RAIO-0418-59506

April 11, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 368 (eRAI No. 9242) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 368 (eRAI No. 9242)," dated February 14, 2018

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Questions from NRC eRAI No. 9242:

- 04.06-1
- 04.06-2

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Samuel Lee, NRC, OWFN-8G9A
Prosanta Chowdhury NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9242

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9242

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9242
Date of RAI Issue: 02/14/2018

NRC Question No.: 04.06-1

10 CFR 50, Appendix A, General Design Criterion (GDC) 4 requires that structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions during normal plant operation as well as during postulated accidents. 10 CFR 52.47 requires the information submitted for a design certification to include performance requirements and design information sufficiently detailed to permit procurement specifications and construction and installation specifications by an applicant. In addition, the Standard Review Plan (SRP), Revision 2, Section 4.6 provides guidance regarding the control rod drive cooling system (CRDS); specifically, Review Procedure 3 directs the reviewer to examine descriptions and drawings to confirm that the systems meet the design requirements, and specifies that the CRDS cooling system should be capable of maintaining the CRDS temperature below the applicant's maximum temperature criterion.

Final Safety Analysis Report (FSAR) Tier 2, Section 4.6.1 states the electric coil operating conditions of the CRDS requires active cooling by water through a CRDS cooling water distribution header to cooling tubes in the drive coils of each control rod drive mechanism (CRDM). Section 4.6.1 adds that the cooling requirements for the CRDMs are provided by the reactor component cooling water system (RCCWS) in Section 9.2.2. The staff reviewed FSAR Tier 2, Section 4.6 and Section 9.2.2, but could not find the cooling requirements for the CRDS.

The applicant is requested to provide in the FSAR, maximum temperature criterion for the CRDM, and RCCWS cooling water temperature values and flow rates required to maintain adequate CRDM cooling for normal operation.

NuScale Response:

The Control Rod Drive Mechanism (CRDM) coils are designed using a Class N insulation system, which is rated to 392 degrees Fahrenheit. In order to provide margin, the Reactor Component Cooling Water System (RCCWS) is designed to limit coil temperatures consistent with those established for one insulation class lower, or 356 degrees Fahrenheit. Therefore, the maximum temperature design criterion for the CRDM is 356 degrees Fahrenheit.

The CRDM RCCWS cooling water parameters were previously provided in the original (RAIO-1117-57110) and supplemental (RAIO-1217-57637) responses to RAI 09.02.02-4 (eRAI 9101). These letters were transmitted to the NRC on November 10, 2017 and December 12, 2017. The CRDM RCCWS parameters are as follows:

<u>Normal Operation:</u>

Flow Rate for Each CRDM = 2 gpm

Heat Load for Each CRDM = 40,200 Btu/hr

RCCWS Temperature in = 80 degrees Fahrenheit

RCCWS Temperature out = 120.5 degrees Fahrenheit

<u>Sizing Basis:</u>

Flow Rate for Each CRDM = 2 gpm

Heat Load for Each CRDM = 40,200 BTU/hr

RCCWS Temperature in = 100 degrees Fahrenheit

RCCWS Temperature out = 140.5 degrees Fahrenheit

These values are based on preliminary hardware designs and may change when site-specific conditions are incorporated and final detailed designs are complete. For this reason and also because the safety function of the CRDM, which is to insert upon a reactor trip, is not affected by the loss of CRDM cooling, these preliminary values for cooling parameters are not added to the FSAR.

Impact on DCA:

FSAR Section 4.6.1 has been been revised as described in the response above and as shown in the markup provided in this response.

electromagnetic coils and housings, including the pressure housings. The major components of the CRDM are annotated, and detailed in the subsequent figures. The power and cooling water connectors are located on top of the mast assembly and sensor coil for ease of access through the removable cover on top of the CNV (Figure 4.6-1). Figure 4.6-3 illustrates the CRDM drive coil and embedded cooling coils shown on the right view without the coil stack housings and mast assembly. The electrical connector on top of the left view is located above the cooling water fittings for separation purposes. Figure 4.6-4 shows the layout of the rod position indicator sensor coil assemblies which are located directly above the rod travel housing. Rod position indication is facilitated by means of electromagnetic induction in the sensor coils, as the top of the control rod drive shaft travels upwards or downwards within the pressure boundary. Figure 4.6-5 provides an overview of the latch mechanism assembly (LMA), with the remote disconnect latch shown separately for better illustration. The three magnetic poles, latches and grippers on the left represent an industry-standard LMA design that performs the rod withdrawal/insertion/reactor trip functions, whereas the remote disconnect grippers (RDG) are relied upon during the remote disconnection/re-connection for NPM refueling only. Figure 4.6-6 illustrates the remote disconnection of the control rod drive shaft from the CRA that is not available in the operating NPM location, in order to preclude inadvertent CRA disengagement.

The CRDM assembly is a hermetically sealed electro-mechanical device, which moves the CRA in and out of the reactor core, and holds the CRA at any elevation within the range of CRA travel. If electrical power is interrupted to the CRDM, the control rod drive shaft is released, and the attached CRA drops into the reactor core.

RAI 04.06-1

The CRDMs are mounted on the RPV head, and the CRDM pressure housings are safety-related American Society of Mechanical Engineers (ASME) Class 1 pressure boundaries. The CRDS components internal to the reactor coolant pressure boundary are designed to function in borated primary coolant with up to 2000 ppm boron at primary coolant pressures and temperatures ranging from ambient conditions to 650 degrees F design temperature and 2,100 psia RPV design pressure. During normal operating conditions the upper portion of the RPV and the CRDM pressure housing are in contact with saturated steam on the inside at 625 degrees F and 1850 psia. The lower portion of the drive rod is submerged in the primary coolant at hot leg temperature flowing upward through the upper riser and CRA guide tubes. The electric coil operating conditions require active cooling by water through a CRDS cooling water distribution header to cooling tubes in the drive coils of each CRDM as shown in Figure 4.6-3. The cooling requirements for the CRDMs are provided by the reactor component cooling water system (RCCWS) in Section 9.2.2. The RCCWS is designed to maintain the CRDM winding temperature below the design maximum temperature of 356 degrees Fahrenheit.

The CRDS cooling line is branched into supply lines inside the containment vessel to each individual CRDM. After passing through the CRDM cooling tubes, the flexible return lines rejoin into a single return header leaving containment. A thermal relief valve is provided on the return header to provide overpressure protection for the CRDS cooling piping during a containment isolation event.

The structural materials of construction for the CRDS are discussed in detail in Section 4.5.1.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9242
Date of RAI Issue: 02/14/2018

NRC Question No.: 04.06-2

GDC 26, "Reactivity Control System Redundancy and Capability," requires two independent reactivity control systems of different design principles that are capable of reliably controlling reactivity changes during normal operation. GDC 29 requires that protection and reactivity control systems be designed to ensure an extremely high probability of functioning in the event of an anticipated operational occurrence. The CRDS is one of those systems, and the areas of review under SRP Section 4.6, "Functional Design of Control Rod Drive System," include functional tests for the CRDS.

Regulatory Guide 1.68, Revision 4, and Design Specific Review Standard (DSRS) Section 14.2 provide guidance for testing of the CRDS. DSRS Section 14.2 states that the applicant should provide test abstracts of SSCs and unique design features, including tests and acceptance criteria.

FSAR Tier 2, Section 14.2 provides the elements of the Initial Test Program (ITP). The information provided regarding these tests are not sufficient to ensure adequacy of the results of the ITP, specifically:

a. FSAR Tier 2, Table 14.2-80 (Test#80), does not include specific acceptance criteria for CRDS performance. No numerical values are specified for ITP rod insertion and withdrawal speeds, the limit for control rod assembly (CRA) position indications within the associated group position or control rod demand position, and the CRA fully withdrawn position. The applicant is requested to provide the design limits within the acceptance criteria of Test #80 or reference a location in the FSAR that provides the values for the design limits.

b. Acceptance criteria 'i' of FSAR Tier 2, Table 14.2-81 (Test #81), does not provide specific acceptance values for drop time. The applicant is requested to provide the drop time within the acceptance criteria of Test #81 or reference a location in the FSAR that provides the values for drop time. In addition, the test should clearly indicate that the drop test involves a "full-height" drop.

c. Acceptance criteria ‘ii' of FSAR Tier 2, Table 14.2-81 (Test #81), specifies the “arithmetic average of all CRA drop times are within TS limits.” However, RG 1.68 and technical specification (TS) surveillance requirement (SR) 3.1.4.3 do not permit the use of arithmetic averages to satisfy CRA drop time testing. RG 1.68 also adds, “those control rods for which the scram times fall outside the two sigma limit of the scram time data for all control rods should be retested a sufficient number of times (e.g., three times) to reasonably ensure proper performance during subsequent plant operations.” Therefore, the applicant is requested to revise Test #81 acceptance criteria ‘ii’ to ensure each CRA drop time is within specified limits and all control rod drop times fall within the two sigma limit per RG 1.68. In addition, the applicant is requested to revise Test #81 acceptance criteria ‘ii’ to provide the specific drop time values or surveillance requirement(s) needed to ensure proper operation of the CRDS.

d. RG 1.68 specifies, “to the extent practical, testing should demonstrate control rod scram times at both hot zero power and cold temperature conditions, with flow and no flow conditions in the reactor coolant system as required to bound conditions under which scram might be required.”

   i. FSAR Tier 2, Table 14.2-81 (Test #81) specifies that the CRA drop time testing is performed when the reactor coolant system (RCS) is at hot zero power (HZP). However, the staff could not find a test of CRA drop times during cold temperature conditions. Therefore, the applicant is requested to provide a CRA drop time test for cold temperature conditions, or provide justification for how testing during only HZP is bounding and demonstrates the control rods will drop within the required time under cold temperature plant conditions.

   ii. FSAR Tier 2, Table 14.2-81 (Test #81), specifies that the CRA drop time testing is performed when the RCS is at HZP. However, the staff could not find a test of CRA drop times during flow conditions. Therefore, the applicant is requested to add CRA drop time test acceptance criteria to the FSAR Tier 2, Table 14.2-107 (Test #107) and Table 14.2-104 (Test #104), which involve reactor trips at 10-20% reactor thermal power and 100% reactor thermal power, respectively.

RG 1.68 Appendix A, A-5 Power Ascension Tests, item (g), specifies, “Check rod scram times from data recorded during scrams that occur during the startup test phase to determine that the scram times remain within allowable limits.” However, the staff could not find this item anywhere in FSAR 14.2, ITP. The applicant is requested to include item (g) in either FSAR 14.2.4 or the Startup Administration Manual (COL Item 14.2-2). If it is to be addressed in the Startup Administrative Manual it can be specified separately. Alternatively, the COL information item could specify that the Startup Administrative Manual will meet RG 1.68 guidelines.

NuScale Response:

Part a. response

Tier 2, Table 14.2-80 (Test #80) is a preoperational test abstract for the Control Rod Drive System (CRDS). As described in Sections 14.2.10 and 14.2.12, test abstracts provide the bases for detailed preoperational and startup test procedures. Detailed preoperational and startup test procedures are developed and submitted to the NRC by Combined License (COL) holders by no later than 60 days prior to the conduct of preoperational and startup testing. While test abstracts provide the acceptance criteria for satisfying the test objectives, the design detail at the time of test abstract development is insufficient to include numerical values for certain parameters.

Rod insertion and withdrawal rates meet design requirements as described in Section 3.9.4.1. Verification that individual CRA position agrees with control rod demand position is not applicable. Therefore, the acceptance criterion to perform this verification was deleted (previously Test #80, Acceptance Criterion ii).The limit associated with individual CRA position indications within the associated group position is found in Technical Specifications as indicated by Test #80, Acceptance Criterion iii. The CRA fully withdrawn position is an unambiguous description that has no numerical value in its design details. References to specific numerical values to be used for the specific acceptance criteria will be provided by the COL holder during development of detailed test procedures.

Part b. response

Tier 2, Table 14.2-81 (Test #81) is a preoperational test abstract for the CRA. As described in Sections 14.2.10 and 14.2.12, test abstracts provide the bases for detailed preoperational and startup test procedures. Detailed preoperational and startup test procedures are developed and submitted to the NRC by COL holders by no later than 60 days prior to the conduct of preoperational and startup testing. While test abstracts provide acceptance criteria for satisfying the test objectives, the design detail at the time of test abstract development is insufficient to include numerical values for certain parameters.

Rod drop times are consistent with bounding safety analysis (in Section 15.0) and in agreement with applicable Technical Specifications. Table 14.2-81 will be modified to include that the drop test involves a "full height" drop.

Part c. response

Tier 2, Table 14.2-81 (Test #81) acceptance criterion ii. has been modified to remove the statement about "arithmetic average" and include the two sigma limit. Acceptance criterion ii. states that "Each CRA drop time is within two sigma of the drop time data for all control rods, or has been verified within Technical Specification limits by a minimum of three additional performances of this test."

Part d. response

- **Subpart i. response**

  NuScale has added a new Tier 2 test abstract, Table 14.2-81A, Control Rod Assembly Ambient Temperature Full-Height Drop Time Test (Test #81A), to perform drop time testing at ambient temperature conditions, consistent with the Test Method and acceptance criteria of Test #81 at HZP.

- **Subpart ii. response**

  As requested, NuScale has added an analysis of rod drop times to the Tier 2, Table 14.2-104, Reactor Trip from 100 Percent Power Test (Test #104). The analysis of rod drop times at 10-20% thermal power has been added to Tier 2, Table 14.2-107 Remote Shutdown Workstation Test (Test #107).

Last paragraph response

During startup testing, NuScale has committed to perform rod drop time testing at ambient conditions (Test #81A), HZP (Test #81), 10-20% thermal power (Test #107), and 100% thermal power (Test #104). Technical specifications dictate the requirements and frequency for rod drop time testing. No additional wording was added to FSAR Section 14.2.4 or COL Item 14.2-2.

Impact on DCA:

New FSAR Table 14.2-81A has been added and Tables 14.2-75, 14.2-80, 14.2-81, 14.2-104 and 14.2-107 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 04.06-2

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

Startup test is required to be performed for each NPM.
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 (Test #77) b. NuScale Power Module Temperatures Test (Test #78) c. Primary and Secondary System Chemistry Test (Test #79) d. Control Rod Drive System - Manual Operation, Rod Speed, and Rod Position Indication Test (Test #80) e. Control Rod Assembly ~~(CRA)~~Full-Height Drop Time Test (Test #81) f. Control Rod Assembly Ambient Temperature Full-Height Drop Time Test (Test #81A) ~~f~~g. Pressurizer Spray Bypass Flow Test (Test #82)
Prerequisites
None
Test Method
i. Identify the specific plant conditions required for each precritical test procedure to maintain technical specification operability. ii. Identify the prerequisites required for each precritical test procedure. iii. Determine the test sequence for precritical testing based on technical specification requirements and test prerequisites.
Acceptance Criterion
The sequence for precritical testing has been determined.

RAI 04.06-2

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

Startup test is required to be performed for each 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 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 RCS is at hot zero power (RCS at normal operating pressure and RCS temperature at the maximum temperature obtainable when heated only by the MHS). iv. All RCS temperatures satisfy the minimum technical specification temperature for criticality. v. The nuclear instrumentation system is calibrated and operable. vi. The SDM 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 positions agree with the control rod demand position within design limits for the full range of CRA travel.~~ ~~iii~~ii. Individual CRA position indications are within the number of steps of their associated group position as required by TS. ~~iv~~iii.The CRA insertion and withdrawal speeds are within the design limits identified in Section 3.9.4.1.

RAI 04.06-2

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

Startup test is required to be performed for each NPM.
This test is performed after initial fuel loading but prior to initial criticality.
Test Objective
Verify each CRA satisfies the CRA drop time acceptance criteria for RCS flow at 0% 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 MHS). iv. All RCS temperatures satisfy the minimum technical specification temperature for criticality. v. The nuclear instrumentation system is calibrated and operable. vi. The SDM is within the limits specified in the core operating limits report. ~~vii. A CRA drop time acceptance criteria for 0% thermal reactor power has been developed and is in agreement with the technical specification CRA drop time surveillance requirement.~~
Test Method
i. Fully ~~W~~withdraw each individual CRA. ii. Interrupt the electrical power to the associated CRDM. iii. Measure the CRA drop time.
Acceptance Criteria
i. Each CRA drop time is ~~less than or equal to the CRA drop time acceptance criteria for HZP~~within Technical Specification limits. ii. ~~The arithmetic average of all CRA drop times is within TS limits~~Each CRA drop time is within two sigma of the drop time data for all control rods, or has been verified within Technical Specification limits by a minimum of three additional performances of this test.

RAI 04.06-2

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

Startup test is required to be performed for each NPM.
This test is performed after initial fuel loading but prior to initial criticality.
Test Objective
Verify each CRA satisfies the CRA drop time acceptance criteria for RCS at ambient temperature.
Prerequisites
i. The core is installed. ii. The NPM is fully assembled. iii. The RCS is at ambient temperature. iv. The nuclear instrumentation system is calibrated and operable. v. The SDM 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 CRDM. iii. Measure the CRA drop time.
Acceptance Criteria
i. Each CRA drop time is within Technical Specification limits. ii. Each CRA drop time is within two sigma of the drop time data for all control rods, or has been verified within Technical Specification limits by a minimum of three additional performances of this test.

RAI 04.06-2

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

Startup test is required to be performed for each NPM.
This test is performed at 100 percent reactor thermal power.
Test Objectives
i. Verify the ability of the NPM to sustain a reactor trip from 100% reactor thermal power and automatically cool the RCS to mode 3 (all RCS temperatures < 420 °F). ii. Assess the dynamic response of the plant to the reactor trip. iii. Verify each fully withdrawn CRA satisfies the CRA drop time acceptance criteria at full flow conditions.
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 MCR. ii. Measure the drop time for each fully withdrawn CRA. iii. Allow the RCS to cool to mode 3.
Acceptance ~~Criterion~~Criteria
i. The reactor trips. ii. The CIVs close. iii. The decay heat removal valves open. iv. The turbine generator bypass valve operates to prevent opening of the main steam safety valve. v. The turbine speed does not exceed overspeed design limits. vi. The reactor vent valves do not open. 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 reactor trip viii. The RCS cools to a stable condition in mode 3 without operator intervention. ix. Each fully withdrawn CRA drop time is within Technical Specification limits.

RAI 04.06-2

Table 14.2-107: Remote Shutdown Workstation Test (Test #107)

Startup test is required to be performed for each NPM.
This test is performed at approximately 10 - 20 percent reactor thermal power.
Test Objectives
i. Verify the NPM safety-related controls can be disabled at the remote shutdown station. ii. Verify the NPM nonsafety-related controls are functional at the remote shutdown station. III. Verify each fully withdrawn CRA satisfies the CRA drop time acceptance criteria with the reactor operating at 10 - 20% reactor thermal power.
Prerequisites
i. Communication exists between the MCR and the remote shutdown station. ii. The reactor is operating in a steady-state condition at 10 - 20% reactor thermal power.
Test Method
i. Using the appropriate operating procedure, the operator manually trips the reactor under test before leaving the MCR. ii. Measure the drop time for each fully withdrawn CRA. iii. Using the appropriate operating procedure, the operator uses manual switches in the remote shutdown station to isolate the module protection system manual actuation switches, override switches, and the enable nonsafety control switches for each nuclear power modules' module protection system in the MCR to prevent spurious actuation of equipment due to fire damage.
Acceptance Criteria
i. An operator verifies that the module protection switch controls in the MCR have been disabled. The displays in the remote shutdown station verify the following NPM status: ii. The reactor is tripped. iii. All CIVs are closed. iv. The DHRS actuation valves are open. v. All RCS temperatures cool to less than 420°F (mode 3, safe shutdown) without operator action. vi. Safety-related components cannot be operated from the remote shutdown station. vii. The nonsafety-related controls in the remote shutdown station controls can be used to place the plant in a configuration specified by the appropriate operating procedure. viii. Each fully withdrawn CRA drop time is within Technical Specification limits.

RAI 04.06-2

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	CHW	Chilled Water System
9	ABS	Auxiliary Boiler System
10	CWS	Circulating Water System
11	SCW	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 Ventilation
23	RWDS	Radioactive Waste Drain System
24	BPDS	Balance-of-Plant Drains
25	FPS	Fire Protection System
26	FDS	Fire Detection
27	MSS	Main Steam
28	CFWS	Feedwater System
29	FWTS	Feedwater Treatment
30	CPS	Condensate Polisher Resin Regeneration System
31	HVD	Heater Vents and Drains
32	CARS	Condenser Air Removal System
33	TGS	Turbine Generator
34	TLOS	Turbine Lube Oil 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	CRDS	Control Rod Drive System Flow-Induced Vibration
45	RVI	Reactor Vessel Internals Flow-Induced Vibration
46	RCS	Reactor Coolant System
47	ECCS	Emergency Core Cooling System
48	DHRS	Decay Heat Removal System
49	ICIS	In-core Instrumentation

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

Test Number	System Abbreviation	Test Abstract
50	MAE	Module Assembly Equipment
51	FHE	Fuel Handling Equipment System
52	RBC	Reactor Building Cranes
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
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
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