August 16, 2017

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. 68 (eRAI No. 8740) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 68 (eRAI No. 8740)," dated June 20, 2017

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 Question from NRC eRAI No. 8740:

- 14.02-1

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 Steven Mirsky at 240-833-3001 or at smirsky@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Demetrius Murray, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8740

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8740

NuScale Power

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

eRAI No.: 8740
Date of RAI Issue: 06/20/2017

NRC Question No.: 14.02-1

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47, "Contents of applications; technical information," requires that an application for a design certification must include performance requirements and design information sufficiently detailed to permit its acceptance by the U.S. Nuclear Regulatory Commission (NRC). Specifically, §52.47(a)(2)(iii), states, in part, that the NRC will take into consideration the following reactor design characteristics that include, "the extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials." In addition, 52.47(c)(2) states, in part, that "an application for certification of a nuclear power reactor design that differs significantly from the light-water reactor designs or uses simplified, inherent, passive, or other innovative means to accomplish its safety functions must provide an essentially complete nuclear power reactor design and must meet the requirements of 10 CFR 50.43(e)." 10 CFR 50.43(e) states:

"Applications for a design certification, combined license, manufacturing license, or operating license that propose nuclear reactor designs which differ significantly from light-water reactor designs that were licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their safety functions, will be approved only if:
(1)(i) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;
(ii) Interdependent effects among the safety features of the design are acceptable, as demonstrated by analysis, appropriate test programs, experience, or a combination thereof; and
(iii) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions; or
(2) There has been acceptable testing of a prototype plant over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions."

Regulatory Guide (RG) 1.68, Section B, states, in part, that "if the facility is using first-of-a-kind (FOAK) SSCs [structures, systems, and components] that are new, unique, or special design feature in the facility, then the in-plant functional testing requirements needed to verify their performance should be identified at an early date to permit the test requirements to be appropriately accounted for in the final test design.”

In addition, Design-Specific Review Standard (DSRS) 14.2 states, in part, the design certification applicant would provide design and test acceptance criteria for preoperational, low power, and power ascension tests that are unique, unusual, FOAK design features or enhanced safety features.

Currently, in NuScale’s Design Certification Application (DCA) Section 14.2.3.3, “Testing of First-of-a-Kind Design Features,” the description includes a portion on FOAK testing. The DCA states, “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.” The only FOAK test the applicant addresses is the reactor internals vibration testingrequired per RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperation and Initial Startup Testing.” At a minimum, the emergency core cooling system, decay heat removal system, high pressure containment, and some instrumentation (for example, Reactor Coolant System flow) are new and unique and have not been previously tested or reviewed by the NRC.

1. Provide the requisite FOAK tests for Section 14.2.3.3 and the test abstracts for Section 14.2.12, “Individual Test Descriptions," for new and unique design features that have not been previously tested or reviewed by the NRC.

Previously approved DC that included design features, which are new and unique and have not been tested previously, included in their initial test program FOAK tests that were only to be performed a limited number of times, typically called First Plant Only Tests (FPOTs). These FPOTs were performed to confirm proper operation of the new design or to confirm testing of prior smaller applications or scale model testing is valid for the full sized plant.

2. NuScale should also identify: 1) if any tests are FPOTs; 2) if NuScale is proposing any tests for only the first plant (or module); and, 3) provide the basis for not performing the test on every plant (or module).

NuScale Response:

In response to part 1) of this RAI, NuScale performs testing of new and unique design features on every module as identified in individual test descriptions in Section 14.2.12. The testing of new and unique design features are performed and described in Section 14.2.12 Individual Test Descriptions. Attached is a revision to FSAR Section 14.2.3.3 that includes Table 14.2-110 which 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.

In Tier 2 FSAR Section 1.5.1, NuScale Testing Programs, testing programs that have been completed or are currently in progress are identified. These tests focus on design features of the NPM for which applicable data or operational experience did not previously exist. The three areas of focus for NuScale testing programs in this FSAR section are fuel, steam generators, and the control rod drive assembly. Reference to FSAR Section 1.5.1 is included in the attached revision to FSAR Section 14.2.3.3.

In response to part 2) of this RAI, Tier 2 FSAR Section 14.2.3.3 contains the CVAP Reactor Module Vibration Test #108 as an on-site, first module only test abstract. FSAR Section 14.2.3.3 has been revised to clarify the description of ITP CVAP flow-induced vibration testing described in the test abstracts for the control rod drive system, reactor vessel internals, and the steam generators.

The basis for not performing CVAP FOAK tests on every module is that future NPMs, after the first NPM, meet the classification requirements of non-prototype Category I designs in accordance with Regulatory Guide 1.20, Revision 3. There are no differences between NPMs other than allowable variations in manufacturing tolerances, which are bounded by the validated safety margins. The test results for the first NPM CVAP testing will be used to inform CVAP testing of subsequent NPMs as described in Section 6.0 of TR-0716-50439.

Impact on DCA:

FSAR Section 14.2.3.3 and FSAR Tables 14.2-44, 14.2-45,14.2-72, and 14.2-108 have been revised as described in the response above and as shown in the markup provided in this response.

Additional Information:

New FSAR Table 14.2-110, ITP Testing of New Design Features, has been added.

- 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.)
- Equipment protection logic

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

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. The CVAP is addressed in ~~Design Control Document (DCD)~~ Section 3.9.2. ~~Table 14.2-108: Reactor Module Vibration Test #108 contains a startup test to support the CVAP.~~

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: Reactor Module Vibration Test #108

RAI 14.02-1

The test results for the CVAP program testing of the first NPM will be used 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.

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.

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

RAI 14.02-1

Table 14.2-44: Control Rod Drive System Flow-Induced Vibration Test # 44

~~The control rod drive system (CRDS) flow-induced vibration (FIV) testing is performed during startup testing.~~ 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 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 ~~DCD~~ Section 3.9.2. The CRDS is discussed in ~~DCD~~ Section 4.6.		
System Function	**System Function Categorization**	**Function Verified by Test #**
None	N/A	N/A
Prerequisites: N/A		
Component Level Tests		
None		

RAI 14.02-1

Table 14.2-45: Reactor Vessel Internals Flow-Induced Vibration Test # 45

~~Reactor vessel internals (RVI) flow-induced vibration testing is performed during startup testing.~~ 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 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 ~~DCD~~ Section 3.9.2. Reactor vessel internals are discussed in ~~DCD~~ Section 5.1.3.3.		
System Function	**System Function Categorization**	**Function Verified by Test #**
None	N/A	N/A
Prerequisites: N/A		
Component Level Tests		
None		

RAI 14.02-1

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

~~SG flow-induced vibration testing is performed during startup testing.~~ 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. There are no preoperational tests for SG.		
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. Visual examination of the SG 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 ~~DCD~~ Section 3.9.2. The steam generators are discussed in ~~DCD~~ 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		

RAI 14.02-1

Table 14.2-108: Reactor Module Vibration Test (Test #108)

This startup test is required to be performed once for NPM #1. This test supports FOAK testing described in Section 14.2.3.3.
This test is performed at 100 percent reactor thermal power. Reactor 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.
Test Objective for NPM #1
i. Perform vibration testing of DHRS steam piping at 100 percent reactor thermal power as described in TR-0716-50439, Section 4.3, to verify vibration amplitudes in the DHRS steam piping confirm the acoustic resonance analysis results described in TR-0716-50439 Section 4.3. ii. Perform visual ~~testing~~examination of the reactor module components specified in Table 5-1 of TR-0716-50439.
Prerequisites
i. The DHRS steam piping is instrumented to obtain acoustic resonance (AR) data.
Test Method
i. Operate the reactor module 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 DHRS steam piping. 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 reactor module and performed a visual ~~inspection~~examination of the reactor module components specified in Table 5-1 of TR-0716-50439.
Acceptance Criterion
i. Measured vibration amplitudes in the DHRS steam piping confirm the acoustic resonance analysis results described in TR-0716-50439 Section 4.3. ii. Visual ~~inspection~~examination results of reactor module components satisfy the acceptance criteria of Table 5-1 of TR-0716-50439.

RAI 14.02-1

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	#43-1
	• valve response to manual ESF action at hot functional test pressure and temperature	#63-6
	• valve response time test at hot functional test pressure and temperature	#63-7
	• valve response to manual reactor trip at 100% power	#104
ECC Valve Design	• valve response to manual ESF action at hot functional test pressure and temperature	#63-6
	• test of valve inadvertent actuation block at design pressure	
DHR Valve Design	• valve response to manual ESF action at hot functional test pressure and temperature	#63-6
	• valve response to manual reactor trip at 100% power	#104
DHR Heat Exchanger Design	• heat exchanger response to manual reactor trip at 100% power	#104
Contained Flooding and Drain (CFD) System	• automatic fill of containment	#42
	• automatic drain of containment	
Containment Evacuation System (CES)	• establish and maintain containment vacuum	#41
	• provide RCS leakage detection	
CNT Level Sensors	• provides containment level input for CFD system automatic fill and drain of containment	#42
RCS Flow Sensors	• provides RCS flow indication during HFT and power ascension testing	#77 #94
Pressurizer Level Sensors	• Provides input for pressurizer level control	#38-1