



July 25, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission  
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Rockville, MD 20852-2738

**SUBJECT:** NuScale Power, LLC Response to NRC Request for Additional Information No. 194 (eRAI No. 8884) on the NuScale Design Certification Application

**REFERENCES:** 1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 194 (eRAI No. 8884)," dated August 21, 2017  
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 205 (eRAI No.8884)," dated October 18, 2017  
3. NuScale Power, LLC Response to NRC "Request for Additional Information No. 205 (eRAI No.8884)," dated January 3, 2019

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. 8884:

- 03.09.02-9
- 03.09.02-10

A majority of the responses to RAI No. 205, eRAI No. 8884, questions were previously provided in References 2 and 3. This completes all responses to eRAI 8884.

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 Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad  
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RAIO-0719-66435

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8884



**Enclosure 1:**

NuScale Response to NRC Request for Additional Information eRAI No. 8884

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

**eRAI No.:** 8884

**Date of RAI Issue:** 08/21/2017

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**NRC Question No.:** 03.09.02-9

10 CFR 50, Appendix A, GDC 4 requires structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Regulatory guides (RG) describe methods that the NRC considers acceptable to use in implementing the agency's regulations. RG 1.20, Rev. 3 states that the vibration measurement program should include description of instrument types and locations. In section 4.2 (Lead Unit Factory Testing) of CVAP TR-0716-50439 only a brief summary of the planned lead unit factory testing is provided. Without the detailed description of the testing plan, the NRC staff cannot reach a safety finding.

Provide a detailed test plan for the lead unit factory testing. Include instrument types and locations and pre-test predictions of the expected and allowable experimental results, considering bias errors and random uncertainties. Update the CVAP technical report to include the requested information.

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### **NuScale Response:**

Lead unit factory testing has been eliminated for both the control rod drive (CRD) shaft and the in-core instrument guide tube (ICIGT).

The CRD shaft sleeve components have been added to the design to shield the CRD shaft from cross flow in the upper riser, consequently, the CRD shaft is no longer susceptible to vortex shedding. The CRD shaft sleeve is susceptible to vortex shedding, but due to the high natural frequency of the component, the safety margin to a vortex shedding lock-in condition is greater



than 100%. See Sections 2.3.3.4, 2.3.3.7, and 3.2.2 of the Comprehensive Vibration Assessment Program (CVAP) technical report TR-0716-50439.

The ICIGT is now welded at the first support it passes through in the upper riser, which has increased the frequency of the component and provides a safety margin greater than 100% for the most limiting modal response, as documented in Section 3.2.2 of the CVAP TR-0716-50439.

The Control Rod Drive System Flow-Induced Vibration Test # 44 and Reactor Vessel Internals Flow-Induced Vibration Test # 45 in FSAR Chapter 14 have been deleted.

**Impact on DCA:**

The CVAP Analysis Technical Report TR-0716-50439 Sections 2.3.3.4, 2.3.3.7, and 3.2.2 will be revised as described in the response above. The technical report is being submitted separately. The FSAR Section 14 has been revised as described in the response and as shown in the markup provided with the response to question 03.09.02-10.

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

**eRAI No.:** 8884

**Date of RAI Issue:** 08/21/2017

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**NRC Question No.:** 03.09.02-10

10 CFR 50, Appendix A, GDC 4 requires structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. In section 4.3 (Lead Unit Initial Startup Testing) of CVAP TR-0716-50439 only a brief summary of the planned lead unit initial startup testing is provided. Also, no instrumentation is defined for this testing. Without the detailed description of the testing plan, the NRC staff cannot reach a safety finding.

Provide the CVAP Measurement Program Test Plan, including initial startup test operating conditions, test durations, instrument types and locations, applicable testing hold points, and pre-test predictions of the expected and allowable experimental results, considering bias errors and random uncertainties (B/U). Note that prototype reactor internals that may be subject to significant FIV necessitate instrumentation. In particular, provide instrumentation plans that will be used to measure susceptible SG tube vibration and contact statistics, ICIGT to CRDM support contact statistics, and DHRS steam piping vibration. Ensure that testing will also be performed for 1 million cycles for Emergency Core Cooling and Decay Heat Removal operations. Explain how end-to-end B/U (vibration/strain/pressure) calculated from comparing pre-test predictions to the test results will be applied to the design FIV analysis results and how updates will be made to the margin of safety estimates. If any margins of safety are not met, provide corrective actions and update the future inspection program to monitor components susceptible to structural failures. Provide details on how acceptance criteria will be checked. Update the CVAP technical report to include the requested information.

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## **NuScale Response:**

The initial startup vibration monitoring plan is outlined in Section 6.0 of the CVAP Measurement and Inspection Plan Technical Report, TR-0918-60894. Vibration monitoring is performed on the lead NPM unit in order to detect unexpected, large amplitude FIV during the initial startup test phase. These measurements supplement the scope of the detailed validation measurement program, discussed in other sections of this technical report, by confirming the lack of large amplitude vibration in the prototype NPM. Optimal sensor type and installation details are provided for each region of interest in Section 6.3 of TR-0918-60894. These regions of interest include the steam generator assembly, in-core instrument guide tubes, control rod drive shaft support and shaft sleeve, and the lower to upper riser slip joint. Pre-test predictions for the purpose of CVAP analysis validation, per Regulatory Guide 1.20, are not performed for this vibration monitoring because there are no bias errors for tests performed at normal operating conditions in the installed NPM. Anticipated vibration amplitudes for the FIV mechanisms being tested are defined in Section 6.0 of TR-0918-60894.

Validation testing for acoustic resonance (AR) is also to be performed during initial startup testing of the prototype NPM. The regions to be tested are the branch connections in the containment system (CNTS) main steam line, including the decay heat removal system (DHRS) steam lines, the main steam isolation valve (MSIV) bypass lines, and main steam drain valve branches. These regions have less than 100% margin to the onset of acoustic resonance, as determined in the design analysis. The test plan and pre-test prediction for this testing are provided in Section 5.2 of TR-0918-60894.

The pre-test prediction for AR in the DHRS steam piping assesses input and measurement uncertainties to inform the range of possible test conditions at which an AR condition occurs. The calculation quantifies the expected secondary flow velocity when the onset of AR occurs and the effect of uncertainties on the predicted safety margin.

This test program is NuScale Power Module Vibration Test #108 in FSAR Table 14.2-108. This test is part of the power ascension testing performed on the first NPM after the first fuel load. Due to the natural circulation design of the NPM, it is not possible to obtain the limiting thermal hydraulic conditions that are necessary to verify the FIV inputs and results for acoustic resonance in the CNTS main steam lines, and confirm the lack of unexpected, large amplitude FIV for other instrumented locations until the NPM is operating near full power conditions. This test is performed for a sufficient duration to ensure one million vibration cycles for the component with the lowest structural natural frequency. Vibration monitoring during a DHRS or



ECCS actuation is not included in this test program because the reduced flow rates following one of these events are not limiting conditions for FIV of the tested regions. However, since the vibration monitoring instruments are installed for the duration of the initial startup testing, they will be used to monitor vibration during power ascension testing with engineered safety feature actuations such as Test #104, Reactor Trip from 100 Percent Power, which includes DHRS actuation. The acceptance criteria for Test #108, as outlined in FSAR Table 14.2-108, are met when acoustic resonance analysis margins in Section 3.2.4 of TR-0716-50439 are confirmed, and by visual examination of the module components according to the inspection program outlined in Table 5-1 of TR-0716-50439 and Section 7.0 of TR-0918-60894.

**Impact on DCA:**

The CVAP Measurement and Inspection Plan Technical Report TR-0918-60894 Section 5.2.1 and Section 6.0 will be revised as described in the response above. The technical report will be submitted separately. The FSAR Table 14.2-108 has been revised as described in the response and as shown in the markup provided in this response.



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

RAI 03.07.02-24S1

**Safe Shutdown Earthquake (SSE) Ground Motion** is the site-specific vibratory ground motion for which safety-related SSC are designed to remain functional. The SSE for a site is a smoothed spectra developed to envelop the ground motion response spectra. The SSE is characterized at the free ground surface. A combined license (COL) applicant may use the SSE for design of site-specific SSC.

**System Description (Tier 1)** includes

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

The system description may include system description tables and figures.

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

RAI 03.09.02-10

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

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

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

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

RAI 06.03-751

The ECCS consists of three reactor vent valves (RVVs), two reactor recirculation valves (RRVs), and associated actuators. The RRVs are designed with a minimum flow coefficient of 55 and a maximum flow coefficient of 100. Each RVV and diffuser, as a combined unit, are designed with a minimum flow coefficient of 375 and a maximum flow coefficient of 490. Additionally, the RVVs are designed with a minimum terminal pressure drop ratio of 0.62 and a maximum terminal pressure drop ratio of 0.90.

RAI 03.09.02-10

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

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

- The RCS supports the CNTS by supplying the RCPB and a fission product boundary via the RPV and other appurtenances.
- The CRDS supports the RCS by maintaining the pressure boundary of the RPV.
- The SGS supports the RCS by supplying part of the RCPB.
- The ECCS supports the RCS by providing a portion of the RCPB for maintaining the RCPB integrity.
- The CNTS supports the RXB by providing a barrier to contain mass, energy, and fission product release from a degradation of the RCPB.
- The ECCS supports the CNTS by providing a portion of the containment boundary for maintaining containment integrity.
- The CNTS supports the DHRS by providing the required pressure boundary for DHR operation.
- The RCS supports the SGS by providing physical support for the SG tube supports and for the integral steam and feed plenums.
- The RCS supports the reactor core by the RVI providing mechanical support to orient, position, and seat the fuel assemblies.
- The RCS supports the CRDS by the RPV and the RVI supporting and aligning the control rods.
- The CNTS supports the DHRS by providing structural support for the DHRS piping.

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

RAI 03.09.02-10, RAI 14.02-1

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 is a FOAK program. The program is implemented consistent with the requirements of the NuScale "Comprehensive Vibration Assessment Program (CVAP) 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.

RAI 03.09.02-10, RAI 14.02-1

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

- ~~Table 14.2-44: Control Rod Drive System Flow-Induced Vibration Test #44~~
- ~~Table 14.2-45: Reactor Vessel Internals Flow-Induced Vibration Test #45~~
- Table 14.2-75: Steam Generator Flow-Induced Vibration Test #72.
- Table 14.2-108: NuScale Power Module Vibration Test #108

RAI 05.04.07-7S1, RAI 14.02-1

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](#).

RAI 05.04.07-7S1

[Section 5.4.3.4 contains a description of the 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.](#)

RAI 14.02-1

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.

RAI 14.02-1

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.

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.

RAI 03.09.02-10

Vibration testing ~~that is performed at the factory~~ is performed in accordance to the requirements of the NuScale "Comprehensive Vibration Assessment Program" as described in the "Comprehensive Vibration Assessment Program (CVAP) 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 14.2.1.3.

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.

RAI 03.09.02-10, RAI 14.02-1

**Table 14.2-44: Control Rod Drive System Flow-Induced Vibration Test # 44 Not Used**

|   |                                       |                                    |
|---|---------------------------------------|------------------------------------|
| Validation testing is performed during factory testing on the control rod drive shaft per Table 4-1 of TR-0716-50439. There are no preoperational tests for CRDS.   |                                       |                                    |
| The CRDS flow-induced vibration testing is performed consistent with the requirements of the NuScale “Comprehensive Vibration Assessment Program” as described in Section 5.0 of the “Comprehensive Vibration Assessment Program (CVAP) Technical Report,” TR-0716-50439. Visual examination of the CRDS components is performed as specified in Table 5-1 of TR-0716-50439. This test is coordinated with Test #108. The CVAP is addressed in Section 3.9.2. The CRDS is discussed in Section 4.6. |                                       |                                    |
| <b>System Function</b>  | <b>System Function Categorization</b> | <b>Function Verified by Test #</b> |
| None  | N/A                                   | N/A                                |
| <b>Prerequisites:</b><br>N/A  |                                       |                                    |
| <b>Component Level Tests</b>  |                                       |                                    |
| None  |                                       |                                    |

RAI 03.09.02-10, RAI 14.02-1

**Table 14.2-45: Reactor Vessel Internals Flow-Induced Vibration Test # 45 Not Used**

|  |                                       |                                    |
|--|---------------------------------------|------------------------------------|
| Validation testing is performed at the factory for the in-core instrument guide tubes per Table 4-1 of TR-0716-50439. There are no preoperational tests for RVI.   |                                       |                                    |
| RVI flow induced vibration testing is performed consistent with the requirements of the NuScale "Comprehensive Vibration Assessment Program" as described in Section 5.0 of the "Comprehensive Vibration Assessment Program (CVAP) Technical Report," TR-0716-50439. Visual examination of the RVI components is performed as specified in Table 5-1 of TR-0716-50439. This test is coordinated with Test #108. The CVAP is addressed in Section 3.9.2. Reactor vessel internals are discussed in Section 5.1.3.3. |                                       |                                    |
| <b>System Function</b>   | <b>System Function Categorization</b> | <b>Function Verified by Test #</b> |
| None   | N/A                                   | N/A                                |
| <b>Prerequisites:</b><br>N/A   |                                       |                                    |
| <b>Component Level Tests</b>   |                                       |                                    |
| None   |                                       |                                    |

RAI 03.09.02-10, RAI 14.02-1

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

|  |                                       |                                    |
|--|---------------------------------------|------------------------------------|
| <p><u>This is a one-time test to be performed prior to loading fuel in the first ever NPM. There are no preoperational tests for the SGS.</u></p>  |                                       |                                    |
| <p>Validation testing is performed at test facilities as separate effects tests on prototypic steam generator tube columns and steam generator inlet flow restrictors per Table 4-1 of TR-0716-50439. <del>There are no preoperational tests for SG.</del></p>   |                                       |                                    |
| <p>SG flow-induced vibration testing is performed consistent with the requirements of the NuScale "Comprehensive Vibration Assessment Program" as described in the "Comprehensive Vibration Assessment Program (CVAP) Technical Report," TR-0716-50439, <u>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.</u> Visual examination of the SG components is performed as specified in Table 5-1 of TR-0716-50439. <del>This test is coordinated with Test #109.</del> The CVAP is addressed in Section 3.9.2. The steam generators are discussed in Section 5.4.1.</p>  |                                       |                                    |
| <b>System Function</b>   | <b>System Function Categorization</b> | <b>Function Verified by Test #</b> |
| None   | N/A                                   | N/A                                |
| <b>Prerequisites:</b>  |                                       |                                    |
| N/A  |                                       |                                    |
| <b>Component Level Tests</b>   |                                       |                                    |
| None   |                                       |                                    |
| <b>Acceptance Criteria:</b>  |                                       |                                    |
| <p>i. <u>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.</u></p> <p>ii. <u>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.</u></p> <p>iii. <u>Steam generator 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.</u></p> |                                       |                                    |

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

|   |
|---|
| <p>This startup test is required to be performed once for NPM #1. This test supports FOAK testing described in Section 14.2.3.3.</p>  |
| <p>This test is performed <u>during the load ramp from zero to 100 percent power and at 100 percent reactor thermal power. NuScale Power Module vibration testing is described in Sections 3.9.2.1.1.1, 3.9.2.3, and 3.9.2.4, and Reference 3.9-5 NuScale Power, LLC, "Comprehensive Vibration Assessment Program (CVAP) Technical Report," TR-0716-50439 "NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report," TR-0918-60894. This test is coordinated with Test #100 and Test #104.</u></p>  |
| <p><b>Test Objective for NPM #1</b></p> <ul style="list-style-type: none"> <li>i. Perform vibration testing of <u>CNTS MS line branch connections, including DHRS steam piping, MSIV bypass lines, and MS drain valve branches during the load ramp up to and</u> at 100 percent reactor thermal power <del>as described in TR-0716-50439, Section 4.3,</del> to verify vibration amplitudes in the <u>DHRS steam piping regions confirm the</u><del>there is no</del> acoustic resonance <u>response analysis results described in TR-0716-50439 Section 4.3.</u></li> <li>ii. <u>Perform monitoring of vibration amplitudes at locations on the SG assembly, ICGTs, 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.</u></li> <li>iii. Perform visual examination of the NuScale Power Module components specified in <del>Table 5-1 of TR-0716-50439</del><u>Section 7.0 of TR-0918-60894.</u></li> </ul> |
| <p><b>Prerequisites</b></p> <ul style="list-style-type: none"> <li>i. The DHRS steam piping, <u>MSIV bypass lines, and MS drain valve branches are</u><del>is</del> instrumented to obtain acoustic resonance (AR) data.</li> <li>ii. <u>Selected SG tube, ICGT, CRD shaft support, and riser slip joint locations are instrumented to provide vibration monitoring.</u></li> </ul>   |
| <p><b>Test Method</b></p> <ul style="list-style-type: none"> <li>i. <u>Perform load ramp up to 100 percent power, then</u> <del>o</del> operate the NuScale Power Module for a sufficient duration at 100 percent power to ensure one million vibration cycles for the component with the lowest structural natural frequency.</li> <li>ii. Monitor the vibration of the <u>DHRS 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, ICGT, CRD shaft support, and riser slip joint locations for detection of any unexpected large amplitude vibration responses.</u> 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.</li> <li>iii. Disassemble the NuScale Power Module and performed a visual examination of the module components specified in <u>Section 7.0 of TR-0918-60894, Table 5-1 of TR-0716-50439.</u></li> </ul>  |
| <p><b>Acceptance Criteria</b></p> <ul style="list-style-type: none"> <li>i. Measured vibration amplitudes in the <u>DHRS CNTS steam piping branches</u> confirm the acoustic resonance analysis results described in <u>Section 5.2 of TR-0918-60894, TR-0716-50439 Section 4.3.</u></li> <li>ii. <u>Measured vibration responses in the SG tube, ICGT, 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.</u></li> <li>iii. Visual examination results of module components satisfy the acceptance criteria of <u>Section 7.0 of TR-0918-60894, Table 5-1 of TR-0716-50439.</u></li> </ul>  |



RAI 03.09.02-10, 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           | 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          | CRDSN/A             | Not Used Control Rod Drive System Flow-Induced Vibration |
| 45          | RVIN/A              | Not Used 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 System                           |

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

| Test Number | System Abbreviation | Test Abstract                                       |
|-------------|---------------------|---|
| 99          | N/A                 | Steam Generator Level Control                       |
| 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                 | <del>Not Used Remote Shutdown Workstation</del>     |
| 108         | N/A                 | NuScale Power Module Vibration                      |