RAIO-0718-61100

July 26, 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 Supplemental Response to NRC Request for Additional Information No. 372 (eRAI No. 9364) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 372 (eRAI No. 9364)," dated February 27, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 372 (eRAI No.9364)," dated April 20, 2018

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

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Question from NRC eRAI No. 9364:

- 14.03.03-9

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 Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

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

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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9364

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9364

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

eRAI No.: 9364
Date of RAI Issue: 02/27/2018

NRC Question No.: 14.03.03-9

10 CFR 52.47(b)(1) requires “The proposed inspections, tests, analyses, and acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the [Atomic Energy] Act, and the Commission's rules and regulations.” In supporting this requirement, the Tier 2 material provides important clarifications to the Tier 1 material and should therefore be as clear as possible with respect to referenced information. The staff notes that references to Tables in the narrative discussion for DCD Tier 2, Table 14.3-1 do not specify Tier 1 or Tier 2. This may provide confusion to a user of this document. For instance, “In accordance with Table 14.2-63, a preoperational test demonstrates that the ECCS safety-related valves listed in Table 2.1-2 stroke fully open…,” refers to both a Table in Tier 2 and Tier 1 without differentiation. Please provide clarification to the language in the DCD.

NuScale Response:

This supplements the RAI No. 9364, question 14.03.03-9 response submitted to the NRC via letter RAIO-0418-59636 on April 20, 2018 (ML18110A359). In a June 21, 2018 teleconference with the NRC on the RAI 9364, question 14.03.03-9 response, NRC commented that the response was insufficient in that NuScale did not make conforming changes to put "Tier 1" modifiers for ITAAC Nos. 02.01.13 through 02.01.20 in Tier 2, Table 14.3-1.

NuScale made the "Tier 1" modifier changes in Tier 2, Table 14.3-1 for ITAAC Nos. 02.01.13 through 02.01.20.

Impact on DCA:

Tier 2, Table 14.3-1 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 08.01-1S1, RAI 08.01-2, RAI 10.02-3, RAI 10.02.03-1, RAI 10.02.03-2, RAI 14.03.03-3S1, RAI 14.03.03-4S1, RAI 14.03.03-6, RAI 14.03.03-6S1, RAI 14.03.03-7, RAI 14.03.03-7S1, RAI 14.03.03-8, RAI 14.03.03-9, RAI 14.03.03-9S1

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 Report 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
02.01.12	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 approximately 90-degree intervals.	X				
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 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 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 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	NPM	The reactor coolant system (RCS) 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 14.2-46, a preoperational test demonstrates that the RCS check valves listed in Tier 1 Table 2.1-2 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 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
02.01.17	NPM	The RCS safety-related excess flow check valves are tested to demonstrate the capability to perform their function to stroke fully closed under excess flow conditions 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 14.2-46, a preoperational test demonstrates that the RCS check valves listed in Tier 1 Table 2.1-2 strokes fully closed under forward flow 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		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 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 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 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				