation signal.	safety-related	Test #63-6
8. The MPS supports the normal DC power system by removing electrical power to the control rod drive system for a reactor trip.	safety-related	Test #63-5
9. The DHRS supports the RCS by opening the DHRS actuation valves on a DHRS actuation signal for DHRS operation.	safety-related	Test #63-6
10. The CNTS supports the DHRS by closing CIVs for the main steam and feedwater systems when actuated by the MPS.	safety-related	Test #63-6
11. The CNTS supports the RCS by closing the CIVs for PZR spray, RCS injection, RCS letdown, and RPV high point degasification when actuated by the MPS.	safety-related	Test #63-6
12. The CNTS supports the Reactor Building by providing a barrier to contain mass, energy, and fission product release by closure of the CIVs upon a containment isolation signal.	safety-related	Test #63-6
13. The ECCS supports the RCS by opening the ECCS RVVs and RRVs when their respective trip valve is actuated by the MPS.	safety-related	Test #63-6

Table 14.2-63: Module Protection System Test #63 (Continued)

14. The ECCS supports the RCS by providing recirculated coolant from the containment to the RPV for the removal of core heat.	safety-related	Test #63-6
15. The ECCS supports the RCS by providing LTOP for maintaining the reactor coolant pressure boundary.	safety-related	Test #63-6
16. The CVCS supports the RCS by isolating dilution sources.	safety-related	Test #63-6
17. The feedwater system (FWS) supports the CNTS by providing secondary isolation of the feedwater lines.	nonsafety-related	Test #63-6
18. The main steam system supports the CNTS by providing secondary isolation of the main steam lines.	nonsafety-related	Test #63-6
19. The FWS supports the DHRS by providing secondary isolation of the feedwater lines, ensuring required boundary conditions for DHRS operation.	nonsafety-related	Test #63-6
20. The neutron monitoring system supports the MPS by providing neutron flux data for various reactor trips.	safety-related	Test #63-1 Neutron Monitoring System Power Range Flux Calibration Test #92
21. ECCS supports MPS by providing instrumentation information signals.	nonsafety-related	Test #63-1
22. The DHRS supports the MPS by providing MPS actuation instrument information signals.	safety-related	Test #63-1
23. The RCS supports the MPS by providing instrument information signals.	nonsafety-related	Test #63-1
24. The RCS supports the MPS by providing instrument information signals for LTOP actuation	safety-related	Test #63-1
25. The CVCS supports the RCS by providing primary coolant makeup in beyond design basis events.	nonsafety-related	Test #63-11
26. The CFDS supports the RCS by providing borated coolant inventory for the removal of core heat during a beyond design basis accident.	nonsafety-related	Test #63-11
27. The MPS supports the DHRS by removing electrical power to the trip solenoids of the DHRS actuation valves on a DHRS actuation signal.	safety-related	Test #63-6
28. The MPS supports the CNTS by providing power to sensors.	safety-related	Test #63-1
29. The MPS supports the DHRS by providing power to sensors.	safety-related	Test #63-1
30. The MPS supports the RCS by providing power to sensors.	safety-related	Test #63-1

Table 14.2-63: Module Protection System Test #63 (Continued)

Prerequisite Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
None		
System Level Test #63-1		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the instrument signals of MPS monitored variables are displayed in the MCR.	Table 7.1-2 lists all of sensors which input to MPS. This test may be performed concurrently with safety display and indication system test #66 -2 for post-accident monitoring Type B and Type C testing described in Section 14.2.12. Inject a single signal as close as practical for each sensor listed in Table 7.1-2 and monitor its response on an MCR workstation and the module-specific safety display instrument panel (if designed for safety display instrument display). If the sensor signal is designed to be disconnected when the NPM is moved then it will be necessary to test the signal from the sensor to the disconnect and then from the disconnect to the MCR display.	Each MPS monitored signal is displayed on an MCR workstation and the module-specific safety display instrument panel (if designed for safety display instrument display).
System Level Test #63-2 (Not Used)		
System Level Test #63-3		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify each ECCS RVV and RRV operates to satisfy its engineered safety feature (ESF)-actuated design stroke time.	This test will verify the stroke time of each RRV and RVV by actuating the valves with RCS pressure below the IAB RCS pressure threshold. i. Close all RVVs and RRVs. ii. Verify RCS pressure is below the IAB RCS pressure threshold specified in Technical Specifications (TS). iii. Actuate ECCS using the manual ECCS actuation switches in the MCR.	Each ECCS RRV and RVV travels from fully closed to fully open in less than or equal to the time specified in TS.
System Level Test #63-4 Test #63-4 is performed concurrently with Test #63-6 which operates all of the ESF actuation valves during hot functional testing. Test #63-4 records the stroke times of DHRS actuation valves as they travel to their ESF-actuated position with the RCS pressure at normal operating pressure.		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify each DHRS actuation valve operates to satisfy its ESF-actuated design stroke time.	Time the operation of all DHRS actuation valves as they actuate to their ESF position during the manual ESF actuation testing in Test #63-6.	Each DHRS actuation valve travels from fully closed to fully open in less than or equal to the time specified in Technical Specifications.
System Level Test #63-5 (Not Used)		

Table 14.2-63: Module Protection System Test #63 (Continued)

<table>
<tr><td colspan="3">System Level Test #63-6
Test 63-6 is performed at hot functional testing concurrently with Turbine Generator System Test #33-1 (reference 14.2.12.33) to allow testing of ESF actuations at normal operating pressure and elevated temperatures. Test #33-1 heats the RCS from ambient conditions to the highest temperature achievable by module heatup system heating. These hot functional testing conditions provide the highest differential pressure and temperature conditions that can be achieved prior to fuel load.</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>i. Verify the MPS can manually actuate ESF equipment from the MCR.
ii. Verify deliberate operator action is required to return the ESF actuated equipment to its non-actuated position.</td><td>Table 7.1-4 lists all of the ESF functions.
All ESF functions are tested.
The RCS is at normal operating pressure supplying bypass steam to the condenser.
i. Initiate a manual ESF actuation signal from the MCR.
ii. a. Attempt to operate the actuated ESF equipment from the MCR.
b. Remove the manual ESF actuation signal and attempt to operate the actuated ESF equipment from the MCR.
c. Use the MCR enable nonsafety control switch to allow operation of the ESF actuated equipment from the MCR.
Repeat for all MCR manual ESF actuations.</td><td>i. The MPS actuates the ESF equipment to perform its safety-related function as described in Table 7.1-4.
Each ECCS valve opens after receipt of an ESF signal and after RCS pressure is decreased to the threshold pressure for operation of the inadvertent actuation block described in Section 6.3.2.2.
[ITAAC 02.01.13]
[ITAAC 02.01.14]
[ITAAC 02.01.15]
[ITAAC 02.01.18]
[ITAAC 02.01.19]
[ITAAC 02.01.20]
[ITAAC 02.05.11]
[ITAAC 02.05.13]
[ITAAC 02.05.16]
ii. a. The actuated equipment cannot be operated from the MCR.
b. The actuated equipment cannot be operated from the MCR.
c. The ESF equipment can be operated from the MCR.
[ITAAC 02.01.13]
[ITAAC 02.01.14]
[ITAAC 02.01.15]
[ITAAC 02.05.16]</td></tr>
<tr><td colspan="3">System Level Test #63-7
Test #63-7 is performed concurrently with Test #63-6 which operates all of the ESF actuation valves during hot functional testing.
Test #63-7 records the stroke times of CIVs as they travel to their ESF-actuated position with the RCS pressure at normal operating pressure.</td></tr>
<tr><td>Test Objective</td><td>Test Method</td><td>Acceptance Criteria</td></tr>
<tr><td>Verify the CIVs operate to satisfy their ESF-actuated design stroke time.</td><td>Table 6.2-5 contains the design closure time for containment isolation valves.
Time the operation of all CIVs as they actuate to their ESF position during the manual ESF actuation testing in Test #63-6.</td><td>i. Each CIV travels from fully open to fully closed in less than or equal to the time listed in Table 6.2-5 after receipt of a containment isolation signal.
[ITAAC 02.01.08]
[ITAAC 02.05.17]</td></tr>
</table>

Table 14.2-63: Module Protection System Test #63 (Continued)

Test Objective	Test Method	Acceptance Criteria
System Level Test #63-8		
This test will verify the time response of MPS reactor trip and ESF actuation signals. The reactor trip test verifies response time through reactor trip breaker actuation. The ESF response time is tested through the de-energization of the associated solenoid valve or the opening of the PZR heater supply breaker. ESF valve response times are tested in Test #63-7.		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the MPS response times from sensor output through: i. reactor trip breaker actuation for the reactor trip function. ii. de-energization of the associated solenoid valve for ESF-actuated valves. iii. opening of the PZR heater supply breaker for the pressurizer heater trip.	Section 7.1.4 contains a description of design basis event actuation delays assumed in the plant safety analysis and listed in Table 7.1-6. The actuation delays do not include ESF actuated component delays for actuated valves. Perform a time response test for the actuation signals listed in Table 7.1-6.	The MPS reactor trip functions listed in Table 7.1-3 and ESF functions listed in Table 7.1-4 have response times that are less than or equal to the design basis safety analysis response time assumptions in Table 7.1-6. [ITAAC 02.05.17]
System Level Test #63-9 (Not Used)		
System Level Test #63-10		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify MCR alarms when automatic operating bypasses are established. ii. Verify MCR alarms when manual operating bypasses are established. iii. Verify MPS maintenance bypasses are indicated in the MCR.	The purpose of this test is to verify MCR alarms, not to verify the logic of the operating bypasses. Any signal that establishes the bypass can be used. Table 7.1-5 contains a list of operating bypasses. i. For automatically established operating bypasses perform the following: a. Simulate the logic required to establish the operating bypass. b. Remove the logic. c. Repeat for each automatically established operating bypass. ii. For manually established operating bypasses perform the following: a. Simulate the logic required to allow the operating bypass to be established. b. Manually establish the operating bypass. c. Repeat the logic. d. Repeat for each manually established operating bypass. iii. a. Place an safety function module (SFM) in maintenance bypass by using the out of service and trip/bypass switches associated with the SFM. b. Repeat tests for all SFMs.	i. Each automatic operating bypass is alarmed in the MCR. [ITAAC 02.05.22] ii. Each manual operating bypass is alarmed in the MCR. [ITAAC 02.05.22] iii. The inoperable status of the SFM is provided in the MCR. [ITAAC 02.05.23]

Table 14.2-63: Module Protection System Test #63 (Continued)

System Level Test #63-11		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify the controls located on the operator workstations in the MCR operate to perform important human actions.	Test is performed to verify operation of redundant trains and divisions. i. Use MPS Division I controls, boron addition system (BAS) supply pump A train, and CVCS makeup pump A train to perform the following: a. Insert a manual containment isolation signal. b. Override the containment isolation signal using the CNTS isolation override switch. c. Enable nonsafety controls using the enable nonsafety control switch. d. Using an operator workstation in the MCR, align the BAS, CVCS, and CNTS to add inventory to the RCS. ii. Use MPS Division II controls, BAS supply pump B train, and CVCS makeup pump B train to perform the following: a. Insert a manual containment isolation signal. b. Override the containment isolation signal using the CNTS isolation override switch. c. Enable nonsafety controls using the enable nonsafety control switch. d. Using an operator workstation in the MCR, align the BAS, CVCS, and CNTS to add inventory to the RCS. iii. Use MPS Division I controls and CFDS pump A train to perform the following: a. Insert a manual containment isolation signal. b. Override the containment isolation signal using the CNTS isolation override switch.	i. Water is added to the RCS. ii. Water is added to the RCS. iii. Water is added to containment. iv. Water is added to containment. [ITAAC 02.05.26] (i., ii., iii., and iv.)

Table 14.2-63: Module Protection System Test #63 (Continued)

	c. Enable nonsafety controls using the enable nonsafety control switch. d. Using an operator workstation in the MCR, align the CFDS and CNTS to add inventory to containment. iv. Use MPS Division II controls and CFDS pump B train to perform the following: a. Insert a manual containment isolation signal. b. Override the containment isolation signal using the CNTS isolation override switch. c. Enable nonsafety controls using the enable nonsafety control switch. d. Using an operator workstation in the MCR, align the CFDS and CNTS to add inventory to containment.	

Table 14.2-64: Plant Protection System Test # 64

Preoperational test is required to be performed once.		
The plant protection system (PPS) is described in Section 7.0.4.3. PPS functions are not verified by PPS tests. PPS functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The PPS supports the normal control room HVAC system (CRVS) by providing actuation and control signals to the control room envelope isolation dampers.	nonsafety-related	Control Room Habitability System Test #18-1
2. The PPS supports the control room habitability system by providing actuation and control signals.	nonsafety-related	Control Room Habitability System Test #18-1
3. The PPS supports the CRVS by providing actuation and control signals to the outside air isolation dampers.	nonsafety-related	Normal Control Room HVAC System Test #19-3
Prerequisites Verify an instrument calibration has been completed, with approved records and calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each variable monitored by PPS is available on a module control system (MCS) or plant control system (PCS) display.	Initiate a single real or simulated instrument signal from each transmitter monitored by PPS.	Each variable listed in Table 7.1-8 is displayed on an MCS or PCS display, or is recorded by the applicable control system historian.
System Level Tests		
None		

Table 14.2-65: Neutron Monitoring System Test # 65

Preoperational test is required to be performed for each NuScale Power Module.		
The neutron monitoring system (NMS) is described in Section 7.0.4.2 and the functions verified by this test and power ascension testing are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The NMS supports the module protection system (MPS) by providing neutron flux data for various reactor trips.	safety-related	Test #63-1 Neutron Monitoring System Power Range Flux Calibration Test #92
2. The NMS supports the MPS by providing information signals for post-accident monitoring (PAM).	nonsafety-related	Test #66-2
3. The NMS supports the MPS by providing information signals for PAM during containment vessel flooded conditions.	nonsafety-related	Test #66-2
Prerequisites Prerequisites associated with NMS testing are identified in the referenced test abstract cited under the “Function Verified by Test #” heading.		
Component Level Tests		
None		
System Level Tests		
None		

Table 14.2-66: Safety Display and Indication System Test # 66

Test #66 Component-level testing for the module-specific safety display and indication system (SDIS) is required to be performed for each NuScale Power Module (NPM). Test #66 Component-level testing for the common SDIS is required to be performed once. Test #66-1 System-level testing for the module-specific SDIS is required to be performed for each NPM to verify proper trending of reactor coolant system (RCS) pressure and temperature. Test #66-2 System-level testing for the module-specific SDIS is required to be performed for each NPM to verify post-accident monitoring (PAM) variables are displayed and alarms retrieved.		
SDIS is described in Section 7.0.4.4 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The SDIS actively supports the Control Building by providing the main control room (MCR) accident monitoring plant conditions.	nonsafety-related	i. Module-specific SDIS component-level tests ii. Common SDIS component-level tests iii. Test #66-1 iv. Test #66-2
2. The SDIS actively supports the plant control system (PCS) by providing plant status and indication data to the plant data historian.	nonsafety-related	i. Module-specific SDIS component-level tests ii. Common SDI component-level tests iii. Test #66-1 iv. Test #66-2
3. The in-core instrumentation system supports the module protection system (MPS) by providing reactor core temperature information.	nonsafety-related	Test #66-2
4. The emergency core cooling system (ECCS) supports MPS by providing PAM instrument information signals.	nonsafety-related	Test #66-2
5. The RCS supports the MPS by providing PAM instrument information signals.	nonsafety-related	Test #66-2
6. The containment system supports the MPS by providing PAM information signals.	nonsafety-related	Test #66-2
7. The fixed area radiation monitoring system supports the Reactor Building by monitoring radiation levels in the building in proximity of the bioshield.	nonsafety-related	Test #66-2
8. The neutron monitoring system (NMS) supports the MPS by providing information signals for PAM.	nonsafety-related	Test #66-2
9. The NMS supports the MPS by providing information signals for PAM during containment vessel flooded conditions.	nonsafety-related	Test #66-2
10. The decay heat removal system supports the MPS by providing PAM instrument information signals.	nonsafety-related	Test #66-2
11. The highly reliable DC power system (EDSS) supports the plant protection system (PPS) by providing common EDSS operating parameter information signals.	nonsafety-related	Component-level test: Common SDIS test iii.

Table 14.2-66: Safety Display and Indication System Test # 66 (Continued)

12. The EDSS supports the MPS by providing module-specific EDSS operating parameter information signals.	nonsafety-related	Component-level test: Module-Specific SDIS test iii.
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test. **Note:** Testing of PAM Type B and Type C displays and alarms is performed in Test #66-1. **Note:** Testing of NPM level, pressure, and temperature and flow instruments is performed in Test #66-2.		
Component Level Tests: Common SDIS Test		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify the proper valve position indication for each valve that provides input to the PPS.	Open and close each valve listed in Table 7.1-8.	The valve opens and closes as indicated by a common SDIS display and an MCR workstation display.
ii. Verify radiation monitor indication is obtained in the MCR for each radiation monitor that provides input to the PPS.	Provide a simulated signal for each radiation monitor monitored by PPS listed in Table 7.1-8.	The radiation signal is displayed by a common SDIS display and an MCR workstation.
iii. Verify EDSS and low voltage AC electrical distribution system (ELVS) voltage indication is obtained in the MCR for voltmeters that provide input to the PPS.	Provide a simulated signal for each EDSS and ELVS voltmeter monitored by PPS listed in Table 7.1-8.	The voltage signal is displayed by a common SDIS display and an MCR workstation.
iv. Verify instrument indication is obtained in the MCR for instruments that provide input to the PPS.	Provide a simulated signal for each instrument monitored by PPS listed in Table 7.1-8.	The instrument signal is displayed by a common SDIS display and an MCR workstation.
Component Level Tests: Module Specific SDI Test		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify the proper valve position indication for each engineered safety feature valves that provide input to MPS.	i. With the NPM assembled, open and close the valves listed in Table 7.1-2. ii. Provide a real or simulated signal for each reactor safety valve position (Table 7.1-2).	i. The valves open and close as indicated by a module-specific SDIS display and an MCR workstation display. ii. The valve opens and closes as indicated by a module-specific SDIS display and an MCR workstation display.
ii. Verify radiation monitor indication is obtained in the MCR for each radiation monitor that provides input to the MPS.	Provide a simulated signal for each radiation monitor monitored by MPS listed in Table 7.1-2.	The radiation monitor signal is displayed by a module-specific SDIS display and an MCR workstation.
iii. Verify EDSS and ELVS voltage indication is obtained in the MCR for each voltmeter that provide input to the MPS.	Provide a simulated signal for each EDSS and ELVS voltmeter monitored by MPS (Table 7.1-2).	The voltage signal is displayed by a module-specific SDIS display and an MCR workstation.
iv. Verify neutron flux indication is obtained in the MCR for each radiation monitor that provides input to the MPS.	Provide a simulated signal for each neutron flux instrument monitored by MPS (Table 7.1-2).	The neutron flux signal is displayed by a module-specific SDIS display and an MCR workstation display.

Table 14.2-66: Safety Display and Indication System Test # 66 (Continued)

v. Verify a neutron flux instrument fault indication is obtained in the MCR for each signal that provides input to the MPS.	Provide a simulated signal for each neutron flux instrument fault monitored by MPS (Table 7.1-2).	The neutron flux instrument fault is displayed by a module-specific SDIS display and an MCR workstation display.
System Level Test #66-1 **Test 66-1 is conducted concurrently with Turbine Generator System Test# 33-1 which warms the RCS from ambient conditions to the highest temperature achievable by module heatup system (MHS) heating.**		
Test Objective	**Test Method**	**Acceptance Criteria**
Verify that the output signals from the NPM level, pressure, temperature and flow instruments listed in Table 7.1-2 properly trend while increasing RCS temperature and pressure. Note: This is not a verification of instrument calibrations.	Increase RCS temperature from ambient to the highest temperature achievable by MHS heating. Using the module control system historian record the engineering values for the output of the instruments described in the test objective. Record data at approximately 50 °F intervals from ambient temperature to the maximum RCS temperature. Note: Instrument signals are provided to the module-specific SDIS display and the main control room workstations.	All instruments track within acceptable design limits. (Use Technical Specification channel check limits, when applicable)
System Level Test #66-2		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify PAM Type B and C variables are displayed on the module-specific SDIS displays in the MCR. ii. Verify alarms associated with PAM Type B and C variables are retrieved in the MCR. iii. Verify module-specific PAM Type D variables are displayed on the module-specific SDIS displays in the MCR.	i. Simulate an injection signal for the PAM Type B and C variables listed in Table 7.1-7. ii. Increase or decrease a simulated injection signal for the PAM Type B and C variables listed in Table 7.1-7 to obtain its associated alarm. iii. Simulate an injection signal for the PAM Type D variables listed in Table 7.1-7	i. The PAM Type B and C variables listed in Table 7.1-7 are retrieved and displayed on the SDI displays in the MCR. [ITAAC 02.05.25] ii. The alarms associated with the PAM Type B and C variables listed in Table 7.1-7 are retrieved and displayed on the SDI displays in the MCR. iii. The PAM Type D variables listed in Table 7.1-7 are retrieved and displayed on the SDIS displays in the MCR.

Table 14.2-67: Fixed-Area Radiation Monitoring System Test # 67

Preoperational test is required to be performed once.		
The fixed-area radiation monitoring system (RMS) is described in Section 12.3.4 and the function verified by this test is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The **RMS** supports the following buildings by monitoring radiation levels: • Annex Building • Control Building • Radioactive Waste Building • Turbine Building • Reactor Building (RXB)	nonsafety-related	Component-level test
RMS function verified by another test is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The RMS supports the RXB by monitoring radiation levels in the building in proximity of the bioshield.	nonsafety-related	**Safety Display and Indication System** Test #66-2
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify each fixed airborne radiation monitor's response to an alarm condition.	Actuate the check source on a fixed airborne radiation monitor listed in Table 12.3-11. Repeat test for the remainder of fixed airborne radiation monitors.	Main control room (MCR) display and local, visual observation indicate the following: i. The MCR audible and visual alarms are received. ii. The local readout, audible alarm and visual alarm are received.
ii. Verify each fixed area radiation monitor's response to an alarm condition.	Actuate the check source on a fixed area radiation monitor listed in Table 12.3-12. Repeat test for the remainder of fixed area radiation monitors.	MCR display and local, visual observation indicate the following: i. The MCR audible and visual alarms are received. ii. The local readout, audible alarm and visual alarm are received.
System Level Tests		
None		

Table 14.2-68: Communication System Test # 68

Preoperational test is required to be performed after construction turnover of the communication system (COMS).		
The COMS is described in Section 9.5.2 and the function verified by this test is:		
System Function	**System Function Categorization**	**Function Verified by Test #**
The COMS supports the following locations by providing voice and data communications within the building and surrounding areas. • Reactor Building • Turbine Building • Radioactive Waste Building • Control Building • Security Buildings • Annex Building • Diesel Generator Building • Administrative and Training Building • Central Utility Building • Warehouse Building • Fire Water Building • Switchyard • Site plant cooling structures • Site water intake/discharge structure • Site utility rack structure	nonsafety-related	Component-level tests i. through iv.
Prerequisites i. Required COMS site acceptance tests have been completed and approved.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify plant public address/general alarm (PA/GA) system can be heard throughout the plant site.	Station test personnel in each required test area of the plant to monitor the PA/GA system. i. Use the public address to provide a test announcement. ii. Use the general alarm system to provide a test alarm.	i. The test announcement is heard at each test site. ii. The test emergency alarm is heard at each test site.
ii. Verify plant radio communications can be heard throughout the plant site.	Station test personnel in each required test area of the plant to communicate using plant radios.	The plant radio communication is obtained at each test site.
iii. Verify the sound powered telephone system can be used for voice communication.	Test each sound powered telephone.	All channels of each sound powered telephone can be used to communicate with another sound powered telephone.
iv. Verify wireless communication throughout the plant site.	Station test personnel in each required test area of the plant to communicate using voice and data communication.	The voice and data communication is obtained at each test site.
v. Verify the central alarm station is equipped with a conventional (landline) telephone service which can be used to communicate with the main control room (MCR) and local law enforcement authorities.	Test the conventional (landline) service from the central alarm station to the MCR and local law enforcement authorities.	The conventional service connects with the MCR and the local law enforcement authorities. [ITAAC 03.16.11]

Table 14.2-68: Communication System Test # 68 (Continued)

vi. Verify that plant radio communications maintains continuous communications between the central alarm station and 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.	Test communications with the plant radio system in areas described in the physical protection program boundaries and areas described in the contingency response event areas.	The radios will provide continuous communications in all test areas. [ITAAC 03.16.12]
vii. Verify all nonportable communication devices (including conventional telephone systems) in the central alarm station remain operable during the loss of normal power.	Remove normal power from the central alarm station nonportable communication devices.	The nonportable communication devices establish connections with the normal power removed. [ITAAC 03.16.13]
System Level Tests		
None		

Table 14.2-69: Seismic Monitoring System Test # 69

The seismic monitoring system (SMS) is described in Section 3.7.4. The SMS is a site-specific system, and the SMS design is the responsibility of the COL applicant as indicated by COL item 3.7-1. COL Item 14.2-6: A COL applicant that references the NuScale Power Plant design certification will provide a test abstract for the seismic monitoring system pre-operational testing.		
System Function	**System Function Categorization**	**Function Verified by Test #**
As described in Section 3.7.4	nonsafety-related	Provided by COL applicant
Prerequisites Provided by COL applicant		
Component Level Tests		
Provided by COL applicant		
System Level Tests		
Provided by COL applicant		

Table 14.2-70: Hot Functional Testing Test # 70

Preoperational testing is required to be performed once for each NuScale Power Module (NPM).			
The following identifies the tests employed in support of the performance of hot functional testing.			
Hot Functional Testing Tests	**Test Objective**	**Verified by Test #**	**Tested Function Categorization**
containment evacuation system (CES)	i. Verifies the automatic operation of the CES to establish and maintain design vacuum for the containment vessel. ii. Verify radiation isolation on high radiation level in the auxiliary boiler system. iii. Verifies the CES supports reactor coolant system (RCS) leakage detection. iv. Verifies the CES level and pressure instrumentation support leak-before-break leakage detection.	i. Containment Evacuation System Test #41-1 ii. Containment Evacuation System Test #41-2 iii. Containment Evacuation System Test #41-3 iv. Containment Evacuation System Test #41-4	nonsafety-related
containment system (CNTS)	i. Verifies each CNTS safety-related check valves open and close under preoperational conditions.	i. Containment System Test #43-2	safety-related
chemical and volume control system (CVCS)	i. Verifies CVCS automatic operation to maintain pressurizer level. ii. Verifies automatic pressurizer pressure control. iii. Verifies CVCS automatic boration and dilution of the RCS.	i. Chemical and Volume Control System Test #38-1 ii. Chemical and Volume Control System Test #38-2 iii. Chemical and Volume Control System Test #38-3	nonsafety-related
emergency core cooling system (ECCS)	i. Each ECCS valve opens after receipt of an engineered safety feature (ESF) signal and after RCS pressure is decreased to the threshold pressure for operation of the inadvertent actuation block.	i. Module Protection System Test #63-6	safety-related
feedwater system (FWS)	i. Verifies the FWS automatically controls flow to the steam generators (SGs) to maintain SG inventory. ii. Verifies the FWS automatically cools the turbine generator bypass steam flow in the main steam desuperheater.	i. Turbine Generator System Test #33-1 ii. Turbine Generator System Test #33-1	nonsafety-related

Table 14.2-70: Hot Functional Testing Test # 70 (Continued)

in-core instrumentation system (ICIS)	i. Verifies proper temperature indication is obtained from the in-core instrumentation system thermocouples.	i. In-Core Instrumentation System Test #49-1	nonsafety-related
liquid radioactive waste system (LRWS)	i. Verifies the LRWS receives and processes a gaseous stream from the pressurizer.	i. Liquid Radioactive Waste System Test #35-1	nonsafety-related
module heatup system (MHS)	i. Verifies the MHS is capable of heating the RCS to a temperature sufficient to obtain criticality. ii. Verifies the MHS is capable of heating the RCS to establish natural circulation flow sufficient to obtain criticality. iii. Verifies a local grab sample can be obtained from an MHS grab sample device indicated on the MHS piping and instrumentation diagram.	i. Turbine Generator System Test #33-1 ii. Turbine Generator System Test #33-1 iii. Turbine Generator System Test #33-1	nonsafety-related
module protection system (MPS)	i. Verifies design responses to manual ESF signals. ii. Verifies containment isolation valves closure times.	i. Module Protection System Test #63-6 ii. Module Protection System Test #63-7	safety-related
process sampling system	i. Verifies sampling capability of the primary sampling points. ii. Verifies sampling capability of the containment sampling points. ii. Verifies sampling capability of the secondary sampling points.	i. Process Sampling System Test #53-1 ii. Process Sampling System Test #53-2 iii. Process Sampling System Test #53-3	nonsafety-related
safety display and indication system (SDIS)	i. Verify that the output signals from the NPM level, pressure, temperature, and flow instruments listed in Table 7.1-2 properly trend while increasing RCS temperature and pressure.	i. Safety Display and Indication System Test #66-1	nonsafety-related

Table 14.2-70: Hot Functional Testing Test # 70 (Continued)

TGS	i. Verifies the TGS automatically controls turbine bypass flow to the main condenser. ii. Verify the maximum main turbine speed that can be obtained using the MHS to heat the RCS. iii. Verifies the ECCS valves close when the CVCS provides water to reset the ECCS valves.	i. Turbine Generator System Test #33-1 ii. Turbine Generator System Test #33-2 iii. Turbine Generator System Test #33-1	nonsafety-related
Prerequisites			
Prerequisites associated with performing hot functional testing are identified in the referenced test abstract cited under the "Verified by Test #" heading.			

Table 14.2-71: Module Assembly Equipment Bolting Test # 71

Preoperational test is required to be performed once.		
The module assembly equipment (MAE) bolting is described in Section 9.1.5. MAE bolting functions are not verified by MAE bolting tests. MAE bolting functions verified by other tests are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. MAE bolting supports the containment system actively by providing material handling to allow for disassembly and reassembly of the containment vessel lower flange.	nonsafety-related	Test #52-2
2. MAE bolting supports the reactor pressure vessel (RPV) actively by providing material handling to allow for disassembly and reassembly of the RPV lower flange.	nonsafety-related	Test #52-2
Prerequisites Prerequisites associated with MAE bolting testing are identified in the referenced test abstract cited under the "Function Verified by Test #" heading.		
Component Level Tests		
None		
System-Level Tests		
None		

Table 14.2-72: Steam Generator Flow-Induced Vibration Test # 72

This is a one-time test to be performed prior to loading fuel in the first ever NuScale Power Module (NPM). There are no preoperational tests for the steam generator system.		
Validation testing is performed at test facilities as separate effects tests on prototypic steam generator (SG) tube columns and SG inlet flow restrictors per Table 4-1 of TR-0716-50439.		
The SG flow-induced vibration testing is performed consistent with the requirements of the NuScale Comprehensive Vibration Assessment Program (CVAP) as described in the "NuScale Comprehensive Vibration Assessment Program Technical Report," TR-0716-50439, and the "NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report" TR-0918-60894. The SG tube column testing consists of in-air and in-water modal testing and primary side flow testing. The SG inlet flow restrictor testing consists of in-air and in-water modal testing and secondary side flow testing. Visual examination of the SG components is performed as specified in Table 5-1 of TR-0716-50439. The CVAP is addressed in Section 3.9.2. The steam generators (SGs) are discussed in Section 5.4.1.		
System Function	**System Function Categorization**	**Function Verified by Test #**
None	N/A	N/A
Prerequisites: N/A		
Component Level Tests		
None		
Acceptance Criteria: i. The SG tube column testing shows that fluid elastic instability and vortex shedding do not occur under primary side flow rates consistent with any operating condition, considering all applicable uncertainties and biases of this separate effects test. Expected safety margins for the test facility and for the NPM SG assembly at 100 percent power operating conditions are provided in Sections 5.1.3 and 5.1.4 of TR-0618-60894. ii. The SG tube column testing shows that for primary side flow rates consistent with 100 percent power operation, the SG tube vibration responses are less than those predicted with the turbulent buffeting analysis methodology. More details are provided in Section 5.1.5 of TR-0618-60894. iii. The SG inlet flow restrictor testing is completed in accordance with Section 5.3 of TR-0918-60894 and results confirm the lack of leakage flow instability for the design.		

Table 14.2-73: Security Access Control Test # 73

Preoperational test is required to be performed once.		
Security access control is described in "NuScale Design of Physical Security Systems", TR-0416-48929.		
System Function	**System Function Categorization**	**Function Verified by Test #**
The security access controls support the security plan described in "NuScale Design of Physical Security Systems", TR-0416-48929.	security-related	Component level test i.
Prerequisites i. Security access control boundary for the protected and vital areas, described in the security technical report, are established.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Verify an access control system with a numbered photo identification badge system which will control access to vital areas within the Reactor Building (RXB) and Control Building (CRB) to authorized personnel.	Use authorized and unauthorized identification badges in all vital area access points in the RXB and CRB identified in "NuScale Design of Physical Security Systems", TR-0416-48929.	i. The access points do not allow access to unauthorized badges. ii. The access points allow authorized personnel. [ITAAC 03.16.04]
System Level Tests		
None		

Table 14.2-74: Security Detection and Alarm Test # 74

Preoperational test is required to be performed once.		
Security detection and alarm is described in "NuScale Design of Physical Security Systems", TR-0416-48929.		
System Function	**System Function Categorization**	**Function Verified by Test #**
The security detection and alarm system acts to satisfy the functional requirements described in "NuScale Design of Physical Security Systems", TR-0416-48929.	security-related	Component level test i-v
Prerequisites i. Required security system site acceptance tests have been completed and approved.		
Component Level Tests		
Test Objective	**Test Method**	**Acceptance Criteria**
i. Unoccupied vital areas will be designed with locking devices and intrusion detection devices that annunciate in the central alarm station.	Access to all unoccupied vital areas that are identified in the "NuScale Design of Physical Security Systems", TR-0416-48929.	i. Verify the access door is locked. ii. Upon entry into the room verify an intrusion alarm is received in the central alarm station. [ITAAC 03.16.05]
ii. Security alarm devices including transmission lines to annunciators are tamper-indicating and self-checking.	a. Insert a signal real or simulated tamper signal. b. Insert a signal real or simulated of a component failure for all alarm devices and transmission lines in the Reactor Building (RXB) and Control Building (CRB). c. Place all security alarm devices in the RXB and CRB on standby power.	Verify alarm annunciation is received in the central alarm station for each test method. The alarm must indicate the type and location of the alarm. [ITAAC 03.16.07]
iii. Intrusion detection and assessment systems provides visual and audible alarm annunciation in the central alarm station.	Put all intrusion detection equipment described in "NuScale Design of Physical Security Systems", TR-0416-48929 into an alarm state.	Verify an audible and visual alarm is received in the central alarm station. [ITAAC 03.16.08]
iv. Intrusion detection system recording equipment will record onsite security alarm annunciation including false alarm, alarm check, and tamper indication and the type of alarm, location, alarm circuit, date, and time.	Place all intrusion detection equipment in the RXB and CRB in the following alarm conditions (as applicable to the equipment): a. False alarm b. Alarm check c. Tamper indication	Verify the intrusion detection system recording system records each alarm to include: a. Location of the alarm b. Type of alarm c. Alarm circuit d. Date e. Time (this test can be done in conjunction with audible and visual alarm testing) [ITAAC 03.16.09]
v. Emergency exits in the RXB and CRB will be alarmed with intrusion detection devices and secured by locking devices that allow prompt egress during an emergency.	i. Attempt to enter each the RXB and CRB exits. ii. Exit each of the RXB and CRB exits.	i. Verify the locking device prevents entry. ii. Verify the exit allows for prompt exit of the building and alarms in the central alarm station when opened. [ITAAC 03.16.10]
System Level Tests		
None		

Table 14.2-75: Initial Fuel Loading Precritical Test # 75

Startup test is required to be performed for each NuScale Power Module.
This test is performed after initial fuel loading but prior to initial criticality.
Test Objectives
i. Identify the sequence for precritical testing (after fuel load and prior to criticality). ii. The pre-critical tests are: a. Reactor Coolant System Flow Measurement Test #77 b. NuScale Power Module Temperatures Test #78 c. Primary and Secondary System Chemistry Test #79 d. Control Rod Drive System - Manual Operation, Rod Speed, and Rod Position Indication Test #80 e. Control Rod Assembly Full-Height Drop Time Test #81 f. Control Rod Assembly Ambient Temperature Full-Height Drop Time Test #81A g. Pressurizer Spray Bypass Flow Test #82
Prerequisites
None
Test Method
i. Identify the specific plant conditions required for each precritical test procedure to maintain Technical Specification (TS) operability. ii. Identify the prerequisites required for each precritical test procedure. iii. Determine the test sequence for precritical testing based on TS requirements and test prerequisites.
Acceptance Criterion
The sequence for precritical testing has been determined.

Table 14.2-76: Initial Fuel Load Test # 76

The Initial Fuel Load Test is required to be performed for each NuScale Power Module (NPM).
This test is performed prior to initial fuel load.
Test Objectives
i. Conduct initial fuel load with no inadvertent criticality. ii. Install fuel assemblies and control components at the locations specified by the design of the initial reactor core.
Prerequisites
i. Plant systems required for initial fuel loading have completed preoperational testing. ii. Plant systems required for initial fuel loading have been aligned per operations procedures. iii. The design of the initial RXC that specifies the final core configuration of fuel assemblies and control components is completed. iv. A core load sequence has been approved. v. Neutron monitoring data from a previous NPM initial fuel loading or calculations showing the predicted response of monitoring channels are available for evaluating monitoring data. vi. The lower reactor pressure vessel (RPV) is installed in the RPV support stand. vii. Reactor Building radiation monitors are functional. viii. Boron concentration in the pool is within Technical Specifications (TS) limits. ix. The nuclear instrumentation system is calibrated and operable.
Test Method
i. Install fuel and control components per approved procedures. ii. Monitor boron concentration inside the RPV periodically during fuel load to ensure it satisfies TS. iii. Monitor neutron counts during the load of each fuel assembly and plot an independent inverse count rate ratio for each source range detector after each fuel load assembly is loaded. iv. Verify neutron count data are consistent with calculations showing the predicted response. For fuel loading of the second NPM and all subsequent NPMs use data obtained from previous fuel loadings. v. Demonstrate the inverse count rate ratio does not show significant approach to criticality. vi. Maintain the status of the core loading.
Acceptance Criterion
i. Each fuel assembly and control component is installed in the location specified by the design of the initial reactor core. ii. There is no indication of inadvertent criticality.

Table 14.2-77: Reactor Coolant System Flow Measurement Test # 77

The Reactor Coolant System Flow Measurement Test is required to be performed for each NuScale Power Module (NPM).
This test is performed after initial fuel loading but prior to initial criticality.
Test Objective
Verify that the reactor coolant system (RCS) flow is sufficient to ensure adequate boron mixing in the RCS coolant.
Prerequisites
i. The core is installed. ii. The NPM is fully assembled. iii. The RCS is at hot zero power (RCS at normal operating pressure with RCS temperature at the maximum temperature obtainable when heated only by the module heatup system). iv. The RCS flow meters have been calibrated.
Test Method
Record RCS flow using main control room indication.
Acceptance Criterion
The RCS flow at hot zero power satisfies the minimum RCS flow assumed in the safety analysis.

Table 14.2-78: NuScale Power Module Temperatures Test # 78

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed after initial fuel loading but prior to initial criticality.
Test Objectives
i. Perform a cross calibration of the resistance temperature detectors (RTDs) monitored by the module protection system (MPS) listed in Table 7.1-2. ii. Verify incore thermocouple resistance leakage satisfies manufacturer's criteria.
Prerequisites
i. The core is installed. ii. The NPM is fully assembled. iii. The calibration of reactor coolant system RTDs has been completed.
Test Method
i. With the reactor coolant system (RCS) at ambient temperature and isothermal conditions record the following data: • Main control room (MCR) indication of RTD temperatures monitored by MPS • MCR indication of incore thermocouples temperatures • Leakage resistance of the incore thermocouples ii. Increase RCS temperature by approximately 50°F. iii. Record RTD and incore thermocouple data at isothermal conditions. iv. Repeat data collection until RCS temperature is at the highest temperature obtainable using only the module heatup system. v. Cross-calibrate RTD temperatures monitored by MPS that monitor the same variable.
Acceptance Criteria
i. The cross calibration of the reactor coolant system RTDs has been completed. ii. The leakage resistance of the fixed incore detectors satisfies manufacturer's recommendations.

Table 14.2-79: Primary and Secondary System Chemistry Test # 79

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed prior to criticality and at approximately 25, 50, 75, and 100 percent reactor thermal power.
Test Objective
Verify water quality in the primary system and secondary system using the process sampling system (PSS).
Prerequisites
i. The PSS instruments have been calibrated. ii. The NPM is fully assembled. iii. The reactor coolant system (RCS) is at hot zero power (RCS at normal operating pressure and RCS temperature at the maximum temperature obtainable when heated only by the module heatup system).
Test Method
i. Use the PSS to sample the normal primary system sample points listed in Table 9.3.2-1. ii. Use the PSS to sample the normal secondary system sample points listed in Table 9.3.2-3. iii. To the extent practical, responses of PSS radiation monitors should be verified by laboratory analyses of grab samples taken at the same process location. iv. Conduct the test prior to criticality and at steady-state condition at approximately 25, 50, 75, and 100 percent reactor thermal power.
Acceptance Criterion
The sample analyses satisfy the limits specified in plant procedures.

Table 14.2-80: Control Rod Drive System - Manual Operation, Rod Speed, and Rod Position Indication Test # 80

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed after initial fuel loading but prior to initial criticality.
Test Objectives
i. Verify the ability to manually fully insert and fully withdraw individual control rod assemblies (CRAs) from the main control room (MCR). ii. Verify CRA rod position indications provide indication of rod movement. iii. Verify individual CRA position indications are within the required number of steps of their associated group position. iv. Verify the rod insertion and withdrawal speeds are within design limits.
Prerequisites
i. The core is installed. ii. The NPM is fully assembled. iii. The reactor coolant system (RCS) is at hot zero power (RCS at normal operating pressure and RCS temperature at the maximum temperature obtainable when heated only by the module heatup system). iv. All RCS temperatures satisfy the minimum Technical Specification (TS) temperature for criticality. v. The nuclear instrumentation system is calibrated and operable. vi. The shutdown margin is within the limits specified in the core operating limits report.
Test Method
i. Individually withdraw and insert each shutdown bank and regulating bank from the MCR a sufficient number of steps to verify that the individual CRA positions are within the required number of steps of their group position as required by TS. Only the tested bank will be withdrawn. All other banks are fully inserted. Repeat the test until all shutdown banks and regulating banks are tested. ii. With all shutdown and regulating banks fully inserted, fully withdraw and then fully insert one CRA. Repeat these steps until all CRAs are tested.
Acceptance Criteria
i. All CRAs can be individually fully withdrawn and fully inserted from the MCR. ii. Individual CRA position indications are within the number of steps of their associated group position as required by TS. iii. The CRA insertion and withdrawal speeds are within the design limits identified in Section 3.9.4.1.

Table 14.2-81: Control Rod Assembly Full-Height Drop Time Test # 81

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed after initial fuel loading but prior to initial criticality.
Test Objective
Verify each control rod assembly (CRA) satisfies the CRA drop time acceptance criteria for reactor coolant system (RCS) flow at 0 percent reactor thermal power.
Prerequisites
i. The core is installed. ii. The NPM is fully assembled. iii. The RCS is at hot zero power (RCS at normal operating pressure and RCS temperature at the maximum temperature obtainable when heated only by the module heatup system). iv. All RCS temperatures satisfy the minimum Technical Specification (TS) temperature for criticality. v. The nuclear instrumentation system is calibrated and operable. vi. The shutdown margin is within the limits specified in the core operating limits report.
Test Method
i. Fully withdraw each individual CRA. ii. Interrupt the electrical power to the associated control rod drive mechanism. iii. Measure the CRA drop time.
Acceptance Criteria
i. Each CRA drop time is within TS limits. ii. Each CRA drop time is within two sigma of the drop time data for all control rods, or has been verified within TS limits by a minimum of three additional performances of this test.

Table 14.2-81a: Control Rod Assembly Ambient Temperature Full-Height Drop Time Test #81A

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed after initial fuel loading but prior to initial criticality.
Test Objective
Verify each control rod assembly (CRA) satisfies the CRA drop time acceptance criteria for reactor coolant system (RCS) at ambient temperature.
Prerequisites
i. The core is installed. ii. The NPM is fully assembled. iii. The RCS is at cold temperature conditions. iv. The nuclear instrumentation system is calibrated and operable. v. The shutdown margin is within the limits specified in the core operating limits report.
Test Method
i. Fully withdraw each individual CRA. ii. Interrupt the electrical power to the associated control rod drive mechanism. iii. Measure the CRA drop time.
Acceptance Criteria
i. Each CRA drop time is within Technical Specification (TS) limits. ii. Each CRA drop time is within two sigma of the drop time data for all control rods, or has been verified within TS limits by a minimum of three additional performances of this test.

Table 14.2-82: Pressurizer Spray Bypass Flow Test # 82

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed after initial fuel loading but prior to initial criticality.
Test Objective
Verify the pressurizer (PZR) spray bypass flow rate is adequate to prevent thermal fatigue of the spray line components and provide sufficient mixing in the PZR to maintain PZR water chemistry similar to the rest of the reactor coolant system (RCS) while avoiding unnecessary energization of the pressurizer heaters.
Prerequisites
i. The core is installed. ii. The NPM is fully assembled. iii. The RCS is at hot zero power (RCS at normal operating pressure and RCS temperature at the maximum temperature obtainable when heated only by the module heatup system).
Test Method
i. With the automatic PZR spray valve closed, adjust the manual spray bypass valve to maintain a continuous spray bypass flow of approximately one gpm. ii. If the continuous bypass spray flow requires the operation of the PZR backup heaters to maintain the PZR pressure setpoint, throttle close the bypass valve until PZR pressure is maintained by the proportional heaters.
Acceptance Criterion
The spray bypass valve flow satisfies design requirements.

Table 14.2-83: Initial Criticality Test # 83

Startup test is required to be performed for each NuScale Power Module.
This test is performed after initial fuel loading.
Test Objective
Achieve initial criticality in a controlled manner.
Prerequisites
i. The reactor coolant system (RCS) is at hot zero power (RCS at normal operating pressure and RCS temperature at the maximum temperature obtainable when heated only by the module heatup system). ii. All RCS temperatures satisfy the minimum Technical Specification (TS) temperature for criticality. iii. The nuclear instrumentation system is calibrated and operable. iv. The shutdown margin is within the limits specified in the core operating limits report. v. An estimated critical position (calculation) has been performed. vi. RCS measured boron is at or near the desired estimated critical position value. vii. The shutdown banks and the regulating banks are fully inserted. viii. A neutron count rate of at least 1/2 counts per second registers on the startup channels, and the signal to noise ratio is greater than 2.
Test Method
i. Shutdown banks are withdrawn in sequence using the sequence of a normal plant startup. Gather data to plot the inverse count rate ratio. The inverse count rate ratio is used to monitor reactivity. ii. Once all shutdown banks are fully withdrawn, then the regulating bank is withdrawn using the sequence of a normal plant startup. The inverse count rate ratio is plotted to monitoring reactivity for the approach to criticality. iii. After criticality is obtained, the regulating bank is confirmed to be above the TS regulating group insertion limit. iv. Should criticality be reached with the regulating bank below the insertion limit specified by the core operating limits requirement, the limiting condition of operation test exception is invoked. The RCS boron will be increased until the regulating bank is withdrawn sufficiently to meet the insertion limit.
Acceptance Criterion
The reactor is critical with the regulating banks above their TS insertion limit.

Table 14.2-84: Post-Critical Reactivity Computer Checkout Test # 84

Startup test is required to be performed for each NuScale Power Module.
This test is performed after initial criticality.
Test Objective
Verify proper operation of the reactivity computer to measure reactivity changes in the core during low-power testing.
Prerequisites
i. The reactor is critical with the neutron flux level within the range for low-power physics testing. ii. The reactor coolant system (RCS) temperature and pressure are stable at the normal no-load values. iii. The neutron flux level and RCS boron concentration are stable. iv. The reactivity computer is installed and internal reactivity computer checks have been completed.
Test Method
i. Withdraw the regulating bank to achieve a positive startup rate below Technical Specification limits. ii. Measure the reactor period or doubling time. iii. Reinsert the regulating bank to re-establish the initial steady-state neutron flux. iv. Measure the negative reactor period or halving time. v. Validate the core response against the reactivity computer input delayed neutron fractions and prompt neutron lifetime using pre-determined test criteria. vi. Adjust and recalibrate reactivity computer until acceptance criteria is met.
Acceptance Criterion
The reactivity computer is calibrated.

Table 14.2-85: Low-Power Test Sequence Test # 85

Startup test is required to be performed for each NuScale Power Module.
This test is performed before initial criticality.
Test Objectives
i. Identify the sequence for low-power testing. ii. The low-power tests are: a. Determination of Zero-Power Physics Testing Range Test #86 b. All Rods Out Boron Endpoint Determination Test #87 c. Isothermal Temperature Coefficient Measurement Test #88 d. Bank Worth Measurement Test #89
Prerequisites
None
Test Method
For each of the tests identified in the test objectives above: i. Identify the specific plant conditions required for each low-power test procedure to maintain Technical Specification (TS) operability. ii. Identify the prerequisites required for each low-power test procedure. iii. Determine the test sequence for low-power testing based on TS requirements and test prerequisites.
Acceptance Criterion
The sequence for low-power testing has been determined.

Table 14.2-86: Determination of Zero-Power Physics Testing Range Test # 86

Startup test is required to be performed for each NuScale Power Module.
This test is performed after initial criticality.
Test Objectives
i. Determine the reactor flux level at which the point of nuclear heating is detectable. ii. Establish the range of neutron flux in which hot zero power (HZP) reactivity measurements are to be performed.
Prerequisites
i. The reactor is critical with the neutron flux level at steady-state below the expected level of nuclear heating. ii. The reactor coolant system (RCS) temperature and pressure is steady-state at the normal HZP conditions. iii. The RCS boron concentration is steady-state. iv. The reactivity computer is operational and recording the core average neutron flux level. v. The regulating bank is positioned to allow reactivity changes by rod motion alone.
Test Method
i. Withdraw the regulating bank to establish a slow startup rate allowing neutron flux level to increase until nuclear heating is observed. ii. Record the reactivity computer neutron flux level and the corresponding main control room flux indication at which nuclear heating occurs. iii. Insert the regulating bank to establish a reactivity computer flux level about one-third of the value at which nuclear heating was observed. This flux level becomes the maximum value for the zero-power testing range.
Acceptance Criterion
The zero power testing range flux level is determined.

Table 14.2-87: All Rods Out Boron Endpoint Determination Test # 87

Startup test is required to be performed for each NuScale Power Module.
This test is performed after initial criticality.
Test Objective
Determine the critical reactor coolant system (RCS) boron concentration for all rods out (ARO) (fully withdrawn shutdown banks and regulating banks) at hot zero power (HZP).
Prerequisites
i. The reactor is critical with the neutron flux level at steady-state below the expected level of nuclear heating. ii. The RCS temperature and pressure is steady-state at the normal HZP conditions. iii. The RCS boron concentration is steady-state. iv. The reactivity computer is operational and recording the core average neutron flux level.
Test Method
i. Add a pre-determined volume of borated water to the RCS and withdraw the regulating bank to maintain critical conditions. The final regulating bank position will be near fully withdrawn and will limit the usable positive reactivity remaining in the rods with the reactor critical. ii. Measure the just-critical boron concentration by chemical analysis. iii. Fully withdraw the regulating bank without adjusting the boron concentration. Measure and calculate the change in reactivity for ARO and the RCS temperature difference from program T_{AVG}, due to an equivalent change in boron concentration. Add the equivalent boron change to the just-critical boron concentration to yield the endpoint for ARO.
Acceptance Criterion
The measured value for the ARO boron endpoint satisfies the design value contained within the test acceptance criteria.

Table 14.2-88: Isothermal Temperature Coefficient Measurement Test # 88

Startup test is required to be performed for each NuScale Power Module.
This test is performed after initial criticality.
Test Objectives
i. Determine the isothermal temperature coefficient. ii. Calculate the moderator temperature coefficient.
Prerequisites
i. The reactor is critical with the neutron flux level at steady-state below the expected level of nuclear heating. ii. The reactor coolant system (RCS) temperature and pressure is steady-state at the normal hot zero power conditions. iii. The RCS boron concentration is steady-state. iv. The reactivity computer is operational and recording the core average neutron flux level. v. The regulating rod bank is positioned near fully withdrawn (near their all rods out position).
Test Method
i. Vary RCS temperature (heatup/cooldown) while maintaining rods and boron concentration constant. ii. Monitor reactivity results and determine the isothermal temperature coefficient. iii. Calculate the moderator temperature coefficient using the isothermal temperature coefficient and design values.
Acceptance Criterion
The moderator temperature coefficient is within the limits specified in the core operating limits report.

Table 14.2-89: Bank Worth Measurement Test # 89

Startup test is required to be performed for each NuScale Power Module.
This test is performed after initial criticality.
Test Objectives
i. Measure the integral and differential worth of the reference bank (the test bank with the highest predicted worth). ii. Measure the worth of the remaining shutdown and regulating banks by control rod exchange (rod swap).
Prerequisites
i. The reactor is critical with the neutron flux level at steady-state within the range for hot zero power (HZP) physics testing. ii. The reactor coolant system (RCS) temperature and pressure is steady-state at the normal HZP conditions. iii. The RCS boron concentration is steady-state. iv. The reactivity computer is operational and recording the core average neutron flux level. v. The regulating rod banks are positioned near fully withdrawn (near their all rods out position).
Test Method
i. The referenced bank rod worth measurement is made by performing a slow controlled boron dilution while the reference bank is inserted to maintain criticality. The rod worth is measured using the reactivity computer. During boron dilution the reference bank step insertions maintain neutron flux within the zero-power physics test range until the referenced bank is fully inserted. ii. A test bank rod worth measurement is made by inserting the test bank while the reference bank is withdrawn. The test bank worth is determined by the final position of the referenced bank.
Acceptance Criterion
The measured worth for each individual bank, and sum of bank worths, is consistent with the predicted value within the test acceptance criteria.

Table 14.2-90: Power-Ascension Test # 90

Startup test is required to be performed for each NuScale Power Module.
This test is performed prior to power-ascension testing.
Test Objective
Identify the sequence for the following power-ascension tests. a. Core Power Distribution Map Test #91 b. Neutron Monitoring System Power Range Flux Calibration Test #92 c. Reactor Coolant System Temperature Instrument Calibration Test #93 d. Reactor Coolant System Flow Calibration Test #94 e. Radiation Shield Survey Test #95 f. Reactor Building Ventilation System Capability Test #96 g. Thermal Expansion Test #97 h. Control Rod Assembly Misalignment Test #98 i. Steam Generator Level Control System Test #99 j. Ramp Change in Load Demand Test #100 k. Step Change in Load Demand Test #101 l. Loss of Feedwater Heater Test #102 m. 100 Percent Load Rejection Test #103 n. Reactor Trip from 100 Percent Power Test #104 o. Island Mode Test for the First NuScale Power Module (Test #105) p. Island Mode Test for Multiple NuScale Power Modules (Test #106) q. NuScale Power Module Vibration Test #108
Prerequisites
None
Test Method
i. Identify the specific plant conditions required for each power-ascension test procedure to maintain Technical Specification (TS) operability. ii. Identify the prerequisites required for each power-ascension test procedure. iii. Determine the test sequence for power-ascension testing based on TS requirements and test prerequisites.
Acceptance Criterion
The sequence for power-ascension testing has been determined.

Table 14.2-91: Core Power Distribution Map Test # 91

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at approximately 25, 50, 75, and 100 percent reactor thermal power
Test Objectives
i. Obtain a core power distribution map during power ascension. ii. Using the data from the core power distribution map verify core power distribution is consistent with design predictions and associated Technical Specifications (TS) limits.
Prerequisites
i. The in-core instrumentation system is operational. ii. The NPM is operating in a steady-state condition at the specified power level. iii. Maintain reactor power, T_{AVG}, and pressurizer level constant during data collection.
Test Method
i. With the plant at power levels of approximately 25, 50, 75, and 100 percent of reactor thermal power, obtain a core power distribution map during power ascension using the module control system and instrument input from the in-core self-powered neutron detectors. ii. Use data from the in-core maps to verify that core power distribution is consistent with design predictions and TS limits.
Acceptance Criterion
Core power distribution is consistent with design predictions and TS limits.

Table 14.2-92: Neutron Monitoring System Power Range Flux Calibration Test # 92

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at approximately 25, 50, 75 and 100 percent reactor thermal power.
Test Objective
Calibrate the neutron monitoring system (NMS) power range neutron flux signals during power ascension.
Prerequisites
i. The in-core instrumentation system (ICIS) is operational. ii. The NPM is operating in a steady-state condition at the specified power level.
Test Method
i. With the plant at power levels of approximately 25, 50, 75 and 100 percent of reactor thermal power, record the following data: • power range neutron flux from the ICIS self-powered neutron detectors • NMS power range (linear power) signal • heat balance data ii. Maintain reactor power, T_{AVG}, and pressurizer level constant during data collection. iii. Calibrate the NMS neutron flux power range (linear power) signal using the recorded data.
Acceptance Criterion
The NMS neutron flux power range (linear power) signal has been calibrated.

Table 14.2-93: Reactor Coolant System Temperature Instrument Calibration Test # 93

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at approximately 25, 50, 75, and 100 percent reactor thermal power.
Test Objective
Calibrate narrow range reactor coolant system (RCS) hot leg temperature instruments, wide range RCS hot leg temperature instruments, and narrow range RCS cold leg temperature instruments.
Prerequisites
i. The in-core instrumentation system (ICIS) is operational. ii. The NPM is operating in a steady-state condition at the specified power level.
Test Method
i. With the plant at power levels of approximately 25, 50, 75, and 100 percent of reactor thermal power, record the following data: • neutron monitoring system flux power range (linear power) signal • RCS narrow range hot leg temperature • RCS wide range hot leg temperature • RCS narrow range cold leg temperature • ICIS core inlet and outlet temperature ii. Maintain reactor power, T_{AVG}, and pressurizer level at steady-state during data collection. iii. Calibrate the RCS narrow range and wide range hot leg temperature instruments and the RCS narrow range cold leg temperature using the recorded data.
Acceptance Criterion
The RCS hot and cold let temperature instruments have been calibrated.

Table 14.2-94: Reactor Coolant System Flow Calibration Test # 94

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at approximately 25, 50, 75, and 100 percent reactor thermal power.
Test Objective
Calibrate the reactor coolant system (RCS) flow instruments during power ascension.
Prerequisites
i. The in-core instrumentation system (ICIS) is operational. ii. The NPM is operating in a steady-state condition at the specified power level. iii. The nuclear instrumentation system is calibrated and operable.
Test Method
i. With the plant at power levels of approximately 25, 50, 75, and 100 percent of reactor thermal power, record the following data: • neutron monitoring system flux power range (linear power) signal • RCS narrow range hot leg temperature • RCS narrow range cold leg temperature • ICIS core inlet and outlet temperature ii. Maintain reactor power, T_{AVG}, and pressurizer level at steady state during data collection. iii. Calibrate the RCS flow instruments using the recorded data.
Acceptance Criterion
The RCS flow instruments have been calibrated.

Table 14.2-95: Radiation Shield Survey Test # 95

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at approximately 25, 50, and 100 percent reactor thermal power.
Test Objective
Verify the adequacy of radiation shields in the Reactor Building (RXB) designed to protect personnel from radiation originating from sources within the reactor vessel.
Prerequisites
i. Radiation survey instruments are calibrated. ii. The NPM is operating in a steady-state condition at the specified power level.
Test Method
i. Measure gamma and neutron radiation dose rates at designated locations at approximately 25, 50, and 100 percent reactor thermal power in accordance with RG 1.69 and ANSI/ANS-6.3.1 (1987, R2007). ii. The designated locations are the accessible areas outside permanent radiation shields in the RXB.
Acceptance Criterion
Radiation dose rates are consistent with design expectations.

Table 14.2-96: Reactor Building Ventilation System Capability Test # 96

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at approximately 50 and 100 percent reactor thermal power.
Test Objective
Verify that the Reactor Building HVAC system (RBVS) maintains the design environment in areas containing equipment that is environmentally qualified for a harsh or mild environment.
Prerequisite
The NPM is operating in a steady-state condition at the specified power level.
Test Method
i. With the plant at power levels of approximately 50 and 100 percent of reactor thermal power and RBVS in normal lineup, record temperature and humidity for the environmental qualification zones listed in Table 3.11-2 that are not under the bioshield. ii. With the plant at power levels of approximately 50 and 100 percent of reactor thermal power and RBVS in normal lineup, record the temperature and humidity in the rooms containing electrical equipment qualified for a mild environment.
Acceptance Criteria
i. Room temperature and humidity in environmental qualification zones listed in Table 3.11-2 that are not under the bioshield satisfy the indoor design conditions for the RBVS contained in Table 9.4.2-2. ii. Room temperature and humidity in rooms containing electrical equipment qualified for a mild environment satisfy the indoor design conditions for the RBVS contained in Table 9.4.2-2.

Table 14.2-97: Thermal Expansion Test # 97

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed during plant heatup and cooldown.
Test Objectives
i. Verify that ASME Code Class 1, 2, and 3 system piping can expand without obstruction and that expansion is within design limits. All ASME Code Class 1, 2, and 3 system piping is within the Reactor Building (RXB). ii. Verify that high-energy piping inside the RXB can expand without obstruction and that expansion is within design limits.
Prerequisite
Temporary instrumentation is installed on piping outside the NPM as required to monitor the deflections for the piping under test.
Test Method
i. Thermal expansion testing is performed in accordance with ASME OM Standard, Part 7 as discussed in Section 3.9.2.1. ii. Record deflection data during plant heatup and cooldown. iii. Identify support movements by recording hot and cold positions of the supports. (Note: piping testing will be determined in COL Item 3.9-13)
Acceptance Criteria
In accordance with ASME OM Standard, Part 7, for the piping systems tested: i. There is no evidence of blocking of the thermal expansion of piping or component, other than by installed supports, restraints, and hangers. ii. Spring hanger movements must remain within the hot and cold setpoints and supports must not become fully retracted or extended. iii. Piping and components return to their approximate baseline cold position.

Table 14.2-98: Control Rod Assembly Misalignment # 98

Startup test is required to be performed for each NuScale Power Module.
This test is performed at approximately 50 and 100 percent reactor thermal power.
Test Objectives
i. Verify that core thermal and nuclear parameters at 50 and 100 percent reactor thermal power are in accordance with predictions with a single high-worth rod fully inserted, during rod movement, and following return of the rod to its bank position. ii. Verify the capability of the in-core neutron flux instrumentation to detect a control rod misalignment equal to or less than the Technical Specification (TS) limits at 50 and 100 percent reactor thermal power. iii. Monitor the power distribution following the recovery of a misaligned control rod assembly (CRA).
Prerequisites
i. The reactor is operating at steady-state conditions and has been at that condition for a sufficient time to reach xenon equilibrium. ii. The reactor power level, reactor coolant system boron concentration, and temperature are stable. iii. The regulating and shutdown banks are positioned as required for the specific measurement, near fully withdrawn for CRA insertion, and at their respective insertion limits for CRA withdrawal.
Test Method
i. For the CRA insertion, insert a group of selected CRAs, one at a time, first to the limit of misalignment specified in TS, then fully inserted, and finally restored to the bank position. Compensate for reactivity changes by dilution and boration as required. ii. For the CRA withdrawal, withdraw one or more selected CRAs, one at a time, to the fully withdrawn position. Compensate for reactivity changes by boration and dilution as required. iii. Record incore and excore instrumentation signals to determine their response and to determine the power distribution and power peaking factors prior to control rod assembly misalignment, at partial misalignment, at full misalignment, and periodically after restoration to normal.
Acceptance Criteria
Measured power distributions and power peaking factors are within TS limits and are consistent with the predictions.

Table 14.2-99: Steam Generator Level Control Test # 99

Startup test is required to be performed for each NuScale Power Module.
This test is performed at approximately 25, 50, 75, and 100 percent reactor thermal power.
Test Objective
i. Verify the ability of steam generator (SG) inventory control systems to sustain a ramp increase in load demand. ii. Assess the dynamic response of SG inventory for ramp increase in load demand.
Prerequisite
i. The feedwater system is operating in SG inventory pressure control (feedwater regulating valves in automatic control).
Test Method
i. Raise reactor thermal power to approximately 25 percent. ii. Use the main control room (MCR) turbine controls to provide a 5 percent of full power per minute load increase in demand at approximately 25, 50, and 75 percent reactor thermal power. iii. Use the MCR turbine controls to provide a 5 percent of full power per minute load decrease in demand at approximately 25, 50, and 75, and 100 percent reactor thermal power.
Acceptance Criteria
i. The SG inventory control systems, with no manual intervention, maintain the following parameters within design limits during and following the transient: a. SG superheat b. SG pressure c. SG inventory d. feed pump speed ii. The SG inventory control systems response is reviewed and compared to expected performance. Necessary adjustments to the control systems have been made prior to proceeding to the next power plateau.

Table 14.2-100: Ramp Change in Load Demand Test # 100

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at approximately 25, 50, 75, and 100 percent reactor thermal power.
Test Objectives
i. Verify the ability of the plant automatic control systems to sustain a ramp increase in load demand. ii. Assess the dynamic response of the plant for ramp increase in load demand.
Prerequisites
i. The NPM is operating in a steady-state condition at the designated power level. ii. The plant's electrical distribution system is aligned for normal operation. iii. The following control systems are in automatic control: a. Reactivity control b. Reactor coolant system (RCS) temperature control c. Pressurizer (PZR) pressure control d. PZR level control e. Turbine control f. Feedwater level control g. Circulating water system basin level control h. Site cooling water system basin level control i. Feedwater heater level control j. Hotwell level control iv. If required, verify instrumentation is installed for piping vibration testing.
Test Method
i. Use the main control room (MCR) turbine controls to provide a 5 percent of full power per minute load increase in demand at approximately 25, 50, and 75 percent reactor thermal power. ii. Use the MCR turbine controls to provide a 5 percent of full power per minute load decrease in demand at approximately 25, 50, and 75, and 100 percent reactor thermal power. iii. Conduct piping vibration testing, as required, during power changes.

Table 14.2-100: Ramp Change in Load Demand Test # 100 (Continued)

Acceptance Criteria
i. The turbine does not trip. ii. The reactor does not trip. iii. The main steam safety valves do not open. iv. The turbine does not overspeed. v. The plant automatic control systems, with no manual intervention, maintain the following parameters within design limits during and following the transient: a. Reactor Power b. RCS temperature c. Pressurizer (PZR) pressure d. PZR level e. Steam generator (SG) superheat f. SG pressure g. SG inventory h. Gland seal temperature i. Circulating water system basin level j. Site cooling water system basin level k. Feedwater heater level l. Main condenser hotwell level m. Main condenser vacuum n. Outlet temperature of turbine bypass desuperheater vi. Control system response is reviewed and compared to expected performance. Necessary adjustments to the control systems have been made prior to proceeding to the next power plateau. vii. Water hammer indications a. Audible indications of water hammer are not observed. b. No damage to pipe supports or restraints. c. No damage to equipment. d. No equipment leakage as a result of the ramp change. viii. Piping vibration - System specific steady state vibration testing criteria are established by the piping designer. Actual acceptance criteria will depend on the selected test method, but may include: a. Limits for stresses calculated based on the observed/measured vibration response of the system. b. No permanent deformation or damage is observed in the piping system or supports. c. Vibration displacements are not excessive, would not potentially cause the piping to come in contact with surrounding SSC, and are such that the movement of supports and flexible joints is within their allowable limits.

Table 14.2-101: Step Change in Load Demand Test # 101

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at approximately 25, 50, 75, and 100 percent reactor thermal power.
Test Objectives
i. Verify the ability of the plant automatic control systems to sustain step load increases and step load decreases in demand. ii. Assess the dynamic response of the plant for a load step demand.
Prerequisites
i. The NPM is operating in a steady-state condition at the specified power level. ii. The plant's electrical distribution system is aligned for normal operation. iii. The following control systems are in automatic control: a. Reactivity control b. Reactor coolant system (RCS) temperature control c. Pressurizer (PZR) pressure control d. PZR level control e. Turbine control f. Feedwater level control g. Circulating water system basin level control h. Site cooling water system basin level control i. Feedwater heater level control j. Hotwell level control
Test Method
i. Use the main control room (MCR) turbine controls to provide a 10 percent step load increase in demand at approximately 25, 50, and 75 percent reactor thermal power. ii. Use the MCR turbine controls to provide a 10 percent step load decrease in demand at approximately 25, 50, 75, and 100 percent reactor thermal power.
Acceptance Criteria
i. The turbine does not trip. ii. The reactor does not trip. iii. The main steam safety valves do not open. iv. The turbine does not overspeed. v. The plant automatic control systems, with no manual intervention, maintain the following parameters within design limits during and following the transient: a. Reactor Power b. RCS temperature c. Pressurizer (PZR) pressure d. PZR level e. Steam generator (SG) superheat f. SG pressure g. SG inventory h. Gland seal temperature i. Circulating water system basin level j. Site cooling water system basin level k. Feedwater heater level l. Main condenser hotwell level m. Main condenser vacuum n. Outlet temperature of turbine bypass desuperheater vi. Control system response is reviewed and compared to expected performance. Necessary adjustments to the control systems have been made prior to proceeding to the next power plateau. vii. Water hammer indications a. Audible indications of water hammer are not observed. b. No damage to pipe supports or restraints. c. No damage to equipment. d. No equipment leakage as a result of the step load change.

Table 14.2-102: Loss of Feedwater Heater Test # 102

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at approximately 50 and 90 percent reactor thermal power.
Test Objectives
i. Verify the ability of the plant automatic control systems to sustain a loss of the high pressure feedwater heater during power operation. ii. Assess the dynamic response of the plant for the loss of the high pressure feedwater heater.
Prerequisites
i. The NPM is operating in a steady-state condition at the specified power level. ii. The plant's electrical distribution system is aligned for normal operation. iii. The following control systems are in automatic control: a. Reactivity control b. Reactor coolant system (RCS) temperature control c. Pressurizer (PZR) pressure control d. PZR level control e. Turbine control f. Feedwater level control g. Circulating water system basin level control h. Site cooling water system basin level control i. Feedwater heater level control j. Hotwell level control
Test Method
Close the turbine generator extraction steam supply isolation valve to the high pressure feedwater heater from the main control room at approximately 50 and 90 percent reactor thermal power.
Acceptance Criteria
i. The reactor does not trip. ii. The turbine does not trip. iii. The main steam safety valves do not open. iv. The plant automatic control systems, with no manual intervention, maintain the following parameters within design limits during and following the transient: a. Reactor Power b. RCS temperature c. PZR pressure d. PZR level e. Steam generator (SG) superheat f. SG pressure g. SG inventory h. Gland seal temperature i. Circulating water system basin level j. Site cooling water system basin level k. Feedwater heater level l. Main condenser hotwell level m. Main condenser vacuum n. Outlet temperature of turbine bypass desuperheater

Table 14.2-103: 100 Percent Load Rejection Test # 103

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at approximately 100 percent reactor thermal power.
Test Objectives
i. Verify the ability of the plant automatic control systems to sustain a 100 percent load rejection from full power. ii. Assess the dynamic response of the plant for a 100 percent power load rejection.
Prerequisites
i. The NPM is operating in a steady-state condition at full reactor thermal power. ii. The plant's electrical distribution system is aligned for normal operation. iii. The following control systems are in automatic control: a. Reactivity control b. Reactor coolant system (RCS) temperature control c. Pressurizer (PZR) pressure control d. PZR level control e. Turbine control f. Feedwater level control g. Circulating water system basin level control h. Site cooling water system basin level control i. Feedwater heater level control j. Hotwell level control
Test Method
Manually trip the generator output breaker to provide a 100 percent load rejection.
Acceptance Criteria
i. The turbine trips. ii. The reactor does not trip. iii. The main steam safety valves do not open. iv. The turbine does not overspeed beyond design limits. v. The turbine generator bypass valve opens and modulates steam flow to the condenser to maintain steam generator pressure. vi. The plant automatic control systems, with no manual intervention, maintain the following parameters within design limits during and following the transient: a. Reactor Power b. RCS temperature c. PZR pressure d. PZR level e. Steam generator (SG) superheat f. SG inventory g. Gland seal temperature h. Circulating water system basin level i. Site cooling water system basin level j. Feedwater heater level k. Main condenser hotwell level l. Main condenser vacuum m. Outlet temperature of turbine bypass desuperheater vii. Water hammer indications a. Audible indications of water hammer are not observed. b. No damage to pipe supports or restraints. c. No damage to equipment. d. No equipment leakage as a result of the load rejection.

Table 14.2-104: Reactor Trip from 100 Percent Power Test # 104

Startup test is required to be performed for each NuScale Power Module (NPM).
This test is performed at 100 percent reactor thermal power.
Test Objectives
i. Assess the dynamic response of the plant to a reactor trip. ii. Verify each fully withdrawn control rod assembly (CRA) satisfies the CRA drop time acceptance criteria at full flow conditions. iii. Verify the ability of decay heat removal system (DHRS) to cool the reactor coolant system (RCS) to Mode 3 (all RCS temperatures < 420 °F).
Prerequisites
i. The NPM is operating in a steady-state condition at full reactor thermal power. ii. The plant's electrical distribution system is aligned for normal operation.
Test Method
i. Manually trip the reactor from the main control room. ii. Measure the drop time for each fully withdrawn CRA. iii. Allow the RCS temperature trends to stabilize. iv. Manually initiate DHRS. v. Allow the RCS to cool to mode 3 after DHRS actuation.
Acceptance Criteria
Acceptance criteria to be verified after manual reactor trip: i. The reactor trips. ii. The turbine generator bypass valve operates to prevent opening of the main steam safety valve. iii. The turbine trips. iv. Water hammer indications a. Audible indications of water hammer are not observed. b. No damage to pipe supports or restraints. c. No damage to equipment. d. No equipment leakage as a result of the reactor trip. Acceptance criteria to be verified after DHRS actuation: v. a. DHRS actuation valves open. b. Main steam isolation valves close. c. Feedwater isolation valves close. d. Feedwater regulating valves close e. Secondary main steam isolation valves close f. Secondary main steam bypass isolation valves close g. Pressurizer heater breakers trip vi. The RCS cools to a stable condition in mode 3 (all RCS temperatures < 420 °F) without operator intervention. vii. The RCS cooldown rate is within Technical Specification (TS) limits. viii. Each fully withdrawn CRA drop time is within TS limits.

Table 14.2-105: Island Mode Test for the First NuScale Power Module (Test # 105)

This startup test is required to be performed for the first NuScale Power Module (NPM) in power operation. No other NPMs are in power operation. Test #105 is performed once per facility. Startup Test #106 tests island mode for multiple NPMs.
This test is performed at 100 percent reactor thermal power. Island mode operation is described in Section 8.3.1.1.1
Test Objective for the first NPM in power operation
i. Verify the first NPM in power operation can operate independently from an offsite transmission grid after transition from the transmission grid to island mode. ii. Verify plant electrical loads may be transitioned from island mode to an offsite transmission grid without interruption to the operation of the first NPM in power operation.
Prerequisites
The first NPM in power operation is in normal operation at 100 percent reactor thermal power.
Test Method
Simulate a loss of the transmission grid by opening the switchyard supply breakers (reference Figures 8.3-2a and 8.3-2b).
Acceptance Criteria
i. a. The service unit generator does not trip and changes from droop mode control to isochronous mode to control the loads on site. b. The first NPM in power operation remains at approximately 100 percent reactor thermal power using turbine generator bypass operation. c. Electrical power to plant loads is uninterrupted without loss of voltage or automatic bus transfers. d. The auxiliary alternating current power source starts automatically but does not automatically load its associated bus. ii. The plant electrical loads are transitioned back to the external offsite grid connection when it becomes available.

Table 14.2-106: Island Mode Test for Multiple NuScale Power Modules (Test # 106)

This startup test is required to be performed once with multiple (at least two) NuScale Power Module (NPMs) in operation. Test #106 is performed once per facility. Startup Test #105 tests island mode for a single NPM. **COL Item 14.2-7: A COL applicant that references the NuScale Power Plant design certification will select the plant configuration to perform the Island Mode Test (number of NPMs in service).**
This test is performed at 100 percent reactor thermal power for all NPMs under test. Island mode operation is described in Section 8.3.1.1.1
Test Objective for multiple NPM in operation:
i. Verify all NPMs under test can operate independently from an offsite transmission grid after transition from the transmission grid to island mode. ii. Verify plant electrical loads may be transitioned from island mode to an offsite transmission grid without interruption to the operation of the service unit NPM.
Prerequisites
The NPMs selected for test are in normal operation at 100 percent reactor thermal power.
Test Method
Simulate a loss of the transmission grid by opening the switchyard supply breakers (reference Figures 8.3-2a and 8.3-2b).
Acceptance Criteria
i. a. The service unit turbine generator transitions to island mode by changing from droop mode control to isochronous mode control to control the load on the 13.8kV bus it is supplying. b. The service unit NPM remains at approximately 100 percent reactor thermal power using turbine generator bypass operation. c. The non-service unit turbine generators trip. d. The non-service unit NPMs power reduces to approximately 95 percent reactor thermal power using turbine generator bypass operation. e. Electrical power to plant loads is uninterrupted without loss of voltage or automatic bus transfers. f. The auxiliary alternating current power source starts automatically but does not automatically load its associated bus. ii. The plant electrical loads are successfully transitioned back to an external offsite grid connection when it becomes available.

Table 14.2-107: Remote Shutdown Workstation Test # 107

The remote shutdown station (RSS) is described in Section 7.1.1.2.3. Testing associated with the RSS occurs during the performance of factory acceptance testing (FAT) and site acceptance testing (SAT) as described below.
The RSS provides an alternate location to monitor the NuScale Power Module status and operate the module control system (MCS) and plant control system (PCS) during a main control room (MCR) evacuation. The ability to activate the nonsafety MCS and PCS displays and controls at the RSS will be verified during SAT. The ability to isolate the safety-related MCR module protection system (MPS) manual switches using the MCR isolation switches in the RSS as described in Section 7.2.12 will be verified during MPS FAT and SAT. Refer to Table 14.2-61: Module Control System Test #61 and Table 14.2-62: Plant Control System Test #62 for details regarding MCS and PCS FAT and SAT.

Table 14.2-108: NuScale Power Module Vibration Test # 108

This startup test is required to be performed once for NuScale Power Module (NPM) #1. This test supports first-of-a-kind testing described in Section 14.2.3.3.
This test is performed during the load ramp from zero to 100 percent power and at 100 percent reactor thermal power. The NPM vibration testing is described in Sections 3.9.2.1.1.1, 3.9.2.3, and 3.9.2.4; and "NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report," TR-0918-60894. This test is coordinated with Test #100 and Test #104.
Test Objective for NPM #1
i. Perform vibration testing of containment system (CNTS) Main Steam (MS) line branch connections, including decay heat removal system (DHRS) steam piping, main steam isolation valve (MSIV) bypass lines, and MS drain valve branches during the load ramp up to and at 100 percent reactor thermal power to verify vibration amplitudes in the piping regions confirm there is no acoustic resonance (AR) response. ii. Perform monitoring of vibration amplitudes at locations on the steam generator (SG) assembly, incore instrumentation guide tubes, control rod drive (CRD) shafts, and slip joint connection between the lower and upper riser assemblies. More details regarding the instrumentation locations and vibration mechanisms being monitored are provided in Section 6.0 of TR-0918-60894. Vibration monitoring will be performed during the load ramp up to and at 100 percent reactor thermal power and during a test of DHRS actuation, which will be coordinated with Test #104. iii. Perform visual examination of the NPM components specified in Section 7.0 of TR-0918-60894.
Prerequisites
i. The DHRS steam piping, MSIV bypass lines, and MS drain valve branches are instrumented to obtain AR data. ii. Selected SG tube, incore instrumentation guide tubes, CRD shaft support, and riser slip joint locations are instrumented to provide vibration monitoring.
Test Method
i. Perform load ramp up to 100 percent power, then operate the NPM for a sufficient duration at 100 percent power to ensure one million vibration cycles for the component with the lowest structural natural frequency. ii. Monitor the vibration of the CNTS steam piping branches, including the DHRS steam lines, MSIV bypass lines, and MS drain valve branches. Also monitor the vibration of selected SG tube, incore instrumentation guide tubes, CRD shaft support, and riser slip joint locations for detection of any unexpected large amplitude vibration responses. If an unacceptable vibration response develops any time during initial startup testing, the test conditions will be adjusted to stop the vibration and the reason for the vibration anomaly will be investigated prior to continuing with the testing. iii. Disassemble the NPM and perform a visual examination of the module components specified in Section 7.0 of TR-0918-60894.
Acceptance Criteria
i. Measured vibration amplitudes in the CNTS steam piping branches confirm the acoustic resonance analysis results described in Section 5.2 of TR-0918-60894. ii. Measured vibration responses in the SG tube, incore instrumentation guide tubes, CRD shaft support, and riser slip joint locations are within the magnitudes anticipated due to turbulence only, as discussed in Section 6.0 of TR-0918-60894. iii. Visual examination results of module components satisfy the acceptance criteria of Section 7.0 of TR-0918-60894.

Table 14.2-109: List of Test Abstracts

Test Number	System Abbreviation	Test Abstract
1	SFPCS	Spent Fuel Pool Cooling System
2	PCUS	Pool Cleanup System
3	RPCS	Reactor Pool Cooling System
4	PSCS	Pool Surge Control System
5	UHS	Ultimate Heat Sink
6	PLDS	Pool Leakage Detection System
7	RCCWS	Reactor Component Cooling Water System
8	CHWS	Chilled Water System
9	ABS	Auxiliary Boiler System
10	CWS	Circulating Water System
11	SCWS	Site Cooling Water System
12	PWS	Potable Water System
13	UWS	Utility Water System
14	DWS	Demineralized Water System
15	NDS	Nitrogen Distribution System
16	SAS	Service Air System
17	IAS	Instrument Air System
18	CRHS	Control Room Habitability System
19	CRVS	Normal Control Room HVAC System
20	RBVS	Reactor Building HVAC System
21	RWBVS	Radioactive Waste Building HVAC System
22	TBVS	Turbine Building HVAC System
23	RWDS	Radioactive Waste Drain System
24	BPDS	Balance-of-Plant Drain System
25	FPS	Fire Protection System
26	FDS	Fire Detection System
27	MSS	Main Steam System
28	FWS	Feedwater System
29	FWTS	Feedwater Treatment System
30	CPS	Condensate Polishing System
31	HVDS	Feedwater Heater Vents and Drains System
32	CARS	Condenser Air Removal System
33	TGS	Turbine Generator System
34	TLOSS	Turbine Lube Oil Storage System
35	LRWS	Liquid Radioactive Waste System
36	GRWS	Gaseous Radioactive Waste System
37	SRWS	Solid Radioactive Waste System
38	CVCS	Chemical and Volume Control System
39	BAS	Boron Addition System
40	MHS	Module Heatup System
41	CES	Containment Evacuation System
42	CFDS	Containment Flooding and Drain System
43	CNTS	Containment System
44	N/A	Not Used
45	N/A	Not Used
46	RCS	Reactor Coolant System
47	ECCS	Emergency Core Cooling System
48	DHRS	Decay Heat Removal System
49	ICIS	In-core Instrumentation System
50	MAE	Module Assembly Equipment

Table 14.2-109: List of Test Abstracts (Continued)

Test Number	System Abbreviation	Test Abstract
51	FHE	Fuel Handling Equipment
52	RBC	Reactor Building Crane
53	PSS	Process Sampling System
54	EHVS	13.8 kV and Switchyard System
55	EMVS	Medium Voltage AC Electrical Distribution System
56	ELVS	Low Voltage AC Electrical Distribution System
57	EDSS	Highly Reliable DC Power System
58	EDNS	Normal DC Power System
59	BPSS	Backup Power Supply System
60	PLS	Plant Lighting System
61	MCS	Module Control System
62	PCS	Plant Control System
63	MPS	Module Protection System
64	PPS	Plant Protection System
65	NMS	Neutron Monitoring System
66	SDIS	Safety Display and Indication System
67	RMS	Fixed-Area Radiation Monitoring System
68	COMS	Communication System
69	SMS	Seismic Monitoring System
70	HFT	Hot Functional Testing
71	MAEB	Module Assembly Equipment Bolting
72	SG	Steam Generator Flow-Induced Vibration
73	N/A	Security Access Control
74	N/A	Security Detection and Alarm
75	N/A	Initial Fuel Loading Precritical
76	N/A	Initial Fuel Load
77	N/A	Reactor Coolant System Flow Measurement
78	N/A	NuScale Power Module Temperatures
79	N/A	Primary and Secondary System Chemistry
80	N/A	Control Rod Drive System-Manual Operation, Rod Speed, and Rod Position Indication
81	N/A	Control Rod Assembly Full-Height Drop Time
81A	N/A	Control Rod Assembly Ambient Temperature Full-Height Drop Time Test
82	N/A	Pressurizer Spray Bypass Flow
83	N/A	Initial Criticality
84	N/A	Post-Critical Reactivity Computer Checkout
85	N/A	Low-Power Test Sequence
86	N/A	Determination of Zero-Power Physics Testing Range
87	N/A	All Rods Out Boron Endpoint Determination
88	N/A	Isothermal Temperature Coefficient Measurement
89	N/A	Bank Worth Measurement
90	N/A	Power-Ascension
91	N/A	Core Power Distribution Map
92	N/A	Nuclear Monitoring System Power Range Flux Calibration
93	N/A	Reactor Coolant System Temperature Instrument Calibration
94	N/A	Reactor Coolant System Flow Calibration
95	N/A	Radiation Shield Survey
96	N/A	Reactor Building Ventilation System Capability
97	N/A	Thermal Expansion
98	N/A	Control Rod Assembly Misalignment
99	N/A	Steam Generator Level Control

Table 14.2-109: List of Test Abstracts (Continued)

Test Number	System Abbreviation	Test Abstract
100	N/A	Ramp Change in Load Demand
101	N/A	Step Change in Load Demand
102	N/A	Loss of Feedwater Heater
103	N/A	100 Percent Load Rejection
104	N/A	Reactor Trip from 100 Percent Power
105	N/A	Island Mode Test for the First NuScale Power Module
106	N/A	Island Mode Test for Multiple NuScale Power Modules
107	N/A	Remote Shutdown Workstation
108	N/A	NuScale Power Module Vibration

Table 14.2-110: ITP Testing of New Design Features

New System or Component Design	Design Feature Tested in the Initial Test Program	FSAR Section 14.2 Test Number
Containment isolation valves	• valve leak rate test • valve response to manual engineered safety feature (ESF) action at hot functional test pressure and temperature • valve response time test at hot functional test pressure and temperature • valve response to manual reactor trip at 100% power	#43-1 #63-6 #63-7 #104
Emergency core cooling system (ECCS) valve design	• valve response to manual ESF action at hot functional test pressure and temperature • test of valve inadvertent actuation block at design pressure	#63-6 #63-6
ECCS operation	• Containment response to ECCS operation at hot functional test pressure and temperature.	#47-1
Decay heat removal system (DHRS) valve design	• valve response to manual ESF action at hot functional test pressure and temperature • valve response to manual reactor trip at 100% power	#63-6 #104
DHRS heat exchanger design	• heat exchanger response to manual ESF action at hot functional test pressure and temperature • heat exchanger response to manual reactor trip at 100% power	#48-1 #104
Containment flooding and drain system (CFDS)	• automatic fill of containment • automatic drain of containment	#42
Containment evacuation system	• establish and maintain containment vacuum • provide reactor coolant system (RCS) leakage detection	#41
Containment system level sensors	• provides containment level input for CFDS automatic fill and drain of containment	#42
RCS flow sensors	• provides RCS flow indication during hot functional testing and power ascension testing	#77 #94
Pressurizer (PZR) level sensors	• provides input for PZR level control	#38-1
Island mode operation	• NuScale Power Modules can operate independently from offsite transmission grid.	#105 and #106

14.3 Certified Design Material and Inspections, Tests, Analyses, and Acceptance Criteria

14.3.1 Introduction

This section provides guidance regarding the development of certified design material (CDM) in Tier 1, including Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) required under Title 10 of the Code of Federal Regulations (10 CFR) 52.47(b)(1). The scope of ITAAC is sufficient to provide reasonable assurance that, if the ITAAC are successfully completed, the facility has been constructed and can be operated in accordance with the Atomic Energy Act, relevant Nuclear Regulatory Commission (NRC) regulations, and the combined license (COL). The successful completion of ITAAC constitutes the basis for the NRC determination to allow operation of a facility certified under 10 CFR 52.

Tier 1 information is the portion of the design-related information contained in the Final Safety Analysis Report that is approved and certified by the design certification rule. There are two material categories in Tier 1, the CDM and ITAAC. The CDM is in the form of design descriptions, design commitments, tables, and figures, and is binding for the lifetime of a facility. The ITAAC is used to verify the as-built design features. The ITAAC material expires at initial fuel loading.

The Tier 1 design description consists of the system description and design commitments, both of which contain top-level design features. A design feature is either a physical attribute or a performance characteristic of structures, systems, and components (SSC).

The design features in the system description are not verified by ITAAC. Only the design features in the design commitments are verified by ITAAC.

The sections below describe the criteria and methods by which specific technical entries for Tier 1 were selected. The contents of Tier 1 may not directly correspond to these guidelines in all cases because special considerations may warrant a different approach. In this regard, a case-by-case determination is made consistent with the principles inherent in 10 CFR 52 as well as NRC guidance regarding the content of design descriptions and ITAAC.

COL Item 14.3-1: A COL applicant that references the NuScale Power Plant design certification will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for emergency planning.

COL Item 14.3-2: A COL applicant that references the NuScale Power Plant design certification will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for structures, systems, and components within their scope.

14.3.2 Tier 1 Design Description and Inspections, Tests, Analyses, and Acceptance Criteria First Principles

General criteria that provide clarity on the scope and level of detail of design descriptions and ITAAC are discussed below. These criteria are consolidated and grouped into two sets of first principles: 1) Tier 1 design description scope first principles and 2) ITAAC scope first principles.

The development of the scope of Tier 1 is determined based upon policies set forth by the NRC. A "first principles" approach is considered such that the design descriptions and ITAAC in Design Certification Applications are "necessary and sufficient." Thus, in order to determine the appropriate scope of ITAAC, it is important to apply both the first principles for determining the top-level design features that are included in Tier 1 design descriptions, and the first principles for determining whether a Tier 1 design description needs an ITAAC. Consistent with these first principles, the selection of the top-level design features for Tier 1 is based on the safety significance of SSC, their importance in various safety analyses, and their functions for defense-in-depth considerations.

The first principles for determining the scope of design descriptions and ITAAC are described in Section 14.3.2.1 and Section 14.3.2.2.

14.3.2.1 Tier 1 Design Description Scope First Principles

- Design description content is limited to the following:
  - top-level design features of safety-related SSC
  - top-level design features of safety-related or nonsafety-related SSC that protect safety-related components
  - top-level design features of security system physical SSC
  - top-level design features of risk-significant, nonsafety-related SSC determined by results of a probabilistic risk assessment (PRA)

  Refer to Section 14.3.2.1.1 for further discussion of this principle.
- Design descriptions are derived solely from Tier 2 design information.
- The amount of detail in design descriptions is proportional to the safety significance of the system (i.e., a graded approach). Refer to Section 14.3.2.1.2 for further discussion of the graded approach.
- Not all safety-related design features are included in a design description. Refer to Section 14.3.2.1.3 for further discussion of this principle.
- Not all design features contained in the accident analyses must be included in design descriptions. Refer to Section 14.3.2.1.4 for further discussion of this principle.
- Operational programs and post-fuel load testing are not contained in design descriptions. Refer to Section 14.3.2.1.5 for further discussion of this principle.
- Design descriptions do not need to include every component for that system, but instead only includes those SSC that are required to perform the safety-related and risk-significant system functions. Refer to Section 14.3.2.1.6 for further discussion of this principle.
- Some risk-significant design features identified by the PRA do not need to be specifically addressed in the design description because they are indirectly addressed by design features that are addressed by other design commitments. Refer to Section 14.3.2.1.7 for further discussion of this principle.

- To the extent that an SSC is already the subject of a design commitment by reason of design basis accident (DBA) mitigation function, a design commitment does not need to address the function of the SSC to mitigate severe accidents. Other design features that are not specifically installed for severe accident mitigation, but are used for severe accident mitigation do not need to be addressed in Tier 1. Refer to Section 14.3.2.1.8 for further discussion of this principle.
- Design descriptions only include fixed design features that are installed prior to fuel loading and are expected to be in place for the lifetime of the plant. Refer to Section 14.3.2.1.9 for further discussion of this principle.
- No new design information can be contained in Tier 1 that is not already in Tier 2.
- Tier 1 information is not relied upon for the NRC safety determination provided in a Safety Evaluation Report. The NRC safety determination is based solely on the Tier 2 design information. If a system or component function or design feature is not discussed in Tier 2, hence is not part of the NRC safety determination, it does not belong in Tier 1.
- Design descriptions do not contain a level of detail (e.g., minor dimensional details) that would restrict a licensee from making changes that do not affect a safety-related or risk-significant system function.
- Systems with no safety significance are not included in design descriptions.
- Design descriptions do not contain information that the NRC may designate as "Tier 2*". The NRC guidance in NUREG-0800, Section 14.3, states that Tier 2* information is generally not appropriate for treatment in Tier 1 because it is subject to change.
- Design descriptions do not contain processes that are used for designing and constructing a plant because the safety-related function of an SSC is dependent upon its final as-built condition and not the processes used to achieve that condition.
- Design descriptions do not contain discussions of single failures. Rather, the design description contains related top-level design features, such as physical separation and electrical isolation of Class 1E circuits.

14.3.2.1.1 Tier 1 Design Descriptions Are Limited to the Top-Level Design Features

The following describes the top-level design features for the NuScale Power Plant.

A design feature is either a physical attribute or a performance characteristic of an SSC. The top-level design features contained in Tier 1 design descriptions are:

- reactor coolant pressure boundary
- containment pressure boundary
- Seismic Category I Reactor Building (RXB) and Control Building (CRB)
- Radwaste Category RW-IIa Radioactive Waste Building (RWB)
- control room envelope (CRE)
- safety-related equipment qualification

- safety-related component performance
- SSC providing protection of safety-related components
- safety-related protection system (reactor trip and engineered safety features actuation systems (ESFAS))
- components providing radiation protection for personnel and safety-related equipment
- new and spent fuel storage
- security system physical components

Examples of structures included in Tier 1 design descriptions are the Seismic Category I RXB and CRB, fire barriers, flood barriers, radiation shields, and fuel storage racks.

Examples of components included in Tier 1 design descriptions are valves, instruments, and piping systems.

Examples of physical attributes included in Tier 1 design descriptions are safety-related equipment qualification, location of fire barriers, and thickness of radiation shields.

Examples of performance characteristics included in Tier 1 design descriptions are building seismic performance, safety-related piping conformance to American Society of Mechanical Engineers (ASME) Code Section III requirements, valve stroke time, and safety-related components' automatic response to the module protection system (MPS).

14.3.2.1.2 Graded Approach

The extent to which a particular SSC is described in Tier 1 depends upon the safety significance of the SSC. A graded approach is used to determine the type of information and the level of detail in Tier 1 commensurate with the safety significance of the SSC for the design.

The graded approach reflects the wide variation in safety significance from system to system. It is unnecessary and would be inappropriate to provide the same level of detail for every system in Tier 1.

Top-level design information in Tier 1 is extracted from the more detailed design information presented in Tier 2. Limiting the Tier 1 contents to top-level information reflects the graded approach consistent with NRC guidance in NUREG-0800 and in Regulatory Guide (RG) 1.206.

Severe accident design features are described in the design description, and the ITAAC verify that they exist. In general, the capabilities of the design features need not be included in the ITAAC. For example, a design commitment may discuss that a severe accident containment flooding system exists, while the acceptance criteria would discuss that the severe accident containment flooding system exists, but would not specify the capabilities of associated pumps.

14.3.2.1.3 Not all Safety-Related Design Features are Included in a Design Description

Not all safety-related design features need to be explicitly addressed in design descriptions. Examples of safety-related component design features that generally do not warrant discussion in a design description include:

- instrument lines
- fill lines
- drains
- ASME Code Section III valves that have only a passive function
- piping pressure relief valves associated with thermal expansion and anticipated valve leakage
- interlocks aimed specifically at equipment protection for safety-related components
- local controls for safety-related components
- rebar and concrete properties for Seismic Category I structures

14.3.2.1.4 Top-Level Design Features

Not all design features are included in the design descriptions. Only the top-level design features are contained in the appropriate design description and verified by ITAAC. Table 14.3-1 and Table 14.3-2 present a matrix which correlates the top-level design features contained in design commitments with their treatment in Tier 1. Table 14.3-1 and Table 14.3-2 also contains the top-level design features that were developed based upon results of the following plant safety analyses:

- transient and accident analyses
- internal and external hazards analyses
- radiological analyses
- risk-significant design features as determined by the results of a PRA
- design features necessary or important to severe accident mitigation
- fire protection

By capturing the top level design features that are based upon results of plant safety analyses, the integrity of the fundamental analyses associated with the design as presented in Tier 2 are preserved in the certified design as presented in Tier 1.

14.3.2.1.5 Design Descriptions do not Include Operational Programs and Post-Fuel Load Testing

Those aspects of the design that pertain to programs rather than the as-built plant (e.g., Appendix B to 10 CFR Part 50 requires a quality assurance program, and 10 CFR 50.65 requires a maintenance rule program) are not included in Tier 1.

The key aspects of the design are described in Tier 1. Those aspects of the design that cannot be verified until after fuel loading are not included in ITAAC. This is because 10 CFR 52 requires the ITAAC to be satisfied prior to fuel loading. For these, the Initial Test Program verifies various aspects of the design after fuel load, but prior to operation. Examples are the post-fuel load startup and power ascension test program verification of fuel, control rod, and core characteristics, as well as system and integrated plant operating characteristics. The treatment of these issues is similar to their treatment at facilities licensed under 10 CFR 50, in that verification of the satisfactory completion of these requirements are a condition of the license.

14.3.2.1.6 Design Commitments only include Components Required to Perform System Functions in the System Description

Not every design element specified in the certified design rule has a corresponding Tier 1 verification requirement. For example the safety classification of SSC are identified in the design descriptions, but are not verified by ITAAC because there is no specific test for this characteristic. Further, some ITAAC verify system function and do not address individual system components that together yield the required system functional performance.

14.3.2.1.7 Risk-Significant Design Features as Determined by the Results of a Probabilistic Risk Assessment

Some risk-significant design features identified by the PRA do not need to be specifically addressed in the design description because they are indirectly addressed by design features that are addressed by other design commitments. For example, some PRA studies are dependent upon an assessment of the ability of certain SSC to function during seismic events that are more severe than the design basis safe shutdown earthquake (SSE). If equipment is designed and qualified for the seismic design basis, the design process is such that the added capability assumed in the PRA will inherently be present.

The risk-significant design features that are included in the design descriptions and have associated ITAAC are listed in Table 14.3-1 and Table 14.3-2.

14.3.2.1.8 Design Features Necessary or Important to Severe Accident Mitigation

There are some SSC that mitigate DBAs as well as provide an important success path for severe accident mitigation. The severe accident analysis design features that are included in the design descriptions and have associated ITAAC are listed in Table 14.3-1 and Table 14.3-2.

14.3.2.1.9 Design Descriptions Only Include Fixed Design Features Installed Prior to Fuel Loading and Expected to be in Place for the Lifetime of the Plant

Those aspects of the design that pertain to portable items or consumables rather than fixed design features are not included in Tier 1. Because hardware such as fuel cannot be installed in the reactor until after completion of the ITAAC and because the fuel will be periodically replaced, fuel is not an appropriate topic for ITAAC.

14.3.2.2 Inspections, Tests, Analyses, and Acceptance Criteria Scope First Principles

The following criteria are considered when determining which information warrants inclusion in the ITAAC entries:

- The design commitment is extracted directly from the design descriptions and differences in text are minimized, unless intentional.
- The NRC safety determination is based solely on the Tier 2 design information. ITAAC are not relied upon for the NRC safety determination provided in a Safety Evaluation Report.
- The ITAAC are an important part of the NRC construction verification program, but do not verify every design and construction feature included in the certified design. The ITAAC are not meant to be a one-for-one check of detailed design and construction features that are verified by the normal construction quality programs.
- An inspection, test, or analysis, or a combination thereof, may verify one or more provisions in the design commitment, as defined by the ITAAC.

14.3.3 Organization of Tier 1

The design descriptions, interface requirements, and site parameters are derived from Tier 2 information. Tier 1 information includes

- preamble material which includes a table of contents, a list of tables, and a list of figures.
- an introduction section (described in Section 14.3.4).
- unit-specific design descriptions and ITAAC (described in Section 14.3.5). This section includes
  - systems that are fully within the scope of the NuScale Power Plant design certification.
  - The in-scope portion of those systems that are only partially within the scope of the NuScale Power Plant design certification.
- shared or common SSC and non-SSC design descriptions and ITAAC (described in Section 14.3.6).
- interface material (described in Section 14.3.7).
- site parameters (described in Section 14.3.8).

14.3.3.1 Design Descriptions (Certified Design Material)

The design descriptions serve as requirements for the lifetime of a plant to assure that the plant does not deviate from the certified design. The design descriptions use a system-based structure that is different than the structure of Tier 2. Consequently, developing the design description entries for a system is based on multiple Tier 2 chapters having technical information related to that system.

The design description consists of a system description and design commitments. System description tables and figures are used where appropriate.

The top-level design features in Tier 1 are extracted from the more detailed design information in Tier 2 using the first principles described in Section 14.3.2.1.

System Description and Design Commitments

The purpose of the system description is to provide a concise description of the safety-related and risk-significant system functions, safety classification, and general location. The system description only describes those portions of the system that perform safety-related and risk-significant functions.

The level of detail in system descriptions uses a graded approach commensurate with the safety and risk significance of a system.

Design commitments are provided in numbered paragraphs that are used to develop the design commitment column in the ITAAC table as discussed in Section 14.3.2.2. Design commitments cover design features, such as seismic and ASME Code classifications, Class 1E power sources and divisions, equipment to be qualified for harsh environments (and other than harsh for certain instrumentation and controls (I&C) equipment).

System Description Tables

A table may be used in cases where portions of the system description can be more concisely presented in tabular form. System description tables are generally only referenced in the ITAAC acceptance criteria. System description tables are used to identify design features such as ASME Code class, valve active functions, Class 1E classification of electrical equipment, or required response time of equipment.

System Description Figures

A figure may be included in Tier 1 if it is necessary to describe something that cannot be adequately described in the system description and tables. Figures are provided to convey information in support of system descriptions, in cases where information can be more concisely presented in a figure. Figures are intended to depict a simplified schematic arrangement of the significant SSC.

14.3.3.2 Inspections, Tests, Analyses, and Acceptance Criteria Tables

A table of ITAAC entries is provided for each system that has design commitments in the design description. A three-column format for the ITAAC table is used. All three columns of the ITAAC table must be read and interpreted together.

The first column of the ITAAC table identifies the design commitment to be verified. This column contains the specific text of the design commitment, which is extracted from the design commitments contained in the design description.

The second column of the ITAAC table identifies the proposed method by which the licensee will verify the design commitment described in column 1. The methods used are inspections, tests, analyses, or a combination of the three.

- Inspections are used when verification can be done by visual observation, physical examination, 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.
- Tests mean 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 an ITAAC acceptance criterion is met.

  In addition to testing equipment at its final location, alternative testing methods may be used including factory testing, test facility testing, and laboratory testing. Testing can also include type testing such as might be performed to demonstrate qualification to meet environmental requirements. 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.
- Analyses are used when verification can be done by calculation, mathematical computation, or engineering or technical evaluations.

The third column of the ITAAC table identifies the specific acceptance criteria for the inspections, tests, or analyses described in column 2 that, if met, demonstrate that the licensee has met the design commitments in column 1. Acceptance criteria are objective and clear to avoid confusion over whether or not acceptance criteria have been satisfied.

Using the criteria listed above, ITAAC table entries were developed for each selected system. This was achieved by evaluating the design features defined in the design descriptions and preparing an ITAAC table entry for each design description entry that satisfied the above selection criteria.

Having established the design features for which ITAAC are appropriate, the ITAAC table was completed by selecting the method to be used for verification (either an inspection, a test, or an analysis, or a combination of these) and the acceptance criteria against which the as-built design features are measured. The proposed verification activity is identified in the second column of the ITAAC table.

Where ITAAC is verified by a preoperational test, the test will be established in accordance with the Initial Test Program described in Section 14.2 and RG 1.68. Conversion or extrapolation of test results from the test conditions to design condition may be necessary to satisfy specific ITAAC.

Selection of acceptance criteria is dependent upon the specific design characteristic being verified by the ITAAC table entry. In most cases the appropriate acceptance criteria are self- evident and are based upon the design descriptions. For many of the ITAAC, the acceptance criterion is a statement that the as-built facility has the design feature identified in the design description. A guiding principle for acceptance criteria preparation is the recognition that the criteria should be objective and unambiguous.

In some cases, the ITAAC contain numerical values from Tier 2 that are not specifically identified in the design description or the design commitment column of the ITAAC table. This is acceptable because the design description defines the important design feature that merits Tier 1 treatment. The numerical value in the acceptance criterion is a measurement standard for determining if the as-built facility is in compliance with the design commitment.

The use of objective and unambiguous terms for the acceptance criteria minimizes opportunities for multiple, subjective (and potentially conflicting) interpretations as to whether an acceptance criterion has, or has not, been met. In some cases, the acceptance criteria may be more general because the detailed supporting information in Tier 2 does not lend itself to concise verification. Numerical values for SSC are specified as ITAAC acceptance criteria when values consistent with the design commitments are possible, or when failure to meet the stated acceptance criterion would clearly indicate a failure to properly implement the design or meet the safety analysis.

Where appropriate, the detailed design information provided in Tier 2 includes supporting information for various inspections, tests, and analyses that is used to satisfy the acceptance criteria. This information describes an acceptable means of satisfying an ITAAC.

The details in Tier 2 are not referenced in Tier 1 and are not part of the CDM.

For numerical values in the acceptance criteria, ranges or tolerances are generally included. This is necessary and acceptable because:

- Specification of a single-value acceptance criterion is impractical because minute deviations would represent noncompliance.
- Tolerances recognize that legitimate site variations can occur in complex construction projects.
- Minor variations in plant parameters within the tolerance bounds have no effect on plant safety.

14.3.3.3 Systems Within the Scope of Tier 1

The results of the ITAAC screenings of SSC that are either fully or partially within the scope of the NuScale Power Plant design certification are provided in Table 14.3-1 and Table 14.3-2. These tables identify those SSC that are addressed in Tier 1.

Tier 1 does not include systems that have been determined to not require design descriptions or ITAAC.

14.3.4 Tier 1 Chapter 1, Introduction

Tier 1 Chapter 1 contains the definitions and general provisions used in design descriptions and ITAAC. The intent of these entries is to avoid ambiguities and misinterpretations by providing front-end guidance to users of Tier 1.

Definitions are included for terms used in Tier 1 that could be subject to various interpretations. The intent is to be consistent with Tier 2 information and to reflect NRC guidance regarding various terms. Should questions on terminology arise, the definitions would aid in understanding the intent of the information in Tier 1.

General provisions are included for treatment of individual items, implementation of ITAAC (including ITAAC format), discussion of matters related to operations, and interpretation of figures. The rated reactor core thermal power is not specified because the maximum power level with any special conditions will be specified in the operating license.

Tier 2 Table 1.1-1 is used to interpret Tier 1. The information in Table 1.1-1 will not be duplicated in Tier 1 Chapter 1 in order to prevent the treatment of acronyms and abbreviations as Tier 1 CDM.

The figure legend contained in Tier 2 Chapter 1 is used to interpret Tier 1 system description figures. The information in Figures 1.7-1 through 1.7-3 will not be duplicated in Tier 1 Chapter 1 in order to prevent the treatment of figure legends as Tier 1 CDM.

14.3.5 Tier 1 Chapter 2, Unit-Specific Structures, Systems, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria

Tier 1 Chapter 2 contains design descriptions and associated ITAAC for unit-specific systems that support a single NuScale Power Module (NPM). The unit-specific system design is identical between units. If a unit-specific system meets the first principles for entry into Tier 1 as described in Section 14.3.2.1, then its design description and ITAAC are entered into Tier 1. Tier 1 Chapter 2 includes an entry for each unit-specific system that is either fully or partially within the scope of the NuScale Power Plant design certification as identified in Table 14.3-1.

The design descriptions of a given unit-specific system are the same for all units.

However, unlike single-unit facility designs, each ITAAC for a given unit-specific system must be completed for each unit.

The design description for a unit-specific system will only be recorded once in Tier 1, but the ITAAC for that system must be completed for each unit.

14.3.6 Tier 1 Chapter 3, Shared Structures, Systems, and Components and Non-Structures, Systems, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria

Tier 1 Chapter 3 contains design descriptions and associated ITAAC for systems that support multiple NPMs (shared or common systems). If a shared or common system meets the first principles for entry into Tier 1 as described in Section 14.3.2.1, then its design description and ITAAC are entered into Tier 1. Tier 1 Chapter 3 includes an entry for the shared or common systems that are either fully or partially within the scope of the NuScale Power Plant design certification as identified in Table 14.3-2. Additionally, Tier 1 Chapter 3 addresses non-SSC design and construction activities that are applicable to more than one system or NPM such as human factors engineering.

Shared or common systems that must be completed to support the operation of the first NPM have their ITAAC completed once. If shared or common systems require a portion of the system to be completed to support the operation of the first NPM, then the applicable ITAAC will be in Tier 1 Chapter 2 and must be completed for each associated NPM.

Entries in this chapter of Tier 1 have the same structure as the unit-specific material discussed in Section 14.3.5; that is, design description text, tables, figures, and a table of ITAAC entries.

14.3.7 Tier 1 Chapter 4, Interface Requirements

Tier 1 Chapter 4 provides the interface requirements. Interface requirements are design features that are met by the site-specific portions of a facility that are not within the scope of the certified design. The interface requirements define the design features that ensure the site-specific portion of the design is in conformance with the certified design. The site-specific portions of the design are those portions of the design that are dependent on characteristics of the site.

Tier 1 Chapter 4 also identifies the scope of the design to be certified by specifying the systems that are completely or partially out of scope of the certified design. Thus, interface requirements are defined for: (a) systems that are entirely outside the scope of the certified design, and (b) the out-of-scope portions of those systems that are only partially within the scope of the certified design.

The NuScale Power Plant relies upon passive safety features physically located within NuScale Power Plant buildings and structures. No interfaces need to be identified between or among these portions of the facility. Tier 1 Chapter 4 does not include ITAAC or a requirement for COL developed ITAAC for interface requirements.

14.3.8 Tier 1 Chapter 5, Site Parameters

Tier 1 Chapter 5 provides bounding values for site parameters that a COL applicant referencing the NuScale Power Plant design certification will use in the design of a specific site. Compliance with these site parameters is verified during the COL application process, so no ITAAC are necessary for site parameters. Chapter 2 provides a discussion of the envelope of site parameters used for the NuScale Power Plant design. The corresponding Tier 1 Chapter 5 is based on Table 2.0-1. Tier 1 Chapter 5 is limited to a tabular entry; no supporting text material is required.

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.01	NPM	As required by ASME Code Section III NCA-1210, each ASME Code Class 1, 2 and 3 component (including piping systems) of a nuclear power plant requires a Design Report in accordance with NCA-3550. NCA-3551.1 requires that the drawings used for construction be in agreement with the Design Report before it is certified and be identified and described in the Design Report. It is the responsibility of the N Certificate Holder to furnish a Design Report for each component and support, except as provided in NCA-3551.2 and NCA-3551.3. NCA-3551.1 also requires that the Design Report be certified by a registered professional engineer when it is for Class 1 components and supports, Class CS core support structures, Class MC vessels and supports, Class 2 vessels designed to NC-3200 (NC-3131.1), or Class 2 or Class 3 components designed to Service Loadings greater than Design Loadings. A Class 2 Design Report shall be prepared for Class 1 piping NPS 1 or smaller that is designed in accordance with the rules of Subsection NC. NCA-3554 requires that any modification of any document used for construction, from the corresponding document used for design analysis, shall be reconciled with the Design Report. An ITAAC inspection is performed of the NuScale Power Module ASME Code Class 1, 2 and 3 as-built piping system Design Reports and the ASME Code Class 1, 2, 3, and CS as-built component Design Reports to verify that the requirements of ASME Code Section III are met.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.01 (continued)		The inventory of the NPM ASME Code Class, 1, 2 and 3 piping systems verified by ITAAC 02.01.01 is contained in Tier 1 Table 2.1-1: NuScale Power Module Piping Systems (Table 14.3-3a). The table provides the following information for each NPM piping system: 1) a description of the piping system 2) the system containing the piping system 3) the ASME code class of the piping system 4) the location of the piping system (inside or outside of containment) The inventory of the NPM ASME Code Class, 1, 2, 3, and CS components verified by ITAAC 02.01.01 is contained in Tier 1 Table 2.1-2: NuScale Power Module Mechanical Equipment (Table 14.3-3b). Figure 6.6-1: ASME Class Boundaries for NuScale Power Module Piping Systems provides a graphical representation of the NPM piping systems inventoried in Tier 1 Table 2.1-1 (Table 14.3-3a). Both Figure 6.6-1 and Tier 1 Table 2.1-1 (Table 14.3-3a) indicate that the NPM systems containing ASME Code Class, 1, 2 and 3 piping systems are CNTS, CRDS, SGS, DHRS and RCS. Figure 6.6-1 identifies ASME B31.1 piping systems which are not verified by ITAAC 02.01.01. The ASME B31.1 piping systems are not described in Tier 1 Table 2.1-1 (Table 14.3-3a).					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.02	NPM	The ASME Code Section III requires that documentary evidence be available at the construction or installation site before use or installation to ensure that ASME Code Class 1, 2 and 3 components conform to the requirements of the Code. As defined in NCA-9000, a component can be a vessel, 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 Section III. The NuScale Power Module ASME Code Class 1, 2, and 3 components require a Data Report as specified by NCA-1210. The Data Report is prepared by the certificate holder or owner and signed by the certificate holder or owner and the inspector as specified by NCA-8410. The type of individual Data Report forms necessary to record the required code data is specified in Table NCA-8100-1. An ITAAC inspection is performed of the Data Reports for NuScale Power Module ASME Code Class 1, 2, and 3 as-built components listed in Tier 1 Table 2.1-2 (Table 14.3-3b) and interconnecting piping to (1) ensure that the appropriate Data Reports have been provided as specified in Table NCA-8100-1, (2) ensure that the certificate holder or owner and the authorized nuclear inspector have signed the Data Reports, and (3) verify that the requirements of ASME Code are met.	X			X	

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.03	NPM	The ASME Code Section III requires that documentary evidence be available at the construction or installation site before use or installation to ensure that ASME Code CS components conform to the requirements of the Code. The ASME Code Class CS components require a Data Report as specified by NCA-1210. The Data Report is prepared by the certificate holder or owner and signed by the certificate holder or owner and the Inspector as specified by NCA-8410. The type of individual Data Report Forms necessary to record the required Code Data is identified in Table NCA-8100-1. An ITAAC inspection is performed of the Data Reports for the ASME Code Class CS as-built components listed in Tier 1 Table 2.1-2 (Table 14.3-3b) to (1) ensure that the appropriate Data Reports have been provided as specified in Table NCA-8100-1, (2) ensure that the certificate holder or owner and the inspector have signed the Data Reports, and (3) verify that the requirements of ASME Code Section III are met.	X				
02.01.04	NPM	Section 3.6, Protection against Dynamic Effects Associated with Postulated Rupture of Piping, provides the design bases and criteria for the analysis required to demonstrate that safety-related SSC are not impacted by the adverse effects of a high-and moderate-energy pipe failure within the plant. Table 3.6-2: Postulated Break Locations, lists the high-and moderate-energy pipe break locations. An ITAAC inspection is performed to verify that the as-built protective features credited in the reconciled Pipe Break Hazards Analysis Report such as pipe whip restraints, pipe whip or jet impingement barriers, jet impingement shields, or guard pipe have been installed in accordance with design drawings of sufficient detail to show the existence and location of the protective hardware. The as-built inspection is intended to verify that changes to postulated pipe failure locations and protective features or protected equipment made during construction do not adversely affect the safety-related functions of the protected equipment.	X	X			

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.05	NPM	Section 3.6.3, Leak-Before-Break Evaluation Procedures, describes the application of the mechanistic pipe break criteria, commonly referred to as leak-before-break (LBB), to the evaluation of pipe ruptures. The LBB analysis eliminates the need to consider the dynamic effects of postulated pipe breaks for high-energy piping that qualify for LBB. An analysis, which includes material properties of piping and welds, stress analyses, leakage detection capability, and degradation mechanisms, confirms that the as-designed LBB analysis is bounding for the ASME Code Class 2 as-built piping listed in Tier 1 Table 2.1-1 (Table 14.3-3a) and interconnected equipment nozzles. A summary of the results of the plant specific LBB analysis, including material properties of piping and welds, stress analyses, leakage detection capability, and degradation mechanisms is provided in the as-built LBB analysis report.	X				
02.01.06	NPM	Section 5.3.1.5, Fracture Toughness, discusses the fracture toughness properties of the reactor pressure vessel (RPV) beltline material and the Material Surveillance Program. A Charpy V-Notch test of the RPV beltline material specimen is performed by the vendor to ensure that the initial RPV beltline Charpy upper-shelf energy is 75 ft-lb minimum.	X				
02.01.07	NPM	Section 6.2.6, Containment Leakage Testing, provides a discussion of the leakage testing requirements of the containment vessel (CNV), which serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. As discussed in Section 6.2.6, the NuScale CNV is exempted from the integrated leak rate testing specified in the General Design Criterion (GDC) 52. In accordance with Table 14.2-43, a preoperational test demonstrates that the leakage rate for local leak rate tests (Type B and Type C) for pressure containing or leakage-limiting boundaries and containment isolation valves (CIVs) meet the leakage acceptance criterion of 10 CFR Part 50, Appendix J.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.08	NPM	Section 6.2.4.3, Design Evaluation, provides a discussion of how the containment system (CNTS) containment isolation valves close within the required closure time after receipt of a containment isolation signal to meet containment isolation requirements following a radiological release in the CNV. In accordance with Table 14.2-63, a preoperational test demonstrates that each automatic CIV listed in Tier 1 Table 2.1-3 (Table 14.3-3c) travels from the full open to full closed position in less than or equal to the time listed in Table 6.2-5 after receipt of a containment isolation signal.	X				
02.01.09	NPM	Section 6.2.4.2.2, Component Description, provides a discussion of the isolation valves outside containment that are located as close to the containment as practical in accordance with the requirements of 10 CFR Part 50, Appendix A, GDC 55, 56 and 57. An ITAAC inspection is performed to verify the length of piping between each containment penetration and its associated outboard CIVs is less than or equal to the length identified in Tier 1 Table 2.1-1 (Table 14.3-3a).	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.10	NPM	FSAR Section 8.1.4.3 "Regulatory Requirements and Guidance" outlines the applicable General Design Criteria, NRC Regulations, RGs, and Branch Technical Positions, NUREG Reports, SECY Papers, and NRC Bulletins, and also discusses that the NPM CNTS containment electrical penetration assemblies are sized to power their design loads as demonstrated by satisfying the guidance of RG 1.63. An analysis determines the required design electrical rating needed to power the design loads of each NPM CNTS containment electrical penetration assembly listed in Tier 1 Table 2.1-3 (Table 14.3-3c). An ITAAC inspection is performed to verify that the electrical rating of each NPM CNTS containment electrical penetration assembly listed in Tier 1 Table 2.1-3 (Table 14.3-3c) is greater than or equal to the required design electrical rating. This ITAAC inspection may be performed any time after manufacture of the CNTS containment electrical penetration assemblies.	X				
02.01.11		Not used.					
02.01.12	NPM	Section 5.3.1.6, Material Surveillance, discusses the use of specimen capsules installed in specimen guide baskets. An ITAAC inspection is performed to verify that the correct number of guide baskets are attached to the outer surface of the core barrel at about the mid height of the core support assembly at locations where the capsules will be exposed to a neutron flux consistent with the objectives of the RPV surveillance program.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.13	NPM	The CNTS remotely operated CNTS containment isolation valves are tested by remote operation to demonstrate the capability to perform their function to transfer open and transfer closed under preoperational temperature, differential pressure, and flow conditions. In accordance with Table 14.2-63, a preoperational test demonstrates that the CNTS remotely operated CNTS containment isolation valves listed in Tier 1 Table 2.1-2 (Table 14.3-3b) stroke fully open and fully closed by remote operation under preoperational test conditions. Preoperational test conditions are established that approximate design-basis temperature, differential pressure, and flow conditions to the extent practical, consistent with preoperational test limitations.	X				
02.01.14	NPM	The emergency core cooling system (ECCS) safety-related valves are tested by remote operation to demonstrate the capability to perform their function to transfer open and transfer closed under preoperational temperature, differential pressure, and flow conditions. In accordance with Table 14.2-63, a preoperational test demonstrates that the ECCS safety-related valves listed in Tier 1 Table 2.1-2 (Table 14.3-3b) stroke fully open and fully closed by remote operation under preoperational test conditions. Preoperational test conditions are established that approximate design-basis temperature, differential pressure, and flow conditions to the extent practical, consistent with preoperational test limitations.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.15	NPM	The decay heat removal system (DHRS) safety-related valves are tested by remote operation to demonstrate the capability to perform their function to transfer open and transfer closed under preoperational temperature, differential pressure, and flow conditions. In accordance with Table 14.2-63, a preoperational test demonstrates that the DHRS safety-related valves listed in Tier 1 Table 2.1-2 (Table 14.3-3b) stroke fully open and fully closed by remote operation under preoperational test conditions. Preoperational test conditions are established that approximate design basis temperature, differential pressure, and flow conditions to the extent practical, consistent with preoperational test limitations.	X				
02.01.16		Not used.					
02.01.17		Not used.			X		
02.01.18	NPM	The CNTS safety-related hydraulic-operated valves are tested to demonstrate the capability to perform their function to fail to or maintain their safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions. In accordance with Table 14.2-63, a preoperational test demonstrates that each CNTS safety-related hydraulic-operated valves listed in Tier 1 Table 2.1-2 (Table 14.3-3b) repositions to or maintains its safety-related position on loss of motive power (electric power to the valve actuating solenoid(s) is lost, or hydraulic pressure to the valve(s) is lost). Preoperational test conditions are established that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.19	NPM	The ECCS safety-related reactor recirculation valves and reactor vent valves are tested to demonstrate the capability to perform their function to fail to or maintain their safety-related position on loss of electrical power under preoperational temperature, differential pressure, and flow conditions. In accordance with Table 14.2-63, a preoperational test demonstrates that each ECCS safety-related reactor recirculation valve and reactor vent valve listed in Tier 1 Table 2.1-2 (Table 14.3-3b) fails open on loss of electrical power to its corresponding trip valve. Preoperational test conditions are established that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.	X		X		
02.01.20	NPM	The DHRS safety-related hydraulic-operated valves are tested to demonstrate the capability to perform their function to fail to or maintain their safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions. In accordance with Table 14.2-63, a preoperational test demonstrates that each DHRS safety-related hydraulic-operated valves listed in Tier 1 Table 2.1-2 (Table 14.3-3b) fails open loss of motive power (electric power to the valve actuating solenoid(s) is lost, or hydraulic pressure to the valve(s) is lost). Preoperational test conditions are established that approximate design basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.21	NPM	The CNTS safety-related check valves are tested to demonstrate the capability to perform their function to transfer open and transfer closed (under forward and reverse flow conditions, respectively) under preoperational temperature, differential pressure, and flow conditions. Check valves are tested in accordance with the requirements of the ASME OM Code, ISTC-5220, Check Valves. In accordance with Table 14.2-43, a preoperational test demonstrates that the CNTS check valves listed in Tier 1 Table 2.1-2 (Table 14.3-3b) strokes fully open and closed under forward and reverse flow conditions, respectively. Preoperational test conditions are established that approximate design basis temperature, differential pressure and flow conditions to the extent practicable, consistent with preoperational test limitations.	X				
02.01.22	NPM	The CNTS electrical penetrations listed in Tier 1 Table 2.1-3 may be one of two types, one with or without a circuit interrupting device. An ITAAC confirms that each type of penetration is evaluated to confirm it can withstand its maximum fault current. A circuit interrupting device coordination analysis confirms and concludes in a report that the as-built containment electrical penetration assembly listed in Tier 1 Table 2.1-3 that has a circuit interrupting device can withstand fault currents for the time required to clear the fault from its power source. 8.3.1.2.5 Containment Electrical Penetration Assemblies discusses electrical penetration assemblies that are not equipped with protection devices whose maximum fault current in these circuits would not damage the electrical penetration assembly if that fault current was available indefinitely. An analysis of a CNTS as-built containment penetration without a circuit interrupting device confirms and concludes in a report that the maximum fault current is less than the current carrying capability of the CNTS containment electrical penetration.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[(1)] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.23	NPM	Section 6.2.6.5.2 Preservice Design Pressure Leakage Test provides the test requirements for a preservice design pressure leakage test of the CNV. The test verifies no observed leakage at the CNV bolted flange connections under design pressure. The test may be performed any time after manufacture of the containment vessel, prior to the NPM being placed into service.					
02.01.24	NPM	Section 9.1.5.2.2 discusses that the NPM lifting fixture represent a single-failure-proof lifting device in accordance with the requirements of ANSI N14.6. As described in Section 9.1.5.4, the NPM lifting fixture is load tested to 150% (+5%, -0%) of the manufacturer's rating in accordance with ANSI N14.6. As part of the rated load test, critical areas of the NPM lifting fixture, including all load-bearing welds, will undergo nondestructive testing as required by ANSI N14.6. This ITAAC test may be performed any time after manufacture of the NPM lifting fixture (at the factory or later).				X	
02.01.25	NPM	Section 9.1.5.2.2 discusses that the NPM lifting fixture represents a single-failure-proof lifting device in accordance with the requirements of ANSI N14.6. An ITAAC inspection is performed of the NPM lifting fixture to verify the existence of dual load paths. This ITAAC inspection may be performed any time after manufacture of the NPM lifting fixture (at the factory or later).				X	

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.01.26	NPM	Quality Control inspection hold points are used to ensure the as-built ECCS valves, CIVs, DHRS actuation valves, and their associated hydraulic lines are installed consistent with their associated installation specifications, and therefore capable of performing their safety functions. To demonstrate the acceptance criterion for ITAAC 02.01.26 has been satisfied and the associated design commitment fully met, a report will exist and conclude the following: 1) Quality Control inspection hold points exist and have been completed in accordance with the Quality Assurance Program for each of the following attributes of the ECCS valves, CIVs, and DHRS actuation valves: a. Geometric configuration b. Orientation c. Accessibility 2) Quality Control inspection hold points exist and have been completed in accordance with the Quality Assurance Program for each of the following attributes of routing of the hydraulic lines of the ECCS valves, CIVs, and DHRS actuation valves: a. Twisting b. Bend radii c. Crimping d. Support e. Line separation f. Safe shipment feature removal					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.02.01	CVCS	As required by ASME Code Section III NCA-1210, each ASME Code Class 1, 2 and 3 component (including piping systems) of a nuclear power plant requires a Design Report in accordance with NCA-3550. NCA-3551.1 requires that the drawings used for construction be in agreement with the Design Report before it is certified and be identified and described in the Design Report. It is the responsibility of the N certificate holder to furnish a Design Report for each component and support, except as provided in NCA-3551.2 and NCA-3551.3. NCA-3551.1 also requires that the Design Report be certified by a registered professional engineer when it is for Class 1 components and supports, Class CS core support structures, Class MC vessels and supports, Class 2 vessels designed to NC-3200 (NC-3131.1), or Class 2 or Class 3 components designed to service loadings greater than design loadings. A Class 2 Design Report shall be prepared for Class 1 piping NPS 1 or smaller which is designed in accordance with the rules of Subsection NC. NCA-3554 requires that any modification of any document used for construction, from the corresponding document used for design analysis, shall be reconciled with the Design Report. An ITAAC inspection is performed of the chemical and volume control system (CVCS) ASME Code Class 3 as-built piping system Design Report and the ASME Code Class 3 as-built component Design Reports to verify that the requirements of ASME Code Section III are met. The inventory of the CVCS ASME Code Class 3 piping systems verified by ITAAC 02.02.01 is contained in Tier 1 Table 2.2-1: Chemical and Volume Control Piping (Table 14.3-3d). The inventory of the CVCS ASME Code Class 3 components verified by ITAAC 02.02.01 is contained in Tier 1 Table 2.2-2: Chemical and Volume Control System Mechanical Equipment (Table 14.3-3e). Figure 6.6-1: ASME Class Boundaries for NuScale Power Module Piping Systems provides a graphical representation of the CVCS piping systems inventoried in Tier 1 Table 2.2-1 (Table 14.3-3d).	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.02.02	CVCS	The ASME Code Section III requires that documentary evidence be available at the construction or installation site before use or installation to ensure that ASME Code Class 1, 2 and 3 components conform to the requirements of the Code. As defined in NCA-9000, a component can be a vessel, 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 Section III. The chemical and volume control system ASME Code Class 3 components require a Data Report as specified by NCA-1210. The Data Report is prepared by the certificate holder or owner and signed by the certificate holder or owner and the inspector as specified by NCA-8410. The type of individual Data Report forms necessary to record the required code data is specified in Table NCA-8100-1. An ITAAC inspection is performed of the Data Reports for the chemical and volume control system ASME Code Class 3 as-built components listed in Tier 1 Table 2.2-2 (Table 14.3-3e) and interconnecting piping that is described in Section 9.3.4 to (1) ensure that the appropriate Data Reports have been provided as specified in Table NCA-8100-1, (2) ensure that the certificate holder or owner and the authorized nuclear inspector have signed the Data Reports, and (3) verify that the requirements of ASME Code Section III are met.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.02.03	CVCS	The chemical and volume control ASME Code Class 3 valves are tested by remote operation to demonstrate the capability to perform their function to transfer open and transfer closed under preoperational temperature, differential pressure, and flow conditions. In accordance with the information provided in Table 14.2-38, a preoperational test demonstrates that the chemical and volume control system ASME Code Class 3 valves listed in Tier 1 Table 2.2-2 (Table 14.3-3e) stroke fully open and fully closed by remote operation under preoperational test conditions. Preoperational test conditions are established that approximate design basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.	X				
02.02.04	CVCS	Not used.					
02.02.05	CVCS	The chemical and volume control system ASME Code Class 3 air-operated valves are tested to demonstrate the capability to perform their function to fail to or maintain their position on loss of motive power under preoperational temperature, differential pressure, and flow conditions. In accordance with Table 14.2-38, a preoperational test demonstrates that each chemical and volume control system ASME Code Class 3 air-operated valves listed in Tier 1 Table 2.2-2 (Table 14.3-3e) fails closed on loss of motive power (electric power to the valve actuating solenoid(s) is lost, or pneumatic pressure to the valve(s) is lost). Preoperational test conditions are established that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.03.01	CES	Section 5.2.5 Reactor Coolant Pressure Boundary Leakage Detection, discusses that RCS leakage detection systems are designed to detect and, to the extent practicable, identify the source of reactor coolant leakage. The RCS leakage detection systems conform to the guidance of RG 1.45, regarding detection, monitoring, quantifying, and identification of reactor coolant leakage. In accordance with the information provided in Table 14.2-41, a preoperational test demonstrates that the containment evacuation system (CES) 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. Water vapor and non-condensable gases are removed from the containment vessel by the CES. The water vapor is collected and condensed in the CES sample vessel where it is monitored using level and temperature instrumentation. The CES sample vessel level instrumentation is used to quantify and trend leak rates in the containment.	X				
02.03.02	CES	Section 5.2.5, Reactor Coolant Pressure Boundary Leakage Detection, discusses that RCS leakage detection systems are designed to detect and, to the extent practicable, identify the source of reactor coolant leakage. The RCS leakage detection systems conform to the guidance of RG 1.45, regarding detection, monitoring, quantifying, and identification of reactor coolant leakage. In accordance with Table 14.2-41, a preoperational test demonstrates that the CES is capable of detecting a pressure increase in the CES inlet pressure instrumentation (PIT-1001/PIT-1019), which correlates to a detection of an unidentified RCS leakage rate of one gpm within one hour.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.01	MPS	<u>MPS Design and Implementation</u> Section 7.2.1.1, I&C Safety System Development Process, discusses the software lifecycle phases for the MPS. The purpose is to verify software implementation based on licensing commitments to 10 CFR Part 50, Appendix A, GDC 1 (Quality), Appendix B (Quality Assurance Criteria), RGs 1.28, 1.152, 1.168, 1.169, 1.170, 1.171, 1.172, and 1.173, and the associated IEEE standards. The licensee shall perform analyses for each phase and generate technical reports to conclude that the lifecycle phases were implemented per the licensing commitments. Per RG 1.152, a generic waterfall software life cycle model consists of the following phases: (1) concepts, (2) requirements, (3) design, (4) implementation, (5) test, (6) installation, checkout, and acceptance testing, (7) operation, (8) maintenance, and (9) retirement. The ITAAC verifies that output documentation of each Software Lifecycle phase satisfies the requirements of that phase for the MPS and that software were implemented per licensing commitments to 10 CFR Part 50, Appendix A, GDC1 (Quality), Appendix B (Quality Assurance Criteria), RGs 1.28, 1.152, 1.168, 1.169, 1.170, 1.171, 1.172, and 1.173, and the associated IEEE standards. <u>Tunable Parameters</u> Section 7.2.9, Control of Access, identification, and Repair, discusses the protective measures that prevent modification of the MPS tunable parameters without proper configuration and authorization. Guidance on this issue is provided in DI&C-ISG-04 Revision 1, "Highly-Integrated Control Rooms - Communications Issues," under interdivisional communications, staff position 10. A test demonstrates that protective measures restrict modification to the MPS tunable parameters without proper configuration and authorization. This test will be performed by attempting to modify the tunable parameters with the MPS not in the correct configuration or without authorization.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.01 (continued)		<u>Communication Independence</u> Section 7.1.2, Independence, discusses the communication independence between redundant Class 1E digital communication system divisions. The purpose is to verify proper data isolation between redundant divisions. Requirements for independence are given in IEEE Std. 603-1991. Guidance for providing independence between redundant divisions of the Class 1E digital communication system is provided in Digital I&Cs Interim Staff Guidance (ISG) 04. A test demonstrates that independence between redundant divisions of the Class 1E MPS is provided. <u>Automatic Reactor Trip Signals</u> Section 7.1.1.2.1, Protection Systems, describes automatic and manual reactor trips, variables that are monitored to provide input into automatic reactor trip signals, and the features of the reactor trip system (RTS). The reactor trip functions are listed in Table 7.1-3: Reactor Trip Functions. The reactor trip logic for the monitored variables is provided in Figure 7.1-1. The MPS initiates an automatic reactor trip signal when the associated plant condition(s) exist. A test demonstrates that a reactor trip signal is automatically initiated for each reactor trip function listed in Tier 1 Table 2.5-1. The actuation of reactor trip breakers (RTBs) is not required for this test. The verification of the existence of a reactor trip signal is accomplished using main control room (MCR) displays. <u>Automatic ESF Actuation Signals</u> Section 7.1.1.2.1, Protection Systems, describes automatic and manual engineered safety features (ESFs) actuations, variables that are monitored to provide input into automatic ESFs signals, and the features of the ESF systems. The ESFs functions are listed in Table 7.1-4: Module Protection System Engineered Safeguards Functions. The ESFs logic for the monitored variables is provided in Figure 7.1-1.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.01 (continued)		The MPS initiates an automatic ESF actuation signal when the associated plant condition(s) exist. A test demonstrates that an automatic ESF actuation signal is automatically initiated for each of the ESF functions listed in Tier 1 Table 2.5-2. The actuation of ESFs equipment is not required for this test. The verification of the existence of an ESF actuation signal is accomplished using MCR displays. RTBs Open, Automatic Trip Section 7.1.1.2.1, Protection Systems, describes automatic and manual reactor trips, variables that are monitored to provide input into automatic reactor trip signals, and the features of the RTS. The reactor trip functions are listed in Table 7.1-3: Reactor Trip Functions. The reactor trip logic for the monitored variables is provided in Figure 7.1. The MPS initiates an automatic reactor trip signal for the reactor trip functions when the associated plant condition(s) exist. A test demonstrates that the RTBs open when any one of the automatic reactor trip functions is initiated from the MCR. The RTBs are only opened once to satisfy this test objective. RTBs Open, Manual Trip Section 7.1.1.2.1, Protection Systems, describes automatic and manual reactor trips, variables that are monitored to provide input into automatic reactor trip signals, and the features of the RTS. A manual reactor trip is one of the MPS manually actuated functions. A test demonstrates that the RTBs open when a reactor trip is manually initiated from the MCR.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.01 (continued)		Reactor Trip Logic Section 7.1.6, Safety Evaluation, describes the MPS conformance to the GDC in 10 CFR 50 Appendix A. Guidance provided in Design Specific Review Standard Section 7.2.3, Reliability, Integrity, and Completion of Protective Action, states that the design incorporate protective measures that provide for I&C safety systems to fail in a safe state, or into a state that has been demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of power, or adverse environments, are experienced. Section 7.1.6 describes that consistent with GDC 23, the MPS is designed, with sufficient functional diversity as to prevent the loss of a protection function, to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of power, or postulated adverse environments are experienced. Section 7.2.3.2, System Integrity Characteristics, states that the MPS is designed such that in the event of a condition such as a system disconnection or loss of power the MPS fails into a safe state. A test demonstrates that when the loss of electrical power is detected in a separation group of the MPS that separation group fails to a safe state resulting in a reactor trip state for that separation group. ESF Trip Logic Section 7.1.6, Safety Evaluation, describes the MPS conformance to the GDC in 10 CFR 50 Appendix A. Guidance provided in Design Specific Review Standard Section 7.2.3, Reliability, Integrity, and Completion of Protective Action, states that the design incorporate protective measures that provide for I&C safety systems to fail in a safe state, or into a state that has been demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of power, or adverse environments, are experienced.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.01 (continued)		Section 7.1.6 describes that consistent with GDC 23, the MPS is designed, with sufficient functional diversity as to prevent the loss of a protection function, to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of power, or postulated adverse environments are experienced. Section 7.2.3.2, System Integrity Characteristics, states that the MPS is designed such that in the event of a condition such as a system disconnection or loss of power the MPS fails into a safe state. For an ESF function this predefined safe state may be that the actuated component remains as-is. A test demonstrates that when the loss of electrical power is detected in a separation group of the MPS that separation group fails to a safe state for that separation group. MPS Interlocks Section 7.2.4.1, Operating Bypasses, describes MPS operating bypasses for reactor trip functions. Section 7.2.4.1, Operating Bypasses, describes MPS operating bypasses for ESF actuations. The operating bypasses are applied automatically when plant conditions dictate that the safety function is not needed, or that the safety function prevents proper plant operation at a specific mode of operation. A test demonstrates that the MPS interlocks listed in Tier 1 Table 2.5-4 automatically establish an operating bypass for the specified reactor trip or ESF actuations when a real or simulated signal simulates that the associated interlock condition is met; and are automatically removed when the real or simulated signal simulates that the associated permissive condition is no longer satisfied.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.01 (continued)		<u>MPS Permissives</u> Section 7.2.4.1, Operating Bypasses, describes MPS operating bypasses for reactor trip functions. Section 7.2.4.1, Operating Bypasses, describes MPS operating bypasses for ESF actuations. The operating bypasses are applied automatically when plant conditions dictate that the safety function is not needed, or that the safety function prevents proper plant operation at a specific mode of operation. A test demonstrates that the MPS permissives listed in Tier 1 Table 2.5-4 allows the manual bypass of the specified reactor trip or ESF actuations when a real or simulated signal simulates that the associated permissive condition is met; and are automatically removed when the real or simulated signal simulates that the associated permissive condition is no longer satisfied. <u>MPS Overrides</u> Section 7.2.4.1, Operating Bypasses, describes MPS operating bypasses for reactor trip functions. Section 7.2.4.1, Operating Bypasses, describes MPS operating bypasses for ESF actuations. The operating bypasses are applied automatically when plant conditions dictate that the safety function is not needed, or that the safety function prevents proper plant operation at a specific mode of operation. A test demonstrates that the MPS overrides listed in Tier 1 Table 2.5-4 are established when the manual override switch is active and a real or simulated RT-1 interlock is established.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.01 (continued)		Maintenance Bypass Section 7.2.4.2, Maintenance Bypass, describes the MPS maintenance bypass operation mode. An individual protection channel can be placed in a maintenance bypass operation mode to allow manual testing and maintenance during power operation, while ensuring that the minimum redundancy required by the Technical Specifications is maintained. The reactor trip functions are listed in Table 7.1-3: Reactor Trip Functions. The ESFs functions are listed in Table 7.1-4: Module Protection System Engineered Safeguards Functions. A test demonstrates that with a safety function module out of service switch activated, the safety function is placed in trip or bypass based on the position of the safety function module trip/bypass switch. Each separation group of the reactor trip functions listed in Tier 1 Table 2.5-1 and each separation group of the ESFs signals listed in Tier 1 Table 2.5-2 is tested by placing the separation group in maintenance bypass. RTB Arrangement Section 7.0.4.1.2, Reactor Trip System, discusses the arrangement of the protection system RTBs. Figure 7.0-6: Reactor Trip Breaker Arrangement provides the arrangement of the RTBs. This ITAAC verifies that the RTBs conform to the arrangement indicated in Tier 1 Figure 2.5-1. In addition, the ITAAC inspection verifies proper connection of the shunt and undervoltage trip mechanisms and other auxiliary contacts. Different Programmable Technology Section 7.1.5.1, Application of NUREG/CR-6303 Guidelines, discusses that two of the four separation groups and one of the two divisions of RTS and ESFAS will utilize a different programmable technology.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.01 (continued)		A ITAAC inspection is performed to verify that MPS 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.					
02.05.02		Not used.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.03	MPS	Sections 7.1.2, Independence, discusses the independence of the MPS Class 1E I&C current-carrying circuits per the guidance of RG 1.75, which endorses IEEE Std. 384-1992. Physical separation is provided to maintain the independence of Class 1E I&C current-carrying circuits so that the safety functions required during and following any design basis event can be accomplished. Minimum separation distance (as defined in IEEE Std. 384-1992), or barriers or any combination thereof may achieve physical separation as specified in IEEE Std. 384-1992. Separate ITAAC inspections are performed to verify the independence provided by physical separation and the independence provided by electrical isolation. This ITAAC verifies the independence of Class 1E current-carrying circuits by physical separation. An ITAAC inspection is performed of physical separation of the MPS Class 1E current-carrying circuits. The physical separation ITAAC inspection results verify that the following physical separation criteria are met: i. Physical separation between redundant divisions of the MPS Class 1E I&C 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; and such physical separation satisfies the criteria of RG 1.75. The configuration of each as-built barrier agrees with its associated as-built drawing. ii. Physical separation between the MPS Class 1E I&C current-carrying circuits and non-Class 1E I&C 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; and such physical separation satisfies the criteria of RG 1.75. The configuration of each as-built barrier agrees with its associated as-built drawing.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.04	MPS	Sections 7.1.2, Independence, discusses the independence of the MPS Class 1E I&C circuits per the criteria of RG 1.75, which endorses IEEE Std. 384-1992. Electrical isolation is provided between the redundant divisions of the MPS Class 1E I&C circuits, and between Class 1E I&C circuits and non-Class 1E I&C circuits by Class 1E isolation devices so a failure in an I&C circuit does not prevent safety-related function completion in a different Class 1E I&C circuit. An ITAAC inspection is performed to verify the following electrical isolation criteria are met: i. Class 1E electrical isolation devices that satisfy the criteria of RG 1.75 are installed between redundant divisions of the MPS Class 1E I&C circuits. ii. Class 1E electrical isolation devices that satisfy the criteria of RG 1.75 are installed between the MPS Class 1E I&C circuits and non-Class 1E I&C circuits.	X				
02.05.05	MPS	Sections 7.1.2, Independence, discusses the independence of MPS Class 1E circuits per the criteria of RG 1.75, which endorses IEEE Std. 384-1992. Electrical isolation is provided between Class 1E circuits and non-Class 1E circuits by Class 1E isolation devices so a failure in a non-Class 1E circuit does not prevent the safety-related function completion in the Class 1E circuit. i. The ITAAC verifies that: (1) an equipment qualification data report exists for the Class 1E isolation devices, and (2) the equipment qualification data report concludes that the Class 1E isolation devices performs its safety-related function under the design basis environmental conditions specified in the equipment qualification data report. ii. An ITAAC inspection is performed to verify that Class 1E electrical isolation devices are installed between MPS Class 1E circuits and non-Class 1E circuits, which satisfy the guidance of RG 1.75.	X				
02.05.06		Not used.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.07	MPS	Section 7.1.2, Independence, discusses the communication independence between Class 1E digital communication systems and non-Class 1E digital communication systems. The purpose is to verify that logical or software malfunction of the nonsafety-related system cannot affect the functions of the safety system. Requirements for independence are given in IEEE Std. 603-1991. Guidance for providing independence between the Class 1E digital communication system and non-Class 1E digital communication systems is provided in Digital Instrumentation and Controls ISG 04. A vendor test demonstrates that independence between the Class 1E MPS and non-Class 1E digital systems is provided.	X				
02.05.08		Not used.					
02.05.09		Not used.					
02.05.10		Not used.					
02.05.11	MPS	Section 7.1.1.2.1, Protection Systems, describes automatic and manual ESFs actuations, variables that are monitored to provide input into automatic ESFs signals, and the features of the engineered safety feature systems. The ESFs functions are listed in Table 7.1-4: Module Protection System Engineered Safeguards Functions. The ESFs logic for the monitored variables is provided in Figure 7.1. The MPS initiates an automatic ESF actuation signal for the functions listed in Tier 1 Table 2.5-2 when the associated plant condition(s) exist. In accordance with Table 14.2-63, a preoperational test demonstrates that ESF equipment automatically actuates to perform its safety-related function listed in Tier 1 Table 2.5-2 upon an injection of a single simulated MPS signal.	X				
02.05.12		Not used.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.13	MPS	Section 7.1.1.2.1, Protection Systems, describes manual ESFs actuation, variables that are monitored to provide input into automatic ESFs signals, and the features of the ESF system. The ESFs functions that can be manually actuated are shown in Figure 7.1-1 In accordance with Table 14.2-63, a preoperational test demonstrates that the MPS actuates the ESF equipment to perform its safety-related function listed in Tier 1 Table 2.5-2 when manually initiated.	X				
02.05.14		Not used.					
02.05.15		Not used.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.16	MPS	Section 7.2.3.3, Completion of Protective Action, describes compliance with requirements for completion of protective actions, which requires that, once initiated, the reactor trip and ESF proceed to completion and remain in their required position/condition until the actuation system is reset and operator action is taken. IEEE 603-1991 Clause 5.2 states that `The safety systems shall be designed so that, once initiated automatically or manually, the intended sequence of protective actions of the execute features shall continue until completion. Deliberate operation action shall be required to return the safety systems to normal. This requirement shall not preclude the use of equipment protective devices identified in [Clause] 4.11 of the design basis or the provisions for deliberate operator interventions. Seal-in of individual channels is not required." In accordance with Table 14.2-63, a preoperational test demonstrates that: i. upon an MPS reactor trip signal listed in Tier 1 Table 2.5-1, the RTBs open and the RTBs do not automatically close when the MPS reactor trip signal clears. ii. upon an MPS engineered safety feature actuation signal listed in Tier 1 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.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.17	MPS	Section 7.2.12.1, Automatic Control, describes the signals and initiating logic for each reactor trip and required response times. In accordance with Table 14.2-63, a preoperational test demonstrates that the measured time for the reactor trip functions listed in Tier 1 Table 2.5-1 is less than or equal to the maximum values assumed in the accident analysis. Section 7.2.12.1, Automatic Control, describes the signals and initiating logic for each ESF and the required response times. In accordance with Table 14.2-63, a preoperational test demonstrates that the measured time for the ESF functions listed in Tier 1 Table 2.5-2 is less than or equal to the maximum values assumed in the accident analysis. Technical specification SR 3.0.1 bases states that surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire surveillance is performed within the specified frequency. The technical specification bases also describe an allowance for response time to be verified by any series of sequential, overlapping, or total channel measurements.	X				
02.05.18		Not used.					
02.05.19		Not used.					
02.05.20		Not used.					
02.05.21		Not used.					
02.05.22	MPS	Section 7.2.4.2, Maintenance Bypass, describes the MPS maintenance bypass operation mode. An individual protection channel can be placed in a maintenance bypass operation mode to allow manual testing and maintenance during power operation, while ensuring that the minimum redundancy required by the technical specifications is maintained. Section 7.2.4.2 discusses the status indication of MPS manual or automatic bypasses placed in maintenance bypass operation mode. In accordance with Table 14.2-63, a preoperational test demonstrates that each operational MPS manual or automatic bypass is indicated in the MCR.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.23	MPS	Section 7.2.4.2, Maintenance Bypass, describes the MPS maintenance bypass operation mode. An individual protection channel can be placed in a maintenance bypass operation mode to allow manual testing and maintenance during power operation, while ensuring that the minimum redundancy required by the technical specifications is maintained. Section 7.2.4.2 discusses the status indication of MPS maintenance bypasses placed in maintenance bypass operation mode. In accordance with Table 14.2-63, a preoperational test demonstrates that each MPS maintenance bypass is indicated in the MCR.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[(1)] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.24	MPS	This ITAAC is intended to address self-testing features credited towards surveillance or other operational testing. Given the nature of this ITAAC, it is acceptable to verify ITAAC completion during the factory acceptance testing (FAT). Self-testing features include, but are not limited to, watchdog timers, automated channel checks, and signal input comparisons. Section 7.2.15.3, Fault Detection and Self-diagnostics, discusses the self-testing features of the MPS, including the types of faults that should be detected, the system responses to such faults, the required response times, and the ability for alarms and displays in the MCR to provide indication of such faults' existence. These tests of the MPS self-testing features ensure that a) faults requiring detection are detected, b) the system responds appropriately to each fault based on the type of fault, c) the response occurs within a sufficient timeframe to ensure safety function is not lost, and d) that alarms and indications in the main control room indicate the type of fault present. A vendor test demonstrates and 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. • self-testing features verify that detected faults are indicated by alarms and displays.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.05.25	MPS	Section 7.1.1.2.2, Post Accident Monitoring, and Section 7.2.13, Displays and Monitoring, describe the post-accident monitoring (PAM) Type B and C displays and alarms indicated on the safety display and indication system (SDIS) displays in the MCR. PAM Type B and C variables are developed in accordance with the guidance in RG 1.97, Revision 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" which endorses (with certain clarifying regulatory positions specified in Section C of this guide) IEEE Std. 497-2002.1, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." In accordance with Table 14.2-66, a preoperational test demonstrates the ability to retrieve and display the various PAM Type B and C parameters and alarms at the as-built safety display indication displays in the main control room. The intent is to verify that the displays and alarms function during testing of the integrated as-built system; however, separate testing of the actual operation of the PAM alarms and displays using simulated signals may be acceptable where this is not practical.	X				
02.05.26	MPS	Section 18.6, Treatment of Important Human Actions, provides a summary of the treatment of important human actions (TIHA) objectives, scope, methodology, and results. The TIHA methodology and the results are documented in the TIHA results summary report. The TIHA approach is consistent with the applicable provisions of NUREG-0711, Revision 3. In accordance with Table 14.2-63, a preoperational test demonstrates that the minimum inventory of controls identified by the human factors engineering process can be manually operated from the operator workstation in the MCR. The IHA functions verified are contained in Tier 1 Table 2.5-6 (Table 14.3-3f).	X				
02.05.27		Not used.					
02.05.28		Not used.					

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.06.01	NMS	Section 7.1.2.2, Electrical Independence, discusses the independence of the neutron monitoring system (NMS) Class 1E circuits. Electrical isolation is provided between Class 1E circuits and non-Class 1E circuits by Class 1E isolation devices so a failure in a non-Class 1E circuit does not prevent the safety-related function completion in the Class 1E circuit. A type test, analysis, or a combination of type test and analysis will be performed of the Class 1E isolation devices to verify that 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. An ITAAC inspection is performed to verify that Class 1E electrical isolation devices are installed between NMS Class 1E circuits and non-Class 1E circuits.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.06.02	NMS	Sections 7.0.3.2, Safety-Related Systems, 7.1.2.1, Physical Independence, and 7.1.2.4.1, Independence between Redundant Portions of a Safety System, discuss the independence of the NMS Class 1E I&C current-carrying circuits per the guidance of RG 1.75, which endorses IEEE Std. 384-1992. Physical separation is provided to maintain the independence of Class 1E I&C current-carrying circuits so that the safety functions required during and following any design basis event can be accomplished. Minimum separation distance (as defined in IEEE Std. 384-1992), or barriers or any combination thereof may achieve physical separation as specified in IEEE Std. 384-1992. Separate ITAAC inspections are performed to verify the independence provided by physical separation and the independence provided by electrical isolation. This ITAAC verifies the independence of Class 1E current-carrying circuits by physical separation. An ITAAC inspection is performed of physical separation of the NMS Class 1E current-carrying circuits. The physical separation ITAAC inspection results verify that the following physical separation criteria are met: i. Physical separation between redundant divisions of the NMS Class 1E I&C 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; and such physical separation satisfies the criteria of RG 1.75. The configuration of each as-built barrier agrees with its associated as-built drawing. ii. Physical separation between the NMS Class 1E I&C current-carrying circuits and non-Class 1E I&C 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. The configuration of each as-built barrier agrees with its associated as-built drawing.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.06.03	NMS	Sections 7.0.3.2, Safety-Related Systems, 7.1.2.2, Electrical Independence, and 7.1.2.4.1, Independence between Redundant Portions of a Safety System, discuss the independence of the NMS Class 1E I&C circuits per the criteria of RG, which endorses IEEE Std. 384-1992. Electrical isolation is provided between the redundant divisions of the NMS Class 1E I&C circuits, and between Class 1E I&C circuits and non-Class 1E I&C circuits by Class 1E isolation devices so a failure in an I&C circuit does not prevent safety-related function completion in a different Class 1E I&C circuit. An ITAAC inspection is performed to verify the following electrical isolation criteria are met: i. Class 1E electrical isolation devices that satisfy the criteria of RG 1.75 are installed between redundant divisions of the NM system Class 1E I&C circuits. ii. Class 1E electrical isolation devices that satisfy the criteria of RG 1.75 are installed between the NMS Class 1E I&C circuits and non-Class 1E I&C circuits.	X				
02.07.01		Section 11.5.2.2.7, Containment Evacuation System, discusses the operation of the CES. For each high radiation signal listed in Tier 1 Table 2.7-1(Table 14.3-3g), the CES automatically aligns the components identified in Tier 1 Table 2.7-1 (Table 14.3-3g) to the required positions identified in the table. In accordance with Table 14.2-41, a preoperational test demonstrates the CES automatically aligns the components identified in Tier 1 Table 2.7-1 (Table 14.3-3g) to the required positions identified in the table upon initiation of a real or simulated CES high radiation signal from CES-RT-1011.			X		

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.07.02		Section 11.5.2.2.11, Chemical and Volume Control System, discusses the operation of the CVCS. For each high radiation signal listed in Tier 1 Table 2.7-1 (Table 14.3-3g), the CVCS automatically aligns the components identified in Tier 1 Table 2.7-1 (Table 14.3-3g) to the required positions identified in the table. In accordance with Table 14.2-38, a preoperational test demonstrates the CVCS and ABS automatically aligns the components identified in Tier 1 Table 2.7-1 (Table 14.3-3g) to the required positions identified in the table upon initiation of a real or simulated CVCS high radiation signal from CVC-RT-1004, 0A-AB-RIT-1005, and 0B-AB-RIT-1005.			X		

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.08.01	EQ	Section 3.10, Seismic and Dynamic Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the Seismic Category I equipment, including its associated supports and anchorages, is qualified by type test, analysis, or a combination of type test and analysis to perform its function under the design basis seismic loads during and after an SSE. The qualification method employed for the Seismic Category I equipment is the same as the qualification method described for that type of equipment in Section 3.10. This method conforms to IEEE-344-2004 and ASME QME-1-2007 (or later editions), as accepted by the NRC staff in RG 1.100 Revision 3 (or later revision), with specific revision years and numbers as presented in Section 3.10. The scope of equipment for this design commitment is module-specific, safety-related equipment, and module-specific, nonsafety-related equipment that 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 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.) The ITAAC verifies that: (1) a Seismic Qualification Report exists for each Seismic Category I component type, and (2) the Seismic Qualification Report concludes that the Seismic Category I equipment listed in Tier 1 Table 2.8-1 (Table 14.3-3h), including its associated supports and anchorages, performs its function under the seismic design basis load conditions specified in the Seismic Qualification Report. After installation in the plant, an ITAAC inspection is performed to verify that the Seismic Category I equipment listed in Tier 1 Table 2.8-1 (Table 14.3-3h), including its associated supports and anchorages, is installed in its design location in a Seismic Category I structure in a configuration bounded by the Seismic Qualification Report.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.08.02	EQ	Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the electrical equipment, including its connection assemblies, located in a harsh environment is qualified by type test or a combination of type test and analysis to perform its function under design basis harsh environmental conditions, experienced during normal operations, anticipated operational occurrences, DBAs, and post-accident conditions in accordance with 10 CFR 50.49. As defined in IEEE-Std-572-2006, IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations, a connection assembly is any connector or termination combined with related cables or wires as an assembly. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in Section 3.11. The scope of equipment for this design commitment is module-specific, Class 1E equipment located within a harsh environment, and module-specific, nonsafety-related equipment with an augmented equipment qualification design requirement located within the boundaries of the NuScale Power Module. The ITAAC verifies that: (1) an equipment qualification record form exists for the electrical equipment listed in Tier 1 Table 2.8-1 (Table 14.3-3h) and addresses connection assemblies, (2) the equipment qualification record form concludes that the electrical equipment, including its connection assemblies, performs its function under the environmental conditions specified in Section 3.11 and the equipment qualification record form, and (3) the required post-accident operability time for the electrical equipment in the equipment qualification record form is in agreement with Section 3.11. After installation in the plant, an ITAAC inspection is performed to verify that the electrical equipment listed in Tier 1 Table 2.8-1 (Table 14.3-3h), including its connection assemblies, is installed in its design location in a configuration bounded by the equipment qualification record form.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.08.03	EQ	Section 3.11 presents information to demonstrate that the non-metallic parts, materials, and lubricants used in mechanical equipment located in a harsh environment are qualified using a type test or a combination of type test and analysis to perform their function up to the end of their qualified life in design basis harsh environmental conditions experienced during normal operations, anticipated operational occurrences, DBAs, and post-accident conditions. Environmental conditions include both internal service conditions and external environmental conditions for the nonmetallic parts, materials, and lubricant. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in Section 3.11. The scope of equipment for this design commitment is module-specific, safety-related mechanical equipment, and module-specific, nonsafety-related mechanical 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). The ITAAC verifies that: (1) an equipment qualification record form or ASME QME-1 report exists for the non-metallic parts, materials, and lubricants used in mechanical equipment designated for a harsh environment, and (2) the qualification record form concludes that the non-metallic parts, materials, and lubricants used in mechanical equipment listed in Tier 1 Table 2.8-1 perform their intended function up to the end of its qualified life under the design basis environmental conditions (both internal service conditions and external environmental conditions) specified in the qualification record form.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[(1)] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.08.04	EQ	Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment, and Appendix 3C, Methodology for Environmental Qualification of Electrical and Mechanical Equipment, presents information to demonstrate that the Class 1E computer-based I&C systems located in a mild environment is qualified by type test or a combination of type test and analysis to perform its safety-related function under the design basis mild environmental conditions. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in Section 3.11 and Appendix 3C. This method conforms to IEEE-323-2003 (or later editions), as accepted by the NRC staff in RG 1.209, Revision 0 (or later revision), with specific revision years and numbers as presented in Section 3.10. The ITAAC verifies that: (1) an equipment qualification record form exists for the Class 1E computer-based I&C systems listed in Tier 1 Table 2.8-1 (Table 14.3-3h), and (2) the equipment qualification record form concludes that the Class 1E computer-based I&C systems performs its safety-related function under the design basis mild environmental conditions specified in Section 3.11 and Appendix 3C and the equipment qualification record form. After installation in the plant, an ITAAC inspection is performed to verify that the Class 1E computer-based I&C systems listed in Tier 1 Table 2.8-1 (Table 14.3-3h) is installed in its design location in a configuration bounded by its equipment qualification record form.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[(1)] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.08.05	EQ	Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the Class 1E digital equipment is qualified using a type test, analysis, or a combination of type test and analysis to perform its safety-related function when subjected to electromagnetic interference, radio frequency interference, and electrical surges that would exist before, during, and following a DBA. The qualification method employed for Class 1E digital equipment is the same as the qualification method described for that type of equipment in Section 3.11. The ITAAC verifies that: (1) an equipment qualification record form exists for the Class 1E digital equipment listed in Tier 1 Table 2.8-1, and (2) the equipment qualification record form concludes that the Class 1E digital equipment 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.	X				
02.08.06	EQ	Section 3.9.6.1, Functional Design and Qualification of Pumps, Valves, and Dynamic Restraints, and Section 3.10.2, Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation, discuss that the functional qualification of safety-related valves is performed in accordance with ASME QME-1-2007 (or later edition), as accepted in RG 1.100 Revision 3 (or later revision), with specific revision years and numbers as presented in Section 3.9.6.1. The qualification method employed for the valves agrees with the qualification method described in Section 3.10.2. The ITAAC verifies that: (1) a Qualification Report exists for the safety-related valves listed in Tier 1 Table 2.8-1, and (2) the Qualification Report concludes that safety-related valves are capable of performing their safety-related function under the full range of fluid flow, differential pressure, electrical conditions, temperature conditions, and fluid conditions up to and including DBA conditions.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.08.07	EQ	Section 3.9.3.2, Design and Installation of Pressure Relief Devices, discusses that relief valves provide overpressure protection in accordance with the ASME Code Section III. The ITAAC verifies that: (1) the test for each relief valve listed in Tier 1 Table 2.8-1 (Table 14.3-3h) meets the set pressure, capacity, and overpressure design requirements; and (2) each relief valve listed in Tier 1 Table 2.8-1 (Table 14.3-3h) is provided with an ASME Code Certification Mark that identifies the valve's set pressure, capacity, and overpressure.	X				
02.08.08	EQ	Section 5.4.3, Decay Heat Removal System, discusses that the DHRS passive condensers provide the safety-related function of transferring their design heat load from the DHRS during shutdown. After manufacture of the DHRS passive condensers, a type test or a combination of type test and analysis is performed to validate that the DHRS passive condensers are capable of meeting the specified heat transfer performance requirements. Section 5.4.3 discusses the design heat transfer capability of the DHRS passive condensers. The ITAAC verifies that the safety-related passive condensers listed in Tier 1 Table 2.8-1 have a heat removal capacity sufficient to transfer their design heat load.	X				

Table 14.3-1: Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[(1)] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
02.08.09	EQ	Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the CNTS electrical penetration assemblies, including its connection assemblies, located in a harsh environment are qualified by type test or a combination of type test and analysis to perform its safety-related function under design basis harsh environmental conditions, experienced during normal operations, anticipated operational occurrences, DBAs, and post-accident conditions in accordance with 10 CFR 50.49. As defined in IEEE-Std-572-2006, IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations, a connection assembly is any connector or termination combined with related cables or wires as an assembly. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in Section 3.11. The ITAAC verifies that: (1) an equipment qualification record form exists for the CNTS electrical penetration assemblies listed in Tier 1 Table 2.8-1 (Table 14.3-3h) and addresses connection assemblies; (2) the equipment qualification record form concludes that the CNTS electrical penetration assemblies, including its connection assemblies, performs its safety-related function under the environmental conditions specified in Section 3.11 and the equipment qualification record form; and (3) the required post-accident operability time for the CNTS electrical penetration assemblies in the equipment qualification record form is in agreement with Section 3.11. After installation in the plant, an ITAAC inspection is performed to verify that the CNTS electrical penetration assemblies listed in Tier 1 Table 2.8-1 (Table 14.3-3h), including its connection assemblies, is installed in its design location in a configuration bounded by the equipment qualification record form.	X				

Note:
1. References to Sections, Figures, and Tables in Table 14.3-1 refer to Tier 2, unless the reference specifically states Tier 1 Sections, Figures, or Tables.

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.01.01	CRH	Testing is performed on the CRE in accordance with RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, to demonstrate that air exfiltration from the CRE is controlled. RG 1.197 allows two options for CRE testing; either integrated testing (tracer gas testing) or component testing. Section 6.4 Control Room Habitability, describes the testing requirements for the CRE habitability program. Section 6.4 provides the maximum air exfiltration allowed from the CRE. In accordance with Table 14.2-18, a preoperational test using the tracer gas test method demonstrates that the air exfiltration from the CRE does not exceed the assumed unfiltered leakage rate provided in Table 6.4-1: Control Room Habitability System Design Parameters for the dose analysis. Tracer gas testing in accordance with ASTM E741 will be performed to measure the unfiltered in-leakage into the CRE with the control room habitability system (CRHS) operating.			X		
03.01.02	CRH	The CRHS valves are tested by remote operation to demonstrate the capability to perform their function to transfer open and transfer closed under preoperational temperature, differential pressure, and flow conditions. In accordance with Table 14.2-18, a preoperational test demonstrates that each CRHS valve listed in Tier 1 Table 3.1-1 (Table 14.3-4a) strokes fully open and fully closed by remote operation under preoperational test conditions. Preoperational test conditions are established that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.			X		

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.01.03	CRH	The CRHS solenoid-operated valves are tested to demonstrate the capability to perform their function to fail open on loss of motive power under preoperational temperature, differential pressure, and flow conditions. In accordance with Table 14.2-18 a preoperational test demonstrates that each CRHS solenoid-operated valve listed in Tier 1 Table 3.1-1 (Table 14.3-4a) repositions to the open position on loss of motive power (electric power to the valve actuating solenoid(s) is lost, or pneumatic pressure to the valve(s) is lost). Preoperational test conditions are established that approximate design basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.			X		
03.01.04	CRH	Section 6.4.4, Design Evaluation, discusses the thermal mass of the CRB and its contents limit the temperature increase as shown in Table 6.4-3 within the CRE within an acceptable range for the first 72 hours following a DBA. An analysis confirms that the CRE bulk average air temperature is acceptable on a loss of active cooling for the first 72 hours following a DBA.			X		
03.01.05	CRH	Section 6.4.3.2, Off-Normal Operation, discusses the operation of the CRHS, which maintains a positive pressure in the MCR relative to the adjacent areas. Table 6.4-1: Control Room Habitability System Design Parameters provides the required positive pressure in the MCR relative to the adjacent areas. In accordance with Table 14.2-18, a preoperational test demonstrates that the CRHS maintains a positive pressure of greater than or equal to 1/8 inches water gauge in the MCR relative to adjacent areas, while operating in a DBA alignment.			X		

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.02.01	CRV	The normal control room HVAC system (CRVS) control room envelope isolation dampers are tested to demonstrate the capability to perform their function to fail to the closed position on loss of motive power. In accordance with Table 14.2-19, a preoperational test demonstrates that each CRVS air-operated CRE isolation damper listed in Tier 1 Table 3.2-1 (Table 14.3-4b) repositions to the closed position on loss of motive power (electric power to the valve actuating solenoid is lost, or pneumatic pressure to the damper is lost). Preoperational test conditions are established that approximate design differential pressure conditions to the extent practical, consistent with preoperational test limitations. A manual signal, actual automatic signal, or simulated automatic signal may be used to operate the valves because the control logic of the valves is not being verified by this ITAAC.			X		X
03.02.02	CRV	Section 9.4.1.2, System Description, discusses the operation of the CRVS, which maintains a positive pressure in the CRB relative to the outside environment. This is consistent with the requirements of 10 CFR Part 20, Subparts E and H and 10 CFR Part 50, Appendix I. In accordance with Table 14.2-19 a preoperational test demonstrates that the CRVS will maintain 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 a normal operating alignment.			X		
03.02.03	CRV	Section 9.4.1.2.2.1, Normal Operation, provides a discussion of how the CRVS maintains the hydrogen concentration levels in the CRB battery rooms containing batteries below one percent by volume. In accordance with Table 14.2-19, a preoperational test demonstrates that the airflow capability of the CRVS maintains the hydrogen concentration levels in the CRB battery rooms containing batteries below one percent by volume.			X		

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.03.01	RBV	Section 9.4.2.2.2.1, Normal Operation, and Section 9.4.2.2.2.2, Off-normal Operation, discuss the operation of the Reactor Building HVAC system (RBVS) which maintains a negative pressure in the RXB relative to the outside environment. This is consistent with the requirements of 10 CFR Part 20, Subparts E and H and 10 CFR Part 50, Appendix I. In accordance with Table 14.2-20, a preoperational test demonstrates that the RBVS will maintain a negative pressure in the RXB relative to the outside environment, while operating in a normal operating alignment.			X		
03.03.02	RBV	Section 9.4.2.2.2.1, Normal Operation, and Section 9.4.2.2.2.2, Off-normal Operation, discuss the operation of the RBVS which maintains a negative pressure in the RWB relative to the outside environment. This is consistent with the requirements of 10 CFR Part 20, Subparts E and H and 10 CFR Part 50, Appendix I. In accordance with Table 14.2-20, a preoperational test demonstrates that the RBVS will maintain a negative pressure in the RWB relative to the outside environment, while operating in a normal operating alignment.			X		
03.03.03	RBV	Section 9.4.2.2.2.1, Normal Operation, provides a discussion of how the RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below one percent by volume. In accordance with Table 14.2-20, a preoperational test demonstrates that the airflow capability of the RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below one percent by volume.			X		

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.04.01	FHE	Section 9.1.4, Fuel Handling Equipment, describes that the fuel handling machine (FHM) is classified as a Type I crane as defined by the ASME NOG-1, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder).” An ITAAC inspection is performed of the FHM main and auxiliary hoists machinery arrangement to verify the existence of the following single-failure proof features: (a) nonredundant structural components (bridge, trolley, wire rope drum, and hook) are designed to appropriate standards, constructed from base material demonstrated to meet appropriate material properties, and, (b) redundant design features to stop and hold the load following: • specified component failures (e.g., wire rope, drive train, and control system) • operator errors (e.g., two-blocking and overload) This ITAAC inspection may be performed any time after manufacture of the FHM (at the factory or later).	X				
03.04.02	FHE	Section 9.1.4, Fuel Handling Equipment, describes that the FHM is classified as a Type I crane as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). The FHM main hoist is tested in accordance with the applicable requirements of NOG-1. An FAT demonstrates that the single failure proof FHM main hoist is rated load tested at 125% (+5%, -0%) of the manufacturer’s rating.	X				

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.04.03	FHE	Section 9.1.4, Light-Load Handling System and Refueling Cavity Design, describes that the FHM is classified as a Type I crane as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). In accordance with ASME NOG-1 paragraph 7422, the FHM auxiliary hoist is full load tested at a maximum of 100% of the hoist manufacturer's rating. After the full load test is completed, and prior to use of the crane to handle loads, the FHM auxiliary hoist is rated load tested at 125% (+5%, -0%) of the manufacturer's rating in accordance with ASME NOG-1 paragraph 7423. An FAT demonstrates that the single failure proof FHM auxiliary hoist is rated load tested at 125% (+5%, -0%) of the manufacturer's rating.	X				
03.04.04	FHE	Section 9.1.4, Light-Load Handling System and Refueling Cavity Design, describes that the single failure proof FHM is classified as a Type I crane as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). An ITAAC inspection is performed to verify that the ASME Type I as-built FHM welds are nondestructively examined in accordance with the standards of ASME NOG-1 paragraph 4251.4 and the FHM purchase specification. This ITAAC inspection may be performed any time after manufacture of the single failure-proof FHM (at the factory or later).	X				
03.04.05	FHE	Section 9.1.4, Light-Load Handling System and Refueling Cavity Design, provide descriptions of how the limit switches on the FHM gripper mast limits travel and that the fuel handling equipment (FHE) has provisions to limit maximum lift height of a fuel assembly to maintain a water inventory of 10 feet above the top of the fuel assembly for personnel shielding. In accordance with Table 14.2-51, a preoperational test demonstrates that 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.	X				

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.04.06	FHE	Section 9.1.4, Light-Load Handling System and Refueling Cavity Design, provides description of how the new fuel jib crane hook movement is limited to prevent carrying a fuel assembly over the spent fuel pool to prevent a load drop onto the spent fuel racks. In accordance with Table 14.2-51, a preoperational test demonstrates that the new fuel jib crane interlocks prevent the crane from carrying a fuel assembly over the spent fuel racks.	X				
03.05.01	SFS	The ASME Code Section III requires that documentary evidence be available at the construction or installation site before use or installation to ensure that ASME Code Class NF components conform to the requirements of the Code. The Fuel Storage system ASME Code Class NF components require a Data Report as specified by NCA-1210. The Data Report is prepared by the certificate holder or owner and signed by the certificate holder or owner and the Inspector as specified by NCA-8410. The type of individual Data Report Forms necessary to record the required Code Data is specified in Table NCA-8100-1. An ITAAC inspection is performed of the Data Reports for Fuel Storage system ASME Code Class NF as-built fuel storage racks that are described in Section 9.1.2 to (1) ensure that the appropriate Data Reports have been provided as specified in Table NCA-8100-1 and (2) ensure that the certificate holder or owner and the authorized nuclear inspector have signed the Data Reports, and (3) verify that the requirements of ASME Code Section III are met.	X				
03.05.02	SFS	Section 9.1.1, Criticality Safety of Fresh and Spent Fuel Storage and Handling, discusses the criticality analysis of the fuel storage racks. An ITAAC inspection is performed to verify that the as-built fuel storage racks, including any neutron absorbers, conform to the design values for materials and dimensions and their tolerances, as presented in the approved criticality analysis.	X				

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.06.01	UHS	As required by ASME Code Section III NCA-1210, each ASME Code Class 1, 2 and 3 component (including piping systems) of a nuclear power plant requires a Design Report in accordance with NCA-3550. NCA-3551.1 requires that the drawings used for construction be in agreement with the Design Report before it is certified and be identified and described in the Design Report. It is the responsibility of the N certificate holder to furnish a Design Report for each component and support, except as provided in NCA-3551.2 and NCA-3551.3. NCA-3551.1 also requires that the Design Report be certified by a registered professional engineer when it is for Class 1 components and supports, Class CS core support structures, Class MC vessels and supports, Class 2 vessels designed to NC-3200 (NC-3131.1), or Class 2 or Class 3 components designed to service loadings greater than design loadings. NCA-3554 requires that any modification of any document used for construction, from the corresponding document used for design analysis, shall be reconciled with the Design Report. An ITAAC inspection is performed of the ultimate heat sink (UHS) ASME Code Class 3 as-built piping system Design Report to verify that the e requirements of ASME Code Section III are met.				X	

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.06.02	UHS	The ASME Code Section III requires that documentary evidence be available at the construction or installation site before use or installation to ensure that ASME Code Class 1, 2 and 3 components conform to the requirements of the Code. As defined in NCA-9000, a component can be a vessel, 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 Section III. The UHS ASME Code Class 3 components require a Data Report as specified by NCA-1210. The Data Report is prepared by the certificate holder or owner and signed by the certificate holder or owner and the inspector as specified by NCA-8410. The type of individual Data Report forms necessary to record the required code data is specified in Table NCA-8100-1. An ITAAC inspection is performed of the Data Reports for UHS ASME Code Class 3 as-built components listed in Tier 1 Table 3.6-1 (Table 14.3-4c) and interconnecting piping to (1) ensure that the appropriate Data Reports have been provided as specified in Table NCA-8100-1, (2) ensure that the certificate holder or owner and the authorized nuclear inspector have signed the Data Reports, and (3) verify that the requirements of ASME Code are met.				X	
03.06.03	UHS	Section 9.1.2.2.2, Spent Fuel Storage, Section 9.1.3.2.1, Spent Fuel Pool Cooling System, and Section 9.1.3.2.2, Reactor Pool Cooling System, discuss spent fuel pool (SFP) and reactor pool cooling. No piping, openings, doors, or penetrations through the SFP, refueling pool, reactor pool and dry dock walls are installed below the minimum water level required for shielding, spent fuel cooling, DHRS cooling, or containment cooling. Gates, openings, or drains, permanently connected mechanical or hydraulic systems (piping), and other features that by maloperation or failure could reduce the coolant inventory to unsafe levels are not included in the design. An ITAAC inspection is performed to verify that the SFP, refueling pool, reactor pool and dry dock include no drains, piping or other systems below 80 ft building elevation (55 ft pool level as measured from the bottom of the SFP and reactor pool). This inspection is performed by physical measurements in the as-built SFP and reactor pool.	X				

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.07.01	FP	Section 9.5.1.2.6, Fire Protection Design Features, discusses how the fire protection system (FPS) meets the guidance provided by RG 1.189 and applicable NFPA standards. Two separate dedicated 100 percent capacity freshwater storage tanks are provided. An ITAAC inspection is performed to verify that the minimum usable water volume of each firewater storage tank is greater than or equal to 300,000 gallons. If the storage tanks are also used as backup water sources for other non-fire emergencies, the ITAAC inspection verifies that the non-fire emergencies cannot drain the tank below the minimum dedicated useable water volume of 300,000 gallons required for firefighting.					X
03.07.02	FP	Section 9.5.1, Fire Protection Program, discusses how the capacity of each FPS pump is adequate to supply the total flow demand at the pressure required at the pump discharge. Section 9.5.1 provides the design flow of the fire pumps. i. An analysis confirms that the as-built fire pumps 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. In accordance with Table 14.2-25, a preoperational test demonstrates that each fire pump delivers the design flow to the FPS.					X

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.07.03	FP	Section 9.5.1 discusses that (a) safe-shutdown can be achieved assuming that all equipment in any one fire area (except for the MCR and containment) is rendered inoperable by fire and that reentry into the fire area for repairs and operator actions is not possible, (b) that 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, and (c) an independent alternative shutdown capability that is physically and electrically independent of the MCR exists. A safe shutdown analysis of the as-built plant will be performed, including a post-fire safe shutdown circuit analysis performed in accordance with RG 1.189 and NEI 00-01for all possible fire-induced failures that could affect the safe shutdown success path, including multiple spurious actuations. The safe shutdown analysis will verify that: • safe shutdown can be achieved assuming that all equipment in any one fire area (except for the MCR and containment) 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.					X

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.07.04	FP	Appendix 9A, Fire Hazards Analysis, discusses the methodology and presents the fire hazards analysis (FHA) for each fire area. The FHA must reflect the as-built configuration of the plant. The FHA is an analysis of the fire hazards, including combustible loading and ignition sources, and analysis of the fire protection features required to mitigate each postulated fire. An FHA of the as-built plant will be performed in accordance with RG 1.189, as described in Appendix 9A. The FHA will verify (1) combustible loads and ignition sources are accounted for, and (2) fire protection features are suitable for the hazards they are intended for.					X
03.08.01	PL	Section 9.5.3, Lighting Systems, discusses the plant lighting system (PLS) which provides normal illumination of the operator workstations and SDIS panels in the MCR. The PLS is capable of delivering at least 100 foot-candles of illumination to the MCR seated operator stations and 50 foot-candles of illumination to the MCR primary operating areas and remote and auxiliary operating panels. Lower illumination levels may be used within these areas to ensure more favorable visual conditions, or for areas where critical tasks are not performed. In accordance with Table 14.2-60, a preoperational test demonstrates that the PLS provides at least 100 foot-candles illumination at the MCR operator workstations and at least 50 foot-candles at the MCR auxiliary panels.	X			X	
03.08.02	PL	Section 9.5.3 discusses the PLS which provides emergency illumination of the operator workstations and SDIS panels in the MCR. In accordance with Table 14.2-60, a preoperational test demonstrates that the PLS provides at least 10 foot-candles of illumination at the MCR operator workstations and MCR auxiliary panels.	X			X	

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.08.03	PL	Section 9.5.3 discusses the use of eight-hour battery pack emergency lighting fixtures, which provide illumination of at least one foot-candle for post-fire safe shutdown activities outside of the MCR and RSS in accordance with NFPA 804. These units should provide lighting for: • areas required for power restoration / recovery to comply with the guidance of RG 1.189, Fire Protection for Nuclear Power Plants. • areas where normal actions are required for operation of equipment needed during fire; and • escape or access routes for firefighting and the remote shutdown area. In accordance with the requirements in Table 14.2-60, a preoperational test demonstrates that 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-fire safe-shutdown activities are performed.	X			X	
03.09.01	RM	Section 11.5.2.2.1, Normal Control Room HVAC System, discusses the operation of the CRVS. For each high radiation signal listed in Tier 1 Table 3.9-1 (Table 14.3-4d), the CRVS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table. In accordance with Table 14.2-19, a preoperational test demonstrates the CRVS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table upon initiation of a real or simulated CRVS high radiation signal from 00-CRV-RE-1003, 00-CRV-RE-1004, and 00-CRV-RE-1005.			X		

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.09.02	RM	Section 11.5.2.2.1, Normal Control Room HVAC System, and Section 11.5.2.2.2, Control Room Habitability System, discuss the operation of the CRVS and CRHS. For each high radiation signal listed in Tier 1 Table 3.9-1 (Table 14.3-4d), the CRVS and the CRHS automatically align the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table. In accordance with Table 14.2-18, a preoperational test demonstrates the CRVS and the CRHS automatically align the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table upon initiation of a real or simulated CRVS high radiation signal from 00-CRV-RE-1010 and 00-CRV-RE-1011.			X		
03.09.03	RM	Section 11.5.2.2.3, Reactor Building HVAC System, discusses the operation of the RBVS. For each high radiation signal listed in Tier 1 Table 3.9-1 (Table 14.3-4d), the RBVS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table. In accordance with Table 14.2-20, a preoperational test demonstrates the RBVS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table upon initiation of a real or simulated RBVS high radiation signal from 00-RBV-RE-1010, 00-RBV-RE-1011, and 00-RBV-RE-1012.			X		
03.09.04	RM	Section 11.5.2.2.6, Gaseous Radioactive Waste System, discusses the operation of the gaseous radioactive waste system (GRWS). For each high radiation signal listed in Tier 1 Table 3.9-1 (Table 14.3-4d), the GRWS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table. In accordance with Table 14.2-36, a preoperational test demonstrates the GRWS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table upon initiation of a real or simulated GRWS high radiation signal from 00-GRW-RIT-1021A, GRW-RIT-1021B, and GRW-RIT-1026.			X		
03.09.05		Not Used.					
03.09.06		Not Used.					

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.09.07	RM	Section 11.5.2.1.5, Liquid Radioactive Waste System, discusses the operation of the liquid radioactive waste system (LRWS). For each high radiation signal listed in Tier 1 Table 3.9-1 (Table 14.3-4d), the LRWS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table. In accordance with Table 14.2-35, a preoperational test demonstrates the LRWS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table upon initiation of a real or simulated LRWS high radiation signal from 00-LRW-RIT-1021 and 00-LRW-RIT-1022.			X		
03.09.08	RM	Section 11.5.2.2.14, Auxiliary Boiler System, discusses the operation of the ABS. For each high radiation signal listed in Tier 1 Table 3.9-1 (Table 14.3-4d), the ABS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table. In accordance with Table 14.2-9, a preoperational test demonstrates the ABS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table upon initiation of a real or simulated ABS high radiation signal from 00-AB-RIT-1017 and AB-RIT-1017.			X		
03.09.09		Not Used.					
03.09.10	RM	Section 11.5.2.1.4, Pool Surge Control System, discusses the operation of the pool surge control system (PSCS). For each high radiation signal listed in Tier 1 Table 3.9-1 (Table 14.3-4d), the PSCS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table. In accordance with Table 14.2-4, a preoperational test demonstrates the PSCS automatically aligns the components identified in Tier 1 Table 3.9-1 (Table 14.3-4d) to the required positions identified in the table upon initiation of a real or simulated PSCS high radiation signal from 00-PSC-RE-1003.			X		

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.10.01	RBC	Section 9.1.5, Overhead Heavy Load Handling System, describes that the Reactor Building crane (RBC) main hoist is classified as a Type I crane as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). An ITAAC inspection is performed of the RBC main hoist machinery arrangement to verify the existence of the following single-failure proof features: (a) nonredundant structural components (bridge, trolley, wire rope drum, and hook) are designed to appropriate standards, constructed from base material demonstrated to meet appropriate material properties, and, (b) redundant design features to stop and hold the load following: • specified component failures (e.g., wire rope, drive train, and control system) • operator errors (e.g., two-blocking and overload) This ITAAC inspection may be performed any time after manufacture of the RBC (at the factory or later).				X	
03.10.02	RBC	Section 9.1.5 describes that the RBC auxiliary hoists are classified as a Type I crane as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). An ITAAC inspection is performed of the RBC auxiliary hoists machinery arrangement to verify the existence of the following single-failure proof features: (a) nonredundant structural components (wire rope drum, and hook) are designed to appropriate standards, constructed from base material demonstrated to meet appropriate material properties, and, (b) redundant design features to stop and hold the load following: • specified component failures (e.g., wire rope, drive train, and control system) • operator errors (e.g., two-blocking and overload) This ITAAC inspection may be performed any time after manufacture of the RBC auxiliary hoists (at the factory or later).				X	

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.10.03	RBC	Section 9.1.5 describes that the RBC wet hoist is classified as a Type I crane as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). An ITAAC inspection is performed of the RBC wet hoist arrangement to verify the existence of the following single-failure proof features: (a) nonredundant structural components (bridge, trolley, wire rope drum, and hook) are designed to appropriate standards, constructed from base material demonstrated to meet appropriate material properties, and, (b) redundant design features to stop and hold the load following: • specified component failures (e.g., wire rope, drive train, and control system) • operator errors (e.g., two-blocking and overload) This ITAAC inspection may be performed any time after manufacture of the RBC wet hoist (at the factory or later).				X	
03.10.04	RBC	Section 9.1.5 describes that the RBC main hoist is classified as a Type I crane as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). In accordance with ASME NOG-1, the RBC main hoist is full load tested at a maximum of 100% of the hoist manufacturer's rating. After the full load test is completed, and prior to use of the crane to handle loads, the RBC is rated load tested at 125% (+5%, -0%) of the manufacturer's rating in accordance with ASME NOG-1, paragraph 7423. An FAT demonstrates that the single failure-proof RBC main hoist is rated load tested at 125% (+5%, -0%) of the manufacturer's rating.				X	

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.10.05	RBC	Section 9.1.5 describes that the RBC auxiliary hoists are classified as a Type I crane as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). In accordance with ASME NOG-1, each RBC auxiliary hoist is full load tested at a maximum of 100% of the hoist manufacturer's rating. After the full load test is completed, and prior to use of the RBC auxiliary hoists to handle loads, each RBC auxiliary hoist is rated load tested at 125% (+5%, -0%) of the manufacturer's rating in accordance with ASME NOG-1, paragraph 7423. An FAT demonstrates that the single failure proof RBC auxiliary hoists are rated load tested at 125% (+5%, -0%) of the manufacturer's rating.				X	
03.10.06	RBC	Section 9.1.5 describes that the RBC wet hoist is classified as a Type I crane as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). In accordance with ASME NOG-1, the RBC wet hoist is full load tested at a maximum of 100% of the hoist manufacturer's rating. After the full load test is completed, and prior to use of the RBC wet hoist to handle loads, the RBC wet hoist is rated load tested at 125% (+5%, -0%) of the manufacturer's rating in accordance with ASME NOG-1, paragraph 7423. An FAT demonstrates that the single failure proof RBC wet hoist is rated load tested at 125% (+5%, -0%) of the manufacturer's rating.				X	
03.10.07	RBC	Section 9.1.5 discusses that the single failure-proof RBC is classified as a Type I crane as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). An ITAAC inspection is performed to verify that the ASME Type I as-built RBC welds, including wet hoist welds, are nondestructively examined in accordance with the standards of ASME NOG-1. This ITAAC inspection may be performed any time after manufacture of the single failure proof RBC (at the factory or later).				X	
03.10.08		Not Used.					

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.10.09	RBC	Section 9.1.5.2.2 discusses that the MLA is a single-failure-proof lifting device in accordance with the requirements of ANSI N14.6. In accordance with ANSI N14.6, and as described in Section 9.1.5.4 the portions of the MLA that are single load path are load tested to 300% (+5%, -0%) of the manufacturer's rating. As part of the rated load test, critical areas of the MLA, including all load-bearing welds, will undergo nondestructive testing as required by ANSI N14.6. The portions of the MLA that are dual load path are load tested to 150% (+5%, -0%) of the manufacturer's rating in accordance with ANSI N14.6. As part of the rated load test, critical areas of the MLA, including all load-bearing welds, will undergo nondestructive testing as required by ANSI N14.6. This ITAAC test may be performed any time after manufacture of the MLA (at the factory or later).				X	
03.10.10	RBC	Section 9.1.5.2.2 discusses that the MLA is a single-failure-proof lifting device in accordance with the requirements of ANSI N14.6. An ITAAC inspection is performed of the MLA to verify the existence of the single-failure-proof features described in Section 9.1.5.2.2: (a) dual load paths consisting of four lifting arms, and (b) a pinned clevis used to attach the MLA to the RBC hook eye that is constructed to the 10:1 safety factors of the ultimate strength material. This ITAAC inspection may be performed any time after manufacture of the MLA (at the factory or later).				X	

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.11.01	RXB	Section 9.5.1, Fire Protection Program, discusses that fire and smoke barriers separate: (1) safety-related systems from any potential fires in nonsafety-related areas that could affect the ability of safety-related systems to perform their safety-related function; (2) redundant divisions or trains of safety-related systems from each other to prevent damage that could adversely affect a safe shutdown function from a single fire; (3) equipment within a single safety-related electrical division that present a fire hazard to equipment in another safety-related division; (4) electrical circuits (safety-related and nonsafety-related) whose fire-induced failure could cause a spurious actuation that could adversely affect a safe shutdown function. An ITAAC inspection is performed to verify that the following RXB as-built fire barriers and smoke barriers are installed in accordance with the FHA and are qualified for the fire rating specified in the FHA: • fire-rated doors • fire-rated penetration seals • fire-rated dampers • smoke barriers • fire-rated walls, floors, and ceilings The objective of the inspection is to verify that the fire and smoke barriers meet the design requirements, location requirements, and that they are qualified for their intended use based upon visual inspection and review of the as-built drawings and qualification documentation.		X			X

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.11.02	RXB	Section 3.4.1, Internal Flood Protection for Onsite Equipment Failures, discusses the features used to mitigate the consequences of internal flooding, which include structural enclosures, barriers, curbs, sills, and watertight seals. An ITAAC inspection is performed to verify that the following RXB as-built internal flooding barriers are installed in accordance with the internal flooding analysis report and are qualified as specified in the internal flooding analysis report: • flood resistant doors • curbs and sills • walls • water tight penetration seals • NEMA enclosures The objective of the inspection is to verify that the flooding barriers meet the design requirements, location requirements, and that they are qualified for their intended use based upon visual inspection and review of the as-built drawings and qualification documentation.		X			
03.11.03	RXB	Section 2.4.2, Floods, discusses that the maximum flood elevation (including wind-induced wave run-up) is one foot below baseline plant elevation. Section 3.4.2.1, Probable Maximum Flood, states that the probable maximum flood elevation (including wave action) of the design is one foot below the baseline plant elevation (100'-0). An ITAAC inspection is performed to verify that the RXB as-built floor elevation at ground entrances is located above the maximum external flood elevation to protect the RXB from external flooding. The inspection will compare the maximum external flood elevation against the RXB as-built design drawings to verify that the floor elevation at ground entrances is a minimum of one foot above the maximum external flood elevation.		X			

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.11.04	RXB	Section 12.3, Radiation Protection Design Features, provides the design bases for radiation shielding, including type, form and material properties utilized in specific locations. Radiation shielding is provided to meet the radiation zone and access requirements for normal operation and post-accident conditions, and to demonstrate compliance with 10 CFR 50.49, GDC 4, PDC 19, GDC 61, 10 CFR 50.34(f)(2)(vii), and other relevant requirements. Compartment walls, ceilings, and floors, or other barriers provide shielding. An ITAAC inspection is performed of the RXB radiation barriers to verify wall materials and thicknesses. The required thicknesses are specified in Table 12.3-6. Attenuation capabilities are determined based on wall materials and thicknesses, and an analysis and report will conclude that attenuation capabilities are greater than or equal to the approved design.			X		
03.11.05	RXB	Section 12.3.2.2, Design Considerations, provides the design bases for radiation shielding. Radiation shielding is provided to meet the radiation zone requirements for normal operation and control room access requirements for post-accident conditions. Radiation attenuating doors must meet or exceed the radiation attenuation capability of the wall within which they are installed. An ITAAC inspection is performed to verify that 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 in accordance with the approved door schedule design.			X		

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.11.06	RXB	Section 3.8.4.4, Design and Analysis Procedures, and Appendix 3B, NuScale Plant Critical Sections, provide descriptive information, including plans and sections of each Seismic Category I structure, to establish that there is sufficient information to define the primary structural aspects and elements relied upon for the structure to perform the intended safety functions. Critical dimensions are identified in Appendix 3B. The RXB and its design basis loads are discussed in Section 3.8.4.3, Loads and Load Combinations. Critical sections are the subcomponents of individual Seismic Category I structures (i.e., shear walls, floor slabs and roofs, structure-to-structure connections) that are analytically representative of an essentially complete design. Design basis load combinations are shown in Table 3.8.4-1 and Table 3.8.4-2. A reconciliation analysis of the as-built RXB is performed to ensure the RXB maintains its structural integrity in accordance with the approved design under the actual design basis loads, and the in-structure responses for the RXB are enveloped by those in the approved design. The design summary report provides criteria for the reconciliation between design and as-built conditions, as described in Section 3.8.4.5.1.	X				

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.11.07	RXB	Section 3.2.1, Seismic Classification, discusses that per RG 1.29, some SSC that perform no safety-related functions could, if they failed under seismic loading, prevent or reduce the functioning of Seismic Category I SSC. An ITAAC inspection and analysis is performed to verify that the as-built non-Seismic Category I SSC 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 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.	X				

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.11.08	RXB	Section 3.6, Protection against Dynamic Effects Associated with Postulated Rupture of Piping, provides the design bases and criteria for the analysis required to demonstrate that safety-related SSC are not impacted by the adverse effects of a high-and moderate-energy pipe failure within the plant. An ITAAC inspection is performed to verify that the as-built protective features located in the RXB outside the reactor pool bay credited in the reconciled Pipe Break Hazards Analysis Report (such as pipe whip restraints, pipe whip or jet impingement barriers, jet impingement shields, or guard pipe) have been installed in accordance with design drawings of sufficient detail to show the existence and location of the protective hardware. The as-built inspection is intended to verify that changes to postulated pipe failure locations and protective features or protected equipment made during construction do not adversely affect the safety-related functions of the protected equipment.	X	X			
03.12.01	RWB	Section 12.3, Radiation Protection Design Features, provides the design bases for radiation shielding, including type, form and material properties utilized in specific locations. Radiation shielding is provided to meet the radiation zone requirement for normal operation and post-accident conditions and to demonstrate conformance with GDC 61, RG 4.21, RG 8.8, and other relevent requirements. Compartment walls, ceilings, and floors, or other barriers provide shielding. An ITAAC inspection is performed of the RWB radiation barriers to verify wall materials and thicknesses. The required thicknesses are specified in Table 12.3-7. Attenuation capabilities are determined based on wall materials and thicknesses, and an analysis and report will conclude that attenuation capabilities are greater than or equal to the approved design.			X		

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.12.02	RWB	Section 12.3.2.2, Design Considerations, provides the design bases for radiation shielding. Radiation shielding is provided to meet the radiation zone requirements for normal operation and post-accident conditions, and to demonstrate conformance to RG 4.21 and RG 8.8. Radiation attenuating doors must meet or exceed the radiation attenuation capability of the wall within which they are installed. An ITAAC inspection is performed to verify that 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 in accordance with the approved door schedule design.			X		
03.12.03	RWB	The RW-IIa RWB and its design basis loads are discussed in Section 3.8.4.1.3, Radioactive Waste Building. Design basis loads for RW-IIa structures as listed in RG 1.143. A reconciliation analysis of the as-built RW-IIa RWB is performed to ensure that deviations between the drawings used for construction and the as-built RW-IIa RWB are reconciled and the RW-IIa RWB maintains its structural integrity under the design basis loads in accordance with the approved design under the actual design basis loads, and the in-structure responses for the RWB are enveloped by those in the approved design.	X		X		

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.13.01	CRB	Section 9.5.1, Fire Protection Program, discusses that fire and smoke barriers separate: (1) Safety-related systems from any potential fires in nonsafety-related areas that could affect the ability of safety-related systems to perform their safety-related function. (2) Redundant divisions or trains of safety-related systems from each other to prevent damage that could adversely affect a safe shutdown function from a single fire. (3) Equipment within a single safety-related electrical division that present a fire hazard to equipment in another safety-related division. (4) Electrical circuits (safety-related and nonsafety-related) whose fire-induced failure could cause a spurious actuation that could adversely affect a safe shutdown function. An ITAAC inspection is performed to verify that the following CRB as-built fire barriers and smoke barriers are installed in accordance with the FHA and are qualified for the fire rating specified in the FHA: • fire-rated doors • fire-rated penetration seals • fire-rated dampers • smoke barriers The objective of the inspection is to verify that the fire and smoke barriers meet the design requirements, location requirements, and that they are qualified for their intended use based upon visual inspection and review of the as-built drawings and qualification documentation.		X			X

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.13.02	CRB	Section 3.4.1, Internal Flood Protection for Onsite Equipment Failures, discusses the features used to mitigate the consequences of internal flooding, which include structural enclosures, barriers, and watertight seals. An ITAAC inspection is performed to verify that the following CRB as-built internal flooding barriers are installed in accordance with the internal flooding analysis report and are qualified as specified in the internal flooding analysis report: • flood resistant doors • walls • water tight penetration seals • NEMA enclosures The objective of the inspection is to verify that the flooding barriers meet the design requirements, location requirements, and that they are qualified for their intended use based upon visual inspection and review of the as-built drawings and qualification documentation.		X			
03.13.03	CRB	Section 2.4.2, Floods, discusses that the maximum flood elevation (including wind-induced wave run-up) is one foot below baseline plant elevation. Section 3.4.2.1, Probable Maximum Flood, states that the probable maximum flood elevation (including wave action) of the design is one foot below the baseline plant elevation (100'-0). An ITAAC inspection is performed to verify that the CRB as-built floor elevation at ground entrances is located above the maximum external flood elevation to protect the CRB from external flooding. The inspection will compare the maximum external flood elevation against the CRB as-built design drawings to verify that the floor elevation at ground entrances is a minimum of one foot above the maximum external flood elevation.		X			

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.13.04	CRB	Section 3.8.4.4, Design and Analysis Procedures, and Appendix 3B, NuScale Plant Critical Sections, provide descriptive information, including plans and sections of each Seismic Category I structure, to establish that there is sufficient information to define the primary structural aspects and elements relied upon for the structure to perform the intended safety functions. Critical dimensions are identified in Appendix 3B. The CRB at Elevation 120'-0" and below and its design basis loads are discussed in Section 3.8.4.3, Loads and Load Combinations. Critical sections are the subcomponents of individual Seismic Category I structures (i.e., shear walls, floor slabs and roofs, structure-to-structure connections) that are analytically representative of an essentially complete design. Design basis loads load combinations are shown in Table 3.8.4-1 and Table 3.8.4-2. A reconciliation analysis of the as-built CRB at Elevation 120'-0" and below is performed to ensure the CRB maintains its structural integrity in accordance with the approved design under the actual design basis loads, and the in-structure responses for the CRB are enveloped by those in the approved design. The design summary report provides criteria for the reconciliation between design and as-built conditions, as described in Section 3.8.4.5.1.	X				

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.13.05	CRB	Section 3.2.1, Seismic Classification, discusses that per RG 1.29, some SSC that perform no safety-related functions could, if they failed under seismic loading, prevent or reduce the functioning of Seismic Category I SSC. An ITAAC inspection and analysis is performed to verify that the as-built non-Seismic Category I SSC located where a potential for adverse interaction with a Seismic Category I SSC exists will not impair the ability of Seismic Category I SSC to perform their safety functions as demonstrated by one or more of the following criteria: • The collapse of the non-Seismic Category I structure will not cause the non-Seismic Category I structure to strike a Seismic Category I SSC. • The collapse of the non-Category I structure will not impair the integrity of Seismic Category I SSC, nor result in incapacitating injury to control room occupants. • The non-Category I structure will be analyzed and designed to prevent its failure under SSE conditions.	X				

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.14.01	EQ	Section 3.10, Seismic and Dynamic Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the Seismic Category I equipment, including its associated supports and anchorages, is qualified by type test, analysis, or a combination of type test and analysis to perform its function under the design basis seismic loads during and after an SSE. The qualification method employed for the Seismic Category I equipment is the same as the qualification method described for that type of equipment in Section 3.10. This method conforms to IEEE-344-2004 and ASME QME-1-2007 (or later editions), as accepted by the NRC staff in RG 1.100 Revision 3 (or later revision), with specific revision years and numbers as presented in Section 3.10. The scope of equipment for this design commitment is the common, safety-related equipment, and the common, nonsafety-related equipment that 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 UHS • Provides containment of UHS water • Monitors UHS water level The ITAAC verifies that: (1) a Seismic Qualification Report exists for each Seismic Category I component type, and (2) the seismic qualification record form concludes that the Seismic Category I equipment listed in Tier 1 Table 3.14-1 (Table 14.3-4e), including its associated supports and anchorages, performs its function under the seismic design basis load conditions specified in the Seismic Qualification Report. After installation in the plant, an ITAAC inspection is performed to verify that the Seismic Category I equipment listed in Tier 1 Table 3.14-1 (Table 14.3-4e), including its associated supports and anchorages, is installed in its design location in a Seismic Category I structure in a configuration bounded by the Seismic Qualification Report.	X				

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[(1)] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.14.02	EQ	Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the common electrical equipment, including its connection assemblies, located in a harsh environment is qualified by type test or a combination of type test and analysis to perform its function under design basis harsh environmental conditions, experienced during normal operations, anticipated operational occurrences, DBAs, and post-accident conditions in accordance with 10 CFR 50.49. As defined in IEEE-Std-572-2006, IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations, a connection assembly is any connector or termination combined with related cables or wires as an assembly. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in Section 3.11. The scope of equipment for this design commitment is the nonsafety-related equipment that provides monitoring of the UHS water level and the nonsafety-related electrical equipment on the fuel handling machine and reactor building crane used to physically support irradiated fuel. The ITAAC verifies that: (1) an equipment qualification record form exists for the common electrical equipment listed in Tier 1 Table 3.14-1 (Table 14.3-4e) and addresses connection assemblies, (2) the equipment qualification record form concludes that the common electrical equipment, including its connection assemblies, performs its function under the environmental conditions specified in Section 3.11 and the equipment qualification record form, and (3) the required post-accident operability time for the common electrical equipment in the equipment qualification record form is in agreement with Section 3.11. After installation in the plant, an ITAAC inspection is performed to verify that the common electrical equipment listed in Tier 1 Table 3.14-1 (Table 14.3-4e), including its connection assemblies, is installed in its design location in a configuration bounded by the equipment qualification record form.	X				

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.14.03	EQ	The classification of SSC that contain radioactive waste in accordance with RG 1.143 is discussed in Section 3.2.1.4. The scope of the equipment for this design commitment is the nonsafety-related, radioactive waste components and piping which meet both of the following criteria: • Classified as RW-IIa in accordance with RG 1.143 • Designed for processing gaseous radioactive waste As described in Section 11.2.2.4 for the liquid radioactive waste system (LRWS) and Section 11.3.2.4 for the gaseous radioactive waste system (GRWS), component classification applies to components up to and including the first isolation device. Tier 1 Table 3.14-1 (Table 14.3-4e) identifies the components and piping for which this ITAAC is applicable. An ITAAC inspection and reconciliation analysis is performed of the as-built LRWS and GRWS RW-IIa components and piping used for processing gaseous radioactive waste to ensure that deviations between the drawings used for construction and the as-built RW-IIa components and piping are reconciled. A report concludes the as-built RW-IIa components and piping meet the design criteria of RG 1.143, RW-IIa.			X		
03.15.01	HFE	Section 18.11, Design Implementation, describes the implementation of HFE aspects of the plant design. The Design Implementation activities verify that the final MCR is consistent with the verified and validated design resulting from the HFE design process. An ITAAC inspection is performed to verify that the as-built configuration of main control room HSI is consistent with the final as-designed HSI configuration. As used here, the final as-designed HSI configuration is the COL holder's configuration-controlled design, which includes changes made subsequent to integrated system validation under a licensees configuration control process and includes resolution of human engineering discrepancies.					

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.16.01	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. Vital equipment and vital area are discussed in the report. An ITAAC inspection is performed of the as built vital equipment to verify that the equipment is located in a vital area.					
03.16.02	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. Provisions for accessing vital equipment are discussed in the report. An ITAAC inspection is performed of the as built vital equipment location to verify that access to vital equipment, within the nuclear island and structures, requires passage through at least two physical barriers.					

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.16.03	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. A type test, analysis, or a combination of type test and analysis are performed of the bullet-resisting barriers used in the external walls, doors, ceilings and floors in the MCR, central alarm station, and the last access control function for access to the protected area. This qualification will demonstrate that the barriers are bullet-resistant, to Underwriters Laboratories Ballistic Standard 752, "The Standard of Safety for Bullet-Resisting Equipment," Level 4, or National Institute of Justice Standard 0108.01, "Ballistic Resistant Protective Materials," Type III.					
03.16.04	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. The access control system which limits access to vital areas, within the nuclear island and structures, to individuals with unescorted access authorization is discussed in the report. In accordance with Table 14.2-73, a preoperational test demonstrates that the access control system provides authorized access to vital areas, within the nuclear island and structures, only to those individuals with authorization for unescorted access.					

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.16.05	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. The report discusses that unoccupied vital area portals, within the nuclear island and structures, are equipped with locking devices and alarms that annunciate in the central alarm station. In accordance with Table 14.2-74, a preoperational test, inspection, or a combination of test and inspection demonstrates that unoccupied vital areas, within the nuclear island and structures, are locked and alarmed and intrusion is detected and annunciated in the central alarm station as described in Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems."					
03.16.06	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security related information that describes security design bases and requirements for security SSC. The central alarm station and their location are discussed in the report. An ITAAC inspection is performed of the as built central alarm station to verify that it is located inside the protected area and the interior is not visible from the protected area perimeter.					

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.16.07	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. Security alarms, within the nuclear island and structures, are discussed in the report. In accordance with Table 14.2-74, a preoperational test demonstrates that: (1) alarm annunciation indicates the type of alarm and location. (2) security alarm devices, including transmission lines to annunciators, are tamper-indicating and self-checking. (3) an automatic indication is provided when failure of the alarm system or a component occurs or when the system is on standby power.					
03.16.08	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. The intrusion detection and assessment system, within the nuclear island and structures, is discussed in the report. In accordance with Table 14.2-74, a preoperational test demonstrates that the intrusion detection and assessment system, within the nuclear island and structures, provides visual and audible annunciation of alarms in the central alarm station.					

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.16.09	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. The intrusion detection and assessment system, within the nuclear island and structures, is discussed in the report. In accordance with Table 14.2-74, a preoperational test demonstrates that the intrusion detection and assessment system, within the nuclear island and structures, records each onsite security alarm annunciation, including each alarm, false alarm, alarm check, and tamper indication that identifies the type of alarm, location, alarm circuit, date, and time.					
03.16.10		Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. Emergency exits vital area boundaries, within the nuclear island and structures, are discussed in the report. In accordance with Table 14.2-74, a preoperational test, inspection, or a combination of test and inspection demonstrates that 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 egress during an emergency as described in Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems."					

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.16.11	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. The central alarm station's landline telephone service is discussed in the report. In accordance with Table 14.2-68, a preoperational test, inspection, or a combination of test and inspection demonstrates that the central alarm station is equipped with conventional landline telephone service with the MCR and with local law enforcement authorities as described in Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems."					
03.16.12	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. The central alarm station's communication system is discussed in the report. In accordance with Table 14.2-68, a preoperational test, inspection, or a combination of test and inspection demonstrates that the central alarm station is capable of continuous communication with on-duty security force personnel as described in Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems."					

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference(1) (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.16.13	SEC	Section 13.6 discusses that the 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. Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems," provides safeguards and security-related information that describes security design bases and requirements for security SSC. Nonportable communications equipment in the central alarm station is discussed in the report. In accordance with Table 14.2-68, a preoperational test, inspection, or a combination of test and inspection demonstrates that nonportable communications equipment in the central alarm station remains operable (without disruption) from an independent power source in the event of loss of normal power as described in Technical Report TR-0416-48929, "NuScale Design of Physical Security Systems."					
03.17.01	RM	Section 11.5.2.2.9, Containment Flooding and Drain System, discusses the operation of the containment flooding and drain system (CFDS). For each high radiation signal listed in Tier 1 Table 3.17-1 (Table 14.3-4f), the CFDS automatically aligns the components identified in Tier 1 Table 3.17-1 (Table 14.3-4f) to the required positions identified in the table. In accordance with the information presented in Table 14.2-42, a preoperational test demonstrates the CFDS automatically aligns the components identified in Tier 1 Table 3.17-1 (Table 14.3-4f) to the required positions identified in the table upon initiation of a real or simulated CFDS high radiation signal from 0A-CFD-RT-1007.			X		

Table 14.3-2: Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference[1] (Continued)

ITAAC No.	System	Discussion	DBA	Internal/External Hazard	Radiological	PRA & Severe Accident	FP
03.17.02	RM	Section 11.5.2.2.15, Balance-of-Plant Drain System, discusses the operation of the balance-of-plant drain system (BPDS). For each high radiation signal listed in Tier 1 Table 3.17-1 (Table 14.3-4f), the BPDS automatically aligns the components identified in Tier 1 Table 3.17-1 (Table 14.3-4f) to the required positions identified in the table. In accordance with the information presented in Table 14.2-24, a preoperational test demonstrates the BPDS automatically aligns the components identified in Tier 1 Table 3.17-1 (Table 14.3-4f) to the required positions identified in the table upon initiation of a real or simulated BPDS high radiation signal from 0A-BPD-RIT-1010, 0A-BPD-RIT-1001, and 00-BPD-RIT-1034.			X		
03.18.01	RM	Section 11.5.2.2.9, Containment Flooding and Drain System, discusses the operation of the containment flooding and drain system (CFDS). For each high radiation signal listed in Tier 1 Table 3.18-1 (Table 14.3-4g), the CFDS automatically aligns the components identified in Tier 1 Table 3.18-1 (Table 14.3-4g) to the required positions identified in the table. In accordance with the information presented in Table 14.2-42, a preoperational test demonstrates the CFDS automatically aligns the components identified in Tier 1 Table 3.18-1 (Table 14.3-4g) to the required positions identified in the table upon initiation of a real or simulated CFDS high radiation signal from 0B-CFD-RT-1007.			X		
03.18.02	RM	Section 11.5.2.2.15, Balance-of-Plant Drain System, discusses the operation of the BPDS. For each high radiation signal listed in Tier 1 Table 3.18-1 (Table 14.3-4g), the BPDS automatically aligns the components identified in Tier 1 Table 3.18-1 (Table 14.3-4g) to the required positions identified in the table. In accordance with the information presented in Table 14.2-24, a preoperational test demonstrates the BPDS automatically aligns the components identified in Tier 1 Table 3.18-1 (Table 14.3-4g) to the required positions identified in the table upon initiation of a real or simulated BPDS high radiation signal from 0B-BPD-RIT-1010 and 0B-BPD-RIT-1001.			X		

Note:
1. References to Sections, Figures and Tables in Table 14.3-2 refer to Tier 2 unless the reference specifically states Tier 1 Sections, Figures or Tables.

Table 14.3-3a: 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 CVC-HOV-0331 & CVC-HOV-0330 at CNV nozzle CNV6	N/A (see Note 2)	High	No	0 (see Note 1)
CNTS reactor coolant system pressurizer spray line valves CVC-HOV-0325 & CVC-HOV-0324 at CNV nozzle CNV7	N/A (see Note 2)	High	No	0 (see Note 1)
CNTS reactor coolant system discharge line from valves CVC-HOV-0334 & CVC-HOV-0335 at CNV nozzle CNV13 to NPM disconnect flange	3	High	No	0 (see Note 1)
CNTS reactor coolant system RPV high point degasification line from valves CVC-HOV-0401 & CVC-HOV-0402 at CNV nozzle CNV14 to NPM disconnect flange	3	High	No	0 (see Note 1)
CNTS containment evacuation line valves CE-HOV-0001 & CE-HOV-0002 at CNV nozzle CNV10	N/A (see Note 2)	No	No	0 (see Note 1)
CNTS flood and drain line valves CFD-HOV-0022 & CFD-HOV-0021 at CNV nozzle CNV11	N/A (see Note 2)	No	No	0 (see Note 1)
CNTS control rod drive mechanism cooling water supply line valves RCCW-HOV-0185 & RCCW-HOV-0184 at CNV nozzle CNV12	N/A (see Note 2)	No	No	0 (see Note 1)
CNTS control rod drive mechanism cooling water return line valves RCCW-HOV-0190 & RCCW-HOV-0191 at CNV nozzle CNV5	N/A (see Note 2)	No	No	0 (see Note 1)
CNTS steam generator #1 feedwater line valves FW-HOV-0137 & FW-CKV-0136 at CNV nozzle CNV1	N/A (see Note 2)	High	No	0 (see Note 1)
CNTS steam generator #2 feedwater line valves FW-HOV-0237 & FW-CKV-0236 at CNV nozzle CNV2	N/A (see Note 2)	High	No	0 (see Note 1)
CNTS steam generator #1 steam line from CNV nozzle CNV3 to and including valves MS-HOV-0101 & MS-HOV-0103	2	High	No	4
CNTS steam generator #2 steam line from CNV nozzle CNV4 to and including valves MS-HOV-0201 & MS-HOV-0203	2	High	No	4
DHRS #1 lines from steam generator #1 steam line to DHRS Passive Condenser Train 1 including valves DHR-HOV-0101A and DHR-HOV-0101B	2	High	No	N/A
DHRS #1 condensate line from DHRS Passive Condenser Train 1 to CNV nozzle CNV22	2	High	No	N/A
DHRS #2 lines from steam generator #2 steam line to DHRS Passive Condenser Train 2 including valves DHR-HOV-0201A and DHR-HOV-0201B	2	High	No	N/A
DHRS #2 condensate line from DHRS Passive Condenser Train 2 to CNV nozzle CNV23	2	High	No	N/A

Table 14.3-3a: 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 RPV11 to CNV nozzle CNV6	1	High	No	N/A
RCS pressurizer spray line from RPV nozzles RPV14 and RPV15 to CNV nozzle CNV7	1	High	No	N/A
RCS discharge line from RPV nozzle RPV12 to CNV nozzle CNV13	1	High	No	N/A
RCS RPV high point degasification line from RPV nozzle RPV20 to CNV nozzle CNV14	1	High	No	N/A
CNTS flood and drain line from CNV nozzle CNV11 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 CNV12 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 CNV5	2	No	No	N/A
SGS steam generator #1 feedwater line from RPV nozzles RPV3 and RPV5 to CNV nozzle CNV1	2	High	Yes	N/A
SGS steam generator #2 feedwater line from RPV nozzles RPV4 and RPV6 to CNV nozzle CNV2	2	High	Yes	N/A
SGS steam generator #1 steam line from RPV nozzles RPV8 and RPV10 to CNV nozzle CNV3	2	High	Yes	N/A
SGS steam generator #2 steam line from RPV nozzles RPV7 and RPV9 to CNV nozzle CNV4	2	High	Yes	N/A
DHRS #1 condensate line from CNV nozzle CNV22 to SG #1 feedwater line	2	High	No	N/A
DHRS #2 condensate line from CNV nozzle CNV23 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.
3) This table provides the specific equipment identifiers for the components listed in Tier 1 Table 2.1-1.

Table 14.3-3b: NuScale Power Module Mechanical Equipment

Equipment Name	Equipment Identifier	ASME Code Section III Class	Valve Actuator Type	Containment Isolation Valve
RCS integral RPV/SG/Pressurizer	RPV-VSL-0001	1	N/A	N/A
RVI upper core plate	N/A	CS	N/A	N/A
RVI core barrel	N/A	CS	N/A	N/A
RVI reflector blocks	N/A	CS	N/A	N/A
RVI lower core plate	N/A	CS	N/A	N/A
RVI core support blocks	N/A	CS	N/A	N/A
CNTS containment vessel	CNT-VSL-0001	1	N/A	N/A
RCS reactor safety valves (2 Total)	RCS-PSV-0003A RCS-PSV-0003B	1	N/A	No
CNTS pressurizer spray check valve	CVC-CKV-0323	3	N/A	No
CNTS injection check valve	CVC-CKV-0329	3	N/A	No
CNTS discharge excess flow check valve	CVC-CKV-0336	3	N/A	No
ECCS reactor vent valves (3 Total)	ECC-HOV-0001A ECC-HOV-0001B ECC-HOV-0001C	1	Hydraulic	No
ECCS reactor vent valve trip valves (4 Total)	ECC-SV-0101A ECC-SV-0101B ECC-SV-0101C-1 ECC-SV-0101C-2	1	Solenoid	No
ECCS reactor vent valve reset valves (3 Total)	ECC-SV-0103A ECC-SV-0103B ECC-SV-0103C	1	Solenoid	No
ECCS reactor recirculation valves (2 Total)	ECC-HOV-0002A ECC-HOV-0002B	1	Hydraulic	No
ECCS reactor recirculation valve trip valves (2 Total)	ECC-SV-0102A ECC-SV-0102B	1	Solenoid	No
ECCS reactor recirculation valve reset valves (2 Total)	ECC-SV-0104A ECC-SV-0104B	1	Solenoid	No
CNTS solenoid valves	None	1	Solenoid	No
CNTS reactor coolant system injection inboard CIV	CVC-HOV-0331	1	Electro-hydraulic	Yes
CNTS reactor coolant system injection outboard CIV	CVC-HOV-0330	1	Electro-hydraulic	Yes
CNTS pressurizer spray inboard CIV	CVC-HOV-0325	1	Electro-hydraulic	Yes
CNTS pressurizer spray outboard CIV	CVC-HOV-0324	1	Electro-hydraulic	Yes
CNTS reactor coolant system discharge inboard CIV	CVC-HOV-0334	1	Electro-hydraulic	Yes
CNTS reactor coolant system discharge outboard CIV	CVC-HOV-0335	1	Electro-hydraulic	Yes
CNTS reactor pressure vessel high point degasification inboard CIV	CVC-HOV-0401	1	Electro-hydraulic	Yes
CNTS reactor pressure vessel high point degasification outboard CIV	CVC-HOV-0402	1	Electro-hydraulic	Yes
CNTS containment evacuation inboard CIV	CE-HOV-0001	1	Electro-hydraulic	Yes

Table 14.3-3b: NuScale Power Module Mechanical Equipment (Continued)

Equipment Name	Equipment Identifier	ASME Code Section III Class	Valve Actuator Type	Containment Isolation Valve
CNTS containment evacuation outboard CIV	CE-HOV-0002	1	Electro-hydraulic	Yes
CNTS flood and drain inboard CIV	CFD-HOV-0022	1	Electro-hydraulic	Yes
CNTS flood and drain outboard CIV	CFD-HOV-0021	1	Electro-hydraulic	Yes
CNTS reactor component cooling water system supply inboard CIV	RCCW-HOV-0185	1	Electro-hydraulic	Yes
CNTS reactor component cooling water system supply outboard CIV	RCCW-HOV-0184	1	Electro-hydraulic	Yes
CNTS reactor component cooling water system return inboard CIV	RCCW-HOV-0190	1	Electro-hydraulic	Yes
CNTS reactor component cooling water system return outboard CIV	RCCW-HOV-0191	1	Electro-hydraulic	Yes
Control rod drive system thermal relief valve	CRDS-RV-0221	2	N/A	No
CNTS feedwater #1 CIV	FW-HOV-0137	2	Electro-hydraulic	Yes
CNTS feedwater line #1 check valve	FW-CKV-0136	2	N/A	No
Steam generator #1 relief valve	SG-RV-0102	2	N/A	Yes
CNTS feedwater #2 CIV	FW-HOV-0237	2	Electro-hydraulic	Yes
CNTS feedwater line #2 check valve	FW-CKV-0236	2	N/A	No
Steam generator #2 relief valve	SG-RV-0202	2	N/A	Yes
CNTS main steam #1 CIV	MS-HOV-0101	2	Electro-hydraulic	Yes
CNTS main steam line #1 bypass valve CIV	MS-HOV-0103	2	Electro-hydraulic	Yes
CNTS main steam #2 CIV	MS-HOV-0201	2	Electro-hydraulic	Yes
CNTS main steam line #2 bypass valve CIV	MS-HOV-0203	2	Electro-hydraulic	Yes
DHRS actuation valves (4 Total)	DHR-HOV-0101A DHR-HOV-0101B DHR-HOV-0201A DHR-HOV-0201B	2	Electro-hydraulic	No
DHRS passive condensers (2 Total)	DHR-CND-0103 DHR-CND-0203	2	N/A	N/A
CRDM heat exchangers (16 Total)	CRDS-CRD-0001 thru CRDS-CRD-0016	2	N/A	N/A
CRDM cooling water supply flex hoses (16 Total)	CRDS-FHS-0101 thru CRDS-FHS-0116	2	N/A	N/A
CRDM cooling water return flex hoses (16 Total)	CRDS-FHS-0201 thru CRDS-FHS-0216	2	N/A	N/A
CRDM latch housing	N/A	1	N/A	N/A
CRDM rod travel housing	N/A	1	N/A	N/A
CRDM rod travel housing plug	N/A	1	N/A	N/A
CNTS I&C Division I Electrical Penetration Assembly (EPA)	CNV8	1	N/A	N/A

Table 14.3-3b: NuScale Power Module Mechanical Equipment (Continued)

Equipment Name	Equipment Identifier	ASME Code Section III Class	Valve Actuator Type	Containment Isolation Valve
CNTS I&C Division II Electrical Penetration Assembly (EPA)	CNV9	1	N/A	N/A
CNTS PZR Heater Power #1 Electrical Penetration Assembly (EPA)	CNV15	1	N/A	N/A
CNTS PZR Heater Power #2 Electrical Penetration Assembly (EPA)	CNV16	1	N/A	N/A
CNTS I&C Channel A Electrical Penetration Assembly (EPA)	CNV17	1	N/A	N/A
CNTS I&C Channel B Electrical Penetration Assembly (EPA)	CNV18	1	N/A	N/A
CNTS I&C Channel C Electrical Penetration Assembly (EPA)	CNV19	1	N/A	N/A
CNTS I&C Channel D Electrical Penetration Assembly (EPA)	CNV20	1	N/A	N/A
CNTS CRD Power Electrical Penetration Assembly (EPA)	CNV37	1	N/A	N/A
CNTS RPI Group #1 Electrical Penetration Assembly (EPA)	CNV38	1	N/A	N/A
CNTS RPI Group #2 Electrical Penetration Assembly (EPA)	CNV39	1	N/A	N/A
RPV Instrument Seal Assemblies (4 Total)	RPV39 RPV40 RPV41 RPV42	1	N/A	N/A

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 2.1-2.

Table 14.3-3c: NuScale Power Module Electrical Equipment

Equipment Name	Equipment Identifier	Remotely Operated	Loss of Motive Power Position	CIV Closure Time (sec)
ECCS reactor vent valve trip valves (4 Total)	ECC-SV-0101A ECC-SV-0101B ECC-SV-0101C-1 ECC-SV-0101C-2	Yes	Open	N/A
ECCS reactor vent valve reset valves (3 Total)	ECC-SV-0103A ECC-SV-0103B ECC-SV-0103C	Yes	Close	N/A
ECCS reactor recirculation valve trip valves (2 Total)	ECC-SV-0102A ECC-SV-0102B	Yes	Open	N/A
ECCSreactor recirculation valve reset valves (2 Total)	ECC-SV-0104A ECC-SV-0104B	Yes	Close	N/A
CNTS reactor coolant system injection inboard CIV	CVC-HOV-0331	Yes	Closed	≤ 7
CNTS reactor coolant system injection outboard CIV	CVC-HOV-0330	Yes	Closed	≤ 7
CNTS pressurizer spray inboard CIV	CVC-HOV-0325	Yes	Closed	≤ 7
CNTS pressurizer spray outboard CIV	CVC-HOV-0324	Yes	Closed	≤ 7
CNTS reactor coolant system discharge inboard CIV	CVC-HOV-0334	Yes	Closed	≤ 7
CNTS reactor coolant system discharge outboard CIV	CVC-HOV-0335	Yes	Closed	≤ 7
CNTS reactor pressure vessel high point degasification inboard CIV	CVC-HOV-0401	Yes	Closed	≤ 7
CNTS reactor pressure vessel high point degasification outboard CIV	CVC-HOV-0402	Yes	Closed	≤ 7
CNTS containment evacuation inboard CIV	CE-HOV-0001	Yes	Closed	≤ 7
CNTS containment evacuation outboard CIV	CE-HOV-0002	Yes	Closed	≤ 7
CNTS flood and drain inboard CIV	CFD-HOV-0022	Yes	Closed	≤ 7
CNTS flood and drain outboard CIV	CFD-HOV-0021	Yes	Closed	≤ 7
CNTS reactor component cooling water system supply inboard CIV	RCCW-HOV-0185	Yes	Closed	≤ 7
CNTS reactor component cooling water system supply outboard CIV	RCCW-HOV-0184	Yes	Closed	≤ 7
CNTS reactor component cooling water system return inboard CIV	RCCW-HOV-0190	Yes	Closed	≤ 7
CNTS reactor component cooling water system return outboard CIV	RCCW-HOV-0191	Yes	Closed	≤ 7
CNTS feedwater #1 CIV	FW-HOV-0137	Yes	Closed	≤ 7
CNTS feedwater #2 CIV	FW-HOV-0237	Yes	Closed	≤ 7
CNTS main steam #1 CIV	MS-HOV-0101	Yes	Closed	≤ 7
CNTS main steam line #1 bypass valve CIV	MS-HOV-0103	Yes	Closed	≤7
CNTS main steam #2 CIV	MS-HOV-0201	Yes	Closed	7
CNTS main steam line #2 bypass valve CIV	MS-HOV-0203	Yes	Closed	≤7
DHRS actuation valves (4 Total)	DHR-HOV-0101A DHR-HOV-0101B DHR-HOV-0201A DHR-HOV-0201B	Yes	Open	N/A
CNTS I&C Division I Electrical Penetration Assembly (EPA)	CNV8	N/A	N/A	N/A

Table 14.3-3c: NuScale Power Module Electrical Equipment (Continued)

Equipment Name	Equipment Identifier	Remotely Operated	Loss of Motive Power Position	CIV Closure Time (sec)
CNTS I&C Division II Electrical Penetration Assembly (EPA)	CNV9	N/A	N/A	N/A
CNTS PZR Heater Power #1 Electrical Penetration Assembly (EPA)	CNV15	N/A	N/A	N/A
CNTS PZR Heater Power #2 Electrical Penetration Assembly (EPA)	CNV16	N/A	N/A	N/A
CNTS I&C Channel A Electrical Penetration Assembly (EPA)	CNV17	N/A	N/A	N/A
CNTS I&C Channel B Electrical Penetration Assembly (EPA)	CNV18	N/A	N/A	N/A
CNTS I&C Channel C Electrical Penetration Assembly (EPA)	CNV19	N/A	N/A	N/A
CNTS I&C Channel D Electrical Penetration Assembly (EPA)	CNV20	N/A	N/A	N/A
CNTS CRD Power Electrical Penetration Assembly (EPA)	CNV37	N/A	N/A	N/A
CNTS RPI Group #1 Electrical Penetration Assembly (EPA)	CNV38	N/A	N/A	N/A
CNTS RPI Group #2 Electrical Penetration Assembly (EPA)	CNV39	N/A	N/A	N/A

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 2.1-3.

Table 14.3-3d: Chemical and Volume Control System Piping

Piping System Description	ASME Code Section III Class
Demineralized water supply line between valves CVC-AOV-0089 and CVC-AOV-0090	3
Reactor pressure vessel (RPV) discharge line from the NPM disconnect flange downstream of CVC Discharge containment isolation valve CVC-HOV-0335 up to and including RPV discharge isolation valve CVC-AOV-0001 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 CVC-SV-0079 and NPM removable spool piece	3

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 2.2-1.

Table 14.3-3e: Chemical and Volume Control System Mechanical Equipment

Equipment Name	Equipment Identifier	ASME Code Section III Class	Loss of Motive Power Position
Demineralized water system supply isolation valves (2 Total)	CVC-AOV-0089 CVC-AOV-0090	3	Closed
RPV discharge isolation valve	CVC-AOV-0001	3	N/A
RPV high point degasification isolation valve	CVC-SV-0079	3	N/A

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 2.2-2.

Table 14.3-3f: Important Human Actions Controls

Tag No.	Component Description	Operation
CFDS Emergency Flooding		
MPS-HS-1S-0001	Division I enable nonsafety control switch	Enable
MPS-HS-2S-0001	Division II enable nonsafety control switch	Enable
MPS-HS-1S-0004	Division I override switch	Override
MPS-HS-2S-0004	Division II override switch	Override
CFD-AOV-0012	Containment drain inlet valve	Close
CFD-AOV-0013	Containment drain discharge valve	Close
CFD-AOV-0017	Containment drain separator gas discharge valve	Close
CFD-AOV-0010	Module flood isolation valve	Open/Close
CFD-AOV-0002	Pool suction isolation valve	Open
CFD-AOV-0009	CFDS flood/drain selector valve	Open
CFD-AOV-0007	CFDS pump discharge flow control valve	Open
CFD-HV-0011	System priming valve	Open/Close
CFD-HV-0102A	CFDS pump A case vent valve	Open/Close
CFD-HV-0102B	CFDS pump B case vent valve	Open/Close
CFD-HOV-0021	Module flood outboard CIV	Open/Close
CFD-HOV-0022	Module flood inboard CIV	Open/Close
CFD-P-0004A	CFDS pump A	Start/Stop
CFD-P-0004B	CFDS pump B	Start/Stop
CVCS Injection Following Containment Isolation		
MPS-HS-1S-0001	Division I enable nonsafety control switch	Enable
MPS-HS-2S-0001	Division II enable nonsafety control switch	Enable
MPS-HS-1S-0004	Division I override switch	Override
MPS-HS-2S-0004	Division II override switch	Override
BAS-P-0012A	Boric acid supply pump A	Start/Stop
BAS-P-0012B	Boric acid supply pump B	Start/Stop
BAS-AOV-0016	CVCS makeup aligning valve	Open
CVC-AOV-0096	Boric acid supply to CVCS makeup pumps	Open
CVC-AOV-0091	CVCS three-way valve	Open
CVC-AOV-0050	CVCS isolation valve	Open
CVC-AOV-0072	CVCS to module heatup system isolation valve	Open
CVC-AOV-0073	CVCS from module heatup system isolation valve	Open
CVC-MOV-0052	CVCS isolation valve	Open
CVC-FCV-0053	CVCS isolation valve	Open
CVC-AOV-0055	CVCS isolation valve	Open
CVC-HOV-0330 CVC-HOV-0331	RCS injection CIVs (2 Total)	Open/Close
CVC-HOV-0324 CVC-HOV-0325	Pressurizer spray CIVs (2 Total)	Open/Close
CVC-P-0098A	CVCS makeup pump A	Start/Stop
CVC-P-0098B	CVCS makeup pump B	Start/Stop

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 2.5-6.

Table 14.3-3g: Radiation Monitoring - Module-Specific Automatic Actions

Radiation Monitor ID(s)	Variable Monitored	Actuated Component(s)	Component ID(s)	Component Action(s)
CES-RT-1011	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. CES-A0V-0017 2. CES-A0V-0040 3. CES-SV-0037 4. CES-SV-0024A 5. CES-SV-0025A 6. CES-SV-0026A 7. CES-SV-0024B 8. CES-SV-0025B 9. CES-SV-0026B	1. Close 2. Open 3. Close 4. Close 5. Close 6. Close 7. Close 8. Close 9. Close
CVC-RT-1004	Reactor coolant system discharge to regenerative heat exchanger	1. Reactor coolant system discharge to process sampling system isolation valve	1. CVC-AOV-0059	1. Close
0A-AB-RIT-1005	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. CVC-AOV-0074 2. CVC-AOV-0075	1. Close 2. Close
0B-AB-RIT-1005	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. CVC-AOV-0074 2. CVC-AOV-0075	1. Close 2. Close

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 2.7-1.

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
Containment System							
CNV-8	CNTS I&C Division I Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNV-9	CNTS I&C Division II Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNV-15	CNTS PZR Heater Power #1 Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNV-16	CNTS PZR Heater Power #2 Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNV-17	CNTS I&C Channel A Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	Yes	A
CNV-18	CNTS I&C Channel B Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	Yes	A
CNV-19	CNTS I&C Channel C Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	Yes	A
CNV-20	CNTS I&C Channel D Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	Yes	A
CNV-37	CNTS CRD Power Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNV-38	CNTS RPI Group #1 Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
CNV-39	CNTS RPI Group #2 Electrical Penetration Assembly (EPA)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	No	A
MS-HOV-0101	MS #1 CIV (MSIV #1)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
MS-HOV-0201	MS #2 CIV (MSIV #2)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
MS-HOV-0103	MS line #1 Bypass Valve (MSIV Bypass #1)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
MS-HOV-0203	MS line #2 Bypass Valve (MSIV Bypass #2)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
FW-HOV-0137	FW #1 CIV (FWIV #1)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
FW-HOV-0237	FW #2 CIV (FWIV #2)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
FW-HOV-0136	FW line #1 Check Valve	RXB - Top of Module	Harsh	Mechanical	Yes	N/A	A B
FW-HOV-0236	FW line #2 Check Valve	RXB - Top of Module	Harsh	Mechanical	Yes	N/A	A B
CVC-HOV-0334 CVC-HOV-0335	CVC Discharge CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CVC-HOV-0330 CVC-HOV-0331	CVC Injection CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CVC-HOV-0324 CVC-HOV-0325	CVC PZR Spray CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CVC-HOV-0401 CVC-HOV-0402	RPV High Point Degas CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
RCCW-HOV-0184 RCCW-HOV-0185	RCCW Supply CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
RCCW-HOV-0190 RCCW-HOV-0191	RCCW Return CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CE-HOV-0001 CE-HOV-0002	CE CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CFD-HOV-0021 CFD-HOV-0022	CFDS CIVs (2 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A B
CVC-CKV-0323 CVC-CKV-0329 CVC-CKV-0336	CNTS Check Valves and Excess Flow Check Valves (3 Total)	RXB - Top of Module	Harsh	Mechanical	Yes	N/A	A B
CNT-SKD-501 CNT-SKD-502	Hydraulic Skids (2 Total)	RXB - 100′ RXB - 120'	Harsh	Electrical Mechanical	Yes	No	A
CNT-PE-1001A CNT-PE-1001B CNT-PE-1001C CNT-PE-1001D	Containment Pressure Transducers (Narrow Range) (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
CNT-PE-1002A CNT-PE-1002B	Containment Pressure Transducers (Wide Range) (2 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	No	A

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
CNT-LE-1003A CNT-LE-1003B CNT-LE-1003C CNT-LE-1003D	Containment Water Level Sensors (Radar Transceiver) (4 Total)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
MS-TE-1001A MS-TE-1001B MS-TE-1001C MS-TE-1001D	SG #1 Steam Temperature Sensors (RTD) (4 Total)	RXB - Top of Module	Harsh	Electrical	Yes	Yes	A
MS-TE-2001A MS-TE-2001B MS-TE-2001C MS-TE-2001D	SG #2 Steam Temperature Sensors (RTD) (4 Total)	RXB - Top of Module	Harsh	Electrical	Yes	Yes	A
CE-ZSC-0001	CE Inboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CE-ZSO-0001	CE Inboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CE-ZSC-0002	CE Outboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CE-ZSO-0002	CE Outboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CFD-ZSC-0022	CFD Inboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CFD-ZSO-0022	CFD Inboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CFD-ZSC-0021	CFD Outboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CFD-ZSO-0021	CFD Outboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSC-0334	CVCS Inboard RCS Discharge CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSO-0334	CVCS Inboard RCS Discharge CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSC-0335	CVCS Outboard CIV RCS Discharge Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSO-0335	CVCS Outboard CIV RCS Discharge Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSC-0331	CVCS Inboard RCS Injection CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSO-0331	CVCS Inboard RCS Injection CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
CVC-ZSC-0330	CVCS Outboard RCS Injection CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSO-0330	CVCS Outboard RCS Injection CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSC-0325	CVCS Inboard PZR Spray Line CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSO-0325	CVCS Inboard PZR Spray Line CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSC-0324	CVCS Outboard PZR Spray Line CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSO-0324	CVCS Outboard PZR Spray Line CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSC-0401	CVCS Inboard RPV High- Point Degasification CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSO-0401	CVCS Inboard RPV High- Point Degasification CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSC-0402	CVCS Outboard RPV High-Point Degasification CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
CVC-ZSO-0402	CVCS Outboard RPV High-Point Degasification CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW-ZSC-0185	RCCW Supply Inboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW-ZSO-0185	RCCW Supply Inboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW-ZSC-0184	RCCW Supply Outboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW-ZSO-0184	RCCW Supply Outboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW-ZSC-0190	RCCW Return Inboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW-ZSO-0190	RCCW Return Inboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
RCCW-ZSC-0191	RCCW Return Outboard CIV Close Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[(1)]
RCCW-ZSO-0191	RCCW Return Outboard CIV Open Position Sensor	RXB - Top of Module	Harsh	Electrical	Yes	No	A
FW-ZSO-0137A FW-ZSO-0137B	FW Supply to SG1 and DHR HX1 CIV/FWIV Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
FW-ZSC-0137A FW-ZSC-0137B	FW Supply to SG1 and DHR HX1 CIV/FWIV Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
FW-ZSC-0237A FW-ZSC-0237B	FW Supply to SG2 and DHR HX2 CIV/FWIV Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
FW-ZSO-0237A FW-ZSO-0237B	FW Supply to SG2 and DHR HX2 CIV/FWIV Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
MS-ZSC-0101A MS-ZSC-0101B	SG1 Steam Supply CIV/ MSIV Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
MS-ZSO-0101A MS-ZSO-0101B	SG1 Steam Supply CIV/ MSIV Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
MS-ZCS-0103A MS-ZSC-0103B	SG1 Steam Supply CIV/ MS Bypass Isolation Valve Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
MS-ZSO-0103A MS-ZSO-0103B	SG1 Steam Supply CIV/ MS Bypass Isolation Valve Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
MS-ZSC-0201A MS-ZSC-0201B	SG2 Steam Supply CIV/ MSIV Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
MS-ZSO-0201A MS-ZSO-0201B	SG2 Steam Supply CIV/ MSIV Open Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
MS-ZSC-0203A MS-ZSC-0203B	SG2 Steam Supply CIV/ MS Bypass Isolation Valve Close Position Sensors (2 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
MS-ZSO-0203A MS-ZSO-0203B	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							
None	SG Tubes and Tube Supports	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
RPV7 RPV8 RPV9 RPV10	Steam Plenums (4 Total)	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
RPV3 RPV4 RPV5 RPV6	Feedwater Plenums (4 Total)	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
None	Flow Restrictors	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
SG-PSV-1002 SG-PSV-2002	Thermal Relief Valves (2 Total)	RXB - Inside Containment	Harsh	Mechanical	Yes	N/A	B
Control Rod Drive System							
CRDS-ZS-0001A to 0016A CRDS-ZS-0001B to 0016B	Rod Position Indication (RPI) Coils (32 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	No	B
None	Control Rod Drive Shafts	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
None	Control Rod Drive Latch Mechanism	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
None	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							
None	All components	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Neutron Source Assembly							
None	Primary and secondary neutron source rodlets	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
None	Spider body, hub or coupling housing	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A
Reactor Coolant System							
None	Reactor Vessel Internals	RXB - Inside Containment	N/A	N/A	Yes	N/A	N/A

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
RCS-ZSO-003A RCS-ZSC-003A RCS-ZSO-003B RCS-ZSC-003B	Reactor Safety Valve Position Indicators (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	No	B
RCS-PSV-0003A RCS-PSV-0003B	Reactor Safety Valves (2 Total)	RXB - Inside Containment	Harsh	Electrical Mechanical	Yes	N/A	A
RCS-PE-1013A RCS-PE-1013B RCS-PE-1013C RCS-PE-1013D	Narrow Range Pressurizer Pressure Elements (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
RCS-PE-1014A RCS-PE-1014B RCS-PE-1014C RCS-PE-1014D	Wide Range RCS Pressure Elements (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
RCS-LE-1015A RCS-LE-1015B RCS-LE-1015C RCS-LE-1015D	PZR/RPV Level Elements (4 Total)	RXB - Top of Module RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
RCS-TE-1005A RCS-TE-1005B RCS-TE-1005C RCS-TE-1005D RCS-TE-1006A RCS-TE-1006B RCS-TE-1006C RCS-TE-1006D RCS-TE-1007A RCS-TE-1007B RCS-TE-1007C RCS-TE-1007D	Narrow Range RCS Hot Leg Temperature Elements (12 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[(1)]
RCS-TE-1008A RCS-TE-1008B RCS-TE-1008C RCS-TE-1008D	Wide Range RCS Hot Leg Temperature Elements (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
RCS-TE-1011A RCS-TE-1011B RCS-TE-1011C RCS-TE-1011D	Wide Range RCS Cold Leg Temperature Elements (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	B
RCS-FE-1012A RCS-FE-1012B RCS-FE-1012C RCS-FE-1012D	RCS Flow Transmitters (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	Yes	A
RCS-HT-0002A RCS-HT-0002B	PZR Heaters (2 Total)	RXB - Inside Containment	N/A	N/A	Yes	No	N/A
Chemical and Volume Control System							
CVC-AOV-0089 CVC-AOV-0090	DWS Supply Isolation Valves (2 Total)	RXB - 50′	Harsh	Electrical Mechanical	Yes	Yes	A B
Emergency Core Cooling System							
ECC-HOV-0001A ECC-HOV-0001B ECC-HOV-0001C	Reactor Vent Valves (3 Total)	RXB - Inside Containment	Harsh	Mechanical	Yes	No	A
ECC-ZSC-0001A ECC-ZSO-0001A ECC-ZSC-0001B ECC-ZSO-0001B ECC-ZSC-0001C-1 ECC-ZSO-0001C-1 ECC-ZSC-0001C-2 ECC-ZSO-0001C-2	RVV Position Indications (8 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	No	A
ECC-HOV-0002A ECC-HOV-0002B	Reactor Recirculation Valves (2 Total)	RXB - Inside Containment	Harsh	Mechanical	Yes	No	A

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
ECC-ZSC-0002A ECC-ZSO-0002A ECC-ZSC-0002B ECC-ZSO-0002B	RRV Position Indications (4 Total)	RXB - Inside Containment	Harsh	Electrical	Yes	No	A
ECC-SV-0101A ECC-SV-0101B ECC-SV-0101C-1 ECC-SV-0101C-2	RVV Trip Valves (4 Total)	RXB - Pool	Harsh	Electrical Mechanical	Yes	Yes	A B
ECC-SV-0102A ECC-SV-0102B	RRV Trip Valves (2 Total)	RXB - Pool	Harsh	Electrical Mechanical	Yes	Yes	A B
ECC-ZSC-0101A ECC-ZSO-0101A ECC-ZSC-0101B ECC-ZSO-0101B ECC-ZSC-0101C-1 ECC-ZSO-0101C-1 ECC-ZSC-0101C-2 ECC-ZSO-0101C-2	RVV Trip Valve Position Indications (8 Total)	RXB - Pool	Harsh	Electrical	Yes	No	A
ECC-ZSC-0102A ECC-ZSO-0102A ECC-ZSC-0102B ECC-ZSO-0102B	RRV Trip Valve Position Indications (4 Total)	RXB - Pool	Harsh	Electrical	Yes	No	A
ECC-SV-0103A ECC-SV-0103B ECC-SV-0103C	RVV Reset Valves (3 Total)	RXB - Pool	Harsh	Electrical Mechanical	Yes	No	A
ECC-SV-0104A ECC-SV-0104B	RRV Reset Valves (2 Total)	RXB - Pool	Harsh	Electrical Mechanical	Yes	No	A

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
Decay Heat Removal System							
DHR-HOV-0101A DHR-HOV-0101B DHR-HOV-0201A DHR-HOV-0201B	DHRS Actuation Valves (4 Total)	RXB - Top of Module	Harsh	Electrical Mechanical	Yes	Yes	A
DHR-TT-1003A DHR-TT-1003B DHR-TT-2003A DHR-TT-2003B	DHRS Condenser Outlet Temperature Transmitters (4 Total)	RXB - Pool	Harsh	Electrical	Yes	No	A
DHR-PT-1004A DHR-PT-1004B DHR-PT-1004C DHR-PT-2004A DHR-PT-2004B DHR-PT-2004C	DHRS Condenser Outlet Pressure Transmitters (6 Total)	RXB - Pool	Harsh	Electrical	Yes	No	A
DHR-ZSO-0101A DHR-ZSC-0101A DHR-ZSO-0101B DHR-ZSC-0101B DHR-ZSO-0201A DHR-ZSC-0201A DHR-ZSO-0201B DHR-ZSC-0201B	DHRS Valve Position Indicators (8 Total)	RXB - Top of Module	Harsh	Electrical	Yes	No	A
DHR-CND-0103 DHR-CND-0203	Condensers (2 Total)	RXB - Side of Module	N/A	N/A	Yes	N/A	N/A

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
DHR-PT-1001A DHR-PT-1001B DHR-PT-1001C DHR-PT-1001D DHR-PT-2001A DHR-PT-2001B DHR-PT-2001C DHR-PT-2001D	SG Steam Pressure Transmitters (8 Total)	RXB - Top of Module	Harsh	Electrical	Yes	Yes	A
Main Steam System							
MS-AOV-0102 MS-AOV-0202	Secondary Main Steam Isolation Valves (2 Total)	RXB - 100'	Harsh	Electrical Mechanical	Yes	No	A B
MS-AOV-0104 MS-AOV-0204	Secondary Main Steam Isolation Bypass Valves (2 Total)	RXB - 100'	Harsh	Electrical Mechanical	Yes	No	A B
Condensate and Feedwater System							
FW-FCV-0134 FW-FCV-0234	Feedwater Regulating Valves A/B (2 Total)	RXB - 100’	Harsh	Electrical Mechanical	Yes	No	A
FW-CKV-0135 FW-CKV-0235	Feedwater Supply Check Valves (2 Total)	RXB - 100’	Harsh	Mechanical	Yes	N/A	A
Module Protection System							
None	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
None	Power Isolation, Conversion and Monitoring Devices	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
None	ELVS Voltage Sensors	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
None	Under-the-Bioshield Temperature Sensors	RXB - Top of the Module	Harsh	Electrical	Yes	Yes	A

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[(1)]
MPS-BKR-1S-0002A MPS-BKR-1S-0002B MPS-BKR-2S-0002A MPS-BKR-2S-0002B	PZR Heater Trip Breakers (4 Total)	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
MPS-BKR-1S-0001A MPS-BKR-1S-0001B MPS-BKR-2S-0001A MPS-BKR-2S-0001B	Reactor Trip Breakers (4 Total)	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
MPS-HS-AS-TB01 through TB15 MPS-HS-BS-TB01 through TB15 MPS-HS-CS-TB01 through TB15 MPS-HS-DS-TB01 through TB15	Safety Function Module Trip/Bypass Switches (60 Total)	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
MPS-HS-1S-0001 MPS-HS-2S-0001	Enable Nonsafety Control Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
MPS-HS-1S-0002 MPS-HS-2S-0002	MCR Isolation Switches (2 Total)	RXB - 75'	Harsh	Electrical	Yes	Yes	B
MPS-HS-1S-MA08 MPS-HS-2S-MA08	Manual PZR Heater Breaker Trip Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
MPS-HS-1S-MA07 MPS-HS-2S-MA07	Manual LTOP Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
MPS-HS-1S-MA06 MPS-HS-2S-MA06	Manual ECCS Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
MPS-HS-1S-MA05 MPS-HS-2S-MA05	Manual DWSI Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
MPS-HS-1S-MA04 MPS-HS-2S-MA04	Manual DHRS Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E

Table 14.3-3h: Module Specific Mechanical and Electrical/I&C Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
MPS-HS-1S-MA03 MPS-HS-2S-MA03	Manual CVCSI Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
MPS-HS-1S-MA02 MPS-HS-2S-MA02	Manual CSI Actuation Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
MPS-HS-1S-MA01 MPS-HS-2S-MA01	Manual Reactor Trip Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
MPS-HS-1S-0004 MPS-HS-2S-0004	Override Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
MPS-HS-1S-0003 MPS-HS-2S-0003	Operating Bypass Switches (2 Total)	CRB - 76.5'	Mild	Electrical	Yes	Yes	E
Neutron Monitoring System							
None	Excore Neutron Detectors	RXB - Pool	Harsh	Electrical	Yes	Yes	A
None	Excore Signal Conditioning and Processing Equipment	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
None	Excore Power Isolation, Conversion and Monitoring Devices	RXB - 75' RXB - 86'	Mild	Electrical	Yes	Yes	E
In-Core Instrumentation System							
None	In-core instrument string / temperature and flux sensors	RXB - Inside Containment	Harsh	Electrical	Yes	No	A
None	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.

2. This table provides the specific equipment identifiers for the components listed in Tier 1 Table 2.8-1.

Table 14.3-4a: Control Room Habitability System Mechanical Equipment

Equipment Name	Equipment Identifier	Failure Position
Air supply isolation solenoid valves (2 Total)	00-CRH-SV-0007A 00-CRH-SV-0007B	Open
CRE pressure relief isolation valves (2 Total)	00-CRH-SV-0028A 00-CRH-SV-0028B	Open

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 3.1-1.

Table 14.3-4b: Normal Control Room Heating Ventilation and Air Conditioning System Mechanical Equipment

Equipment Name	Equipment Identifier	Actuator Type
CRE isolation dampers (8 Total)	None	Pneumatic

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 3.2-1.

Table 14.3-4c: 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	Equipment Identifier	ASME Code Section III Class
UHS make-up line isolation valve	HV-0001	3

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 3.6-1.

Table 14.3-4d: Radiation Monitoring - NuScale Power Modules 1-12 Automatic Actions

Radiation Monitor ID(s)	Variable Monitored	Actuated Component(s)	Component ID(s)	Component Action(s)
00-CRV-RE-1003 00-CRV-RE-1004 00-CRV-RE-1005	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. None 2. None 3. None 4. None 5. None	1. Close 2. Close 3. Open 4. Open 5. Start
00-CRV-RE-1010 00-CRV-RE-1011	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. None 2. None 3. None 4. None 5. None 6. None 7. None 8. None 9. None 10. None 11. None 12. 00-CRH-SV-0007A 13. 00-CRH-SV-0007B 14. 00-CRH-SV-0028A 15. 00-CRH-SV-0028B	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
00-RBV-RE-1010 00-RBV-RE-1011 00-RBV-RE-1012	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. None 2. None 3. None 4. None 5. None 6. None 7. None 8. 00-RBV-AHU-0004A 9. 00-RBV-AHU-0004B 10. 00-RBV-AHU-0004C 11. 00-RBV-AHU-0004D	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

Table 14.3-4d: Radiation Monitoring - NuScale Power Modules 1-12 Automatic Actions (Continued)

Radiation Monitor ID(s)	Variable Monitored	Actuated Component(s)	Component ID(s)	Component Action(s)
00-GRW-RIT-1021A	GRWS train A charcoal decay bed discharge	1. GRWS train A charcoal bed effluent isolation valve	1. 00-GRW-AOV-0019A	1. Close
00-GRW-RIT-1021B	GRWS train B charcoal decay bed discharge	1. GRWS train B charcoal bed effluent isolation valve	1. 00-GRW-AOV-0019B	1. Close
00-GRW-RIT-1026	GRWS effluent to RBVS	1. GRWS common charcoal bed effluent isolation valve 2. GRWS common charcoal bed effluent isolation valve	1. 00-GRW-AOV-0023 2. 00-GRW-AOV-0024	1. Close 2. Close
00-LRW-RIT-1021 00-LRW-RIT-1022	LRWS discharge to utility water system (UWS)	1. LRWS to UWS isolation valve 2. LRWS to UWS isolation valve	1. 00-LRW-AOV-0059 2. 00-LRW-AOV-0060	1. Close 2. Close
00-AB-RIT-1017	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. 00-AB-AOV-0203 2. 00-AB-AOV-0009A 3. 00-AB-AOV-0009B	1. Close 2. Close 3. Close
00-AB-RIT-1008	ABS high pressure to low pressure steam supply	1. ABS high pressure to low pressure steam supply pressure control valve	1. 00-AB-AOV-0024	1. Close
00-PSC-RE-1003	PSCS tank vent	1. PSCS tank inlet isolation valve 2. PSCS tank outlet isolation valve	1. 00-PSC-MOV-0006 2. 00-PSC-MOV-0009	1. Close 2. Close

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 3.9-1.

Table 14.3-4e: Mechanical and Electrical/Instrumentation and Controls Shared Equipment

Equipment Identifier	Description	Location	EQ Environment	EQ Program	Seismic Category	Class 1E	EQ Category[(1)]
Module Assembly Equipment - Bolting							
00-MAEB-MHE-0001	RPV Support Stand	RXB - UHS	N/A	N/A	I	N/A	N/A
Fuel Handling Equipment							
00-FHE-CRN-0003	Fuel handling machine (FHM)	RXB 100'-0" Elevation	Harsh	Electrical Mechanical	I	No	B
Spent Fuel Storage System							
00-SFS-SF-0001 00-SFS-SF-0002 00-SFS-SF-0003 00-SFS-SF-0004 00-SFS-SF-0005 00-SFS-SF-0006 00-SFS-SF-0007 00-SFS-SF-0008 00-SFS-SF-0009 00-SFS-SF-00010 00-SFS-SF-00011 00-SFS-SF-00012 00-SFS-SF-00013 00-SFS-SF-00014	Fuel Storage Racks (14 Total)	RXB - Spent Fuel Pool	N/A	N/A	I	N/A	N/A
Ultimate Heat Sink							
00-UHS-LI-0101A 00-UHS-LIT-0101A 00-UHS-LI-0101B 00-UHS-LIT-0101B 00-UHS-LI-0102A 00-UHS-LIT-0102A 00-UHS-LI-0102B 00-UHS-LIT-0102B	Pool level instruments (8 total)	RXB - UHS	Harsh	Electrical	I	No	A
00-UHS-0001-BBCX-N	Water Makeup Line	RXB - UHS	N/A	N/A	I	N/A	N/A

Table 14.3-4e: Mechanical and Electrical/Instrumentation and Controls Shared Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	EQ Program	Seismic Category	Class 1E	EQ Category(1)
Reactor Building Cranes							
00-RBC-CRN-0001	Reactor Building crane	RXB 100′-0″′ thru 145′-6″ Elevation	Harsh	Electrical Mechanical	I	No	B
00-RBC-MHE-0001	Module Lifting Adapter	RXB - Various	N/A	N/A	I	N/A	N/A
Reactor Building Components							
None	UHS Pool Liner and Dry Dock Liner	RXB - UHS	N/A	N/A	I	N/A	N/A
00-RBCM-RPD-0001A 00-RBCM-RPD-0001B 00-RBCM-RPD-0002a to 0002h 01 to 12-RBCM-RPD-0003 0A-RBCM-RPD-0004 0B-RBCM-RPD-0004 0A-RBCM-PW-0005A to 0005E 0B-RBCM-PW-0005A to 0005E	RXB over-pressurization vents (34 Total)	RXB	N/A	N/A	I	N/A	A
Liquid Radioactive Waste System							
00-LRW-DGS-0006A 00-LRW-DGS-0006B	Degasifiers (2 Total)	RXB	N/A	N/A	RW-IIa	N/A	N/A
None	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

Table 14.3-4e: Mechanical and Electrical/Instrumentation and Controls Shared Equipment (Continued)

Equipment Identifier	Description	Location	EQ Environment	EQ Program	Seismic Category	Class 1E	EQ Category[1]
Gaseous Radioactive Waste System							
00-GRW-TNK-0012	Charcoal Guard Bed	RWB	N/A	N/A	RW-IIa	N/A	N/A
None	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
00-GRW-TNK-0015A 00-GRW-TNK-0016A 00-GRW-TNK-0017A 00-GRW-TNK-0018A 00-GRW-TNK-0015B 00-GRW-TNK-0016B 00-GRW-TNK-0017B 00-GRW-TNK-0018B	Charcoal Decay Beds (8 Total)	RWB	N/A	N/A	RW-IIa	N/A	N/A
None	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.

2. This table provides the specific equipment identifiers for the components listed in Tier 1 Table 3.14-1.

Table 14.3-4f: Radiation Monitoring - Automatic Actions for NuScale Power Modules 1 - 6

Radiation Monitor ID(s)	Variable Monitored	Actuated Component(s)	Component ID(s)	Component Action(s)
0A-CFD-RT-1007	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. 0A-CFD-AOV-0017	1. Close
0A-BPD-RIT-1010	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. 0A-BPD-P-0027A 2. 0A-BPD-P-0027B 3. 0A-BPD-AOV-0030 4. 0A-BPD-AOV-0051	1. Stop 2. Stop 3. Close 4. Close
0A-BPD-RIT-1001	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. 0A-BPD-P-0002A 2. 0A-BPD-P-0002B 3. 0A-BPD-AOV-0009 4. 0A-BPD-AOV-0050	1. Stop 2. Stop 3. Close 4. Close
00-BPD-RIT-1034	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. 0A-BPD-P-0002A 2. 0A-BPD-P-0002B 3. 0A-BPD-AOV-0009 4. 0A-BPD-AOV-0050	1. Stop 2. Stop 3. Close 4. Close

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 3.17-1.

Table 14.3-4g: Radiation Monitoring - Automatic Actions For NuScale Power Modules 7 - 12

Radiation Monitor ID(s)	Variable Monitored	Actuated Component(s)	Component ID(s)	Component Action(s)
0B-CFD-RT-1007	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. 0B-CFD-AOV-0017	1. Close
0B-BPD-RIT-1010	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. 0B-BPD-P-0027A 2. 0B-BPD-P-0027B 3. 0B-BPD-AOV-0051 4. 0B-BPD-AOV-0030	1. Stop 2. Stop 3. Close 4. Close
0B-BPD-RIT-1001	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. 0B-BPD-P-0002A 2. 0B-BPD-P-0002B 3. 0B-BPD-AOV-0009 4. 0B-BPD-AOV-0050	1. Stop 2. Stop 3. Close 4. Close

Note: This table provides the specific equipment identifiers for the components listed in Tier 1 Table 3.18-1.