RAIO-0418-59775

May 01, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 384 (eRAI No. 9391) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 384 (eRAI No. 9391)," dated March 12, 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 Question from NRC eRAI No. 9391:

- 05.02.05-8

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 Response to NRC Request for Additional Information eRAI No. 9391

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9391

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

eRAI No.: 9391
Date of RAI Issue: 03/12/2018

NRC Question No.: 05.02.05-8

10 CFR 52.47(a)(2) requires that a standard design certification applicant provide a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. Furthermore, 10 CFR 52.47(a)(3)(i) requires that an application for design certification include principal design criteria for the facility and states that Appendix A to 10 CFR Part 50, general design criteria (GDC), establishes the minimum requirements for principal design criteria.

GDC 30, "Quality of reactor coolant pressure boundary," states, in part, "Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage."

RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," Revision 1, Regulatory Positions C.1.1 and C.2.3 provide guidance on leakage detection systems and the use of supplemental instruments/methods for identifying and locating source of leakage.

The NRC staff reviewed the response to RAI 8915 regarding supplemental methods for reactor coolant pressure boundary leakage detection and identified the need for the following additional information for completing the safety evaluation.

1. In the RAI response, NuScale identified several additional methods that can be used for identifying the source of leakage. This information is necessary to support a staff conclusion that the leakage detection system can be used to identify the leakage source consistent with DSRS Section 5.2.5. Because thecurrent FSAR does not have sufficient details for the methods used to identify the source of leakage, the applicant is requested to provide such information in FSAR Tier, 2 Section 5.2.5, "Reactor Coolant Pressure Boundary Leakage Detection."
2. In the RAI response, NuScale stated that "there is no method within the NuScale design to identify the exact location of the leakage during power operation." The applicant is requested to expand its initial response to RAI 8915 by clearly providing the

bases/justification for not having the capability to locate the source of leakage without introducing any adverse safety concerns to the operation of a NuScale module. Such information will better clarify the extent of the NuScale leakage detection design and conformance with GDC 30. The applicant may also wish to assess the wording of COL Item 5.2-7 which indicates “a COL applicant … will establish … procedures that specify operator actions for … locating reactor coolant system leakage …” and appears to be inconsistent with the response to RAI 8915.

NuScale Response:

1. As described in the response to NRC RAI 8915 - 05.02.05-5, NuScale provided a discussion of additional leak detection methods for determining leakage in the containment. NuScale has updated DCD section 5.2.5.1 to provide the additional leak detection methods utilized in the NuScale Power Module (NPM) design.

2. As identified in RG 1.45, the existing fleet of pressurized water reactors (PWRs) utilize several methods to identify the source and approximate location of a leak within containment including monitoring local temperature, local humidity, sump level, radioactivity, or acoustic emission. Once a potential leak is identified, a containment entry is required to perform a visual inspection to determine the exact location.

   As the NPM containment is held at a pressure below the vapor pressure of the lowest surface temperature inside of the containment vessel, accumulation of liquid within the vessel during normal operation is not feasible. Considering this, the use of local instrumentation to identify the exact leak location is not physically achievable as any leakage will enter the containment as a vapor and rapidly expand into the vacuum. Further, access of containment during normal operation to perform visual inspection of suspected leakage areas is not possible due to the vacuum conditions.

   In comparison to a current PWRs, the majority of the reactor coolant system loop is contained within the reactor vessel. Only a handful of flanges and valves are mechanically connected, limiting the number of potential leak points. Additionally, a pressure test followed by a visual leakage exam (performed by the continuous monitoring of the Containment Evacuation System (CES) in accordance with American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code (BPVC) Section XI, IWA-5241(c)) is required for the Class 1 pressure retaining boundary each refueling outage before startup per ASME BPVC, Section XI, Division 1, Table IWB 2500-1 B-P. This test verifies the leak tightness of the RPV and connected piping in containment prior to reactor restart.

   NuScale has updated COL Item 5.2-7 to clarify the intent of the leakage detection capability requirements.

Impact on DCA:

FSAR Table 1.8-2, Section 5.2.5.1 and Section 5.2.5.7 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 01-61, RAI 02.04.13-1, RAI 03.04.02-1, RAI 03.04.02-2, RAI 03.04.02-3, RAI 03.05.01.04-1, RAI 03.05.02-2, RAI 03.06.02-15, RAI 03.06.03-11, RAI 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.08.04-23S1, RAI 03.08.05-14S1, RAI 03.09.02-15, RAI 03.09.02-48, RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-6, RAI 03.09.06-16, RAI 03.09.06-16S1, RAI 03.09.06-27, RAI 03.11-8, RAI 03.11-14, RAI 03.11-14S1, RAI 03.11-18, RAI 03.13-3, RAI 05.02.05-8, RAI 05.04.02.01-13, RAI 05.04.02.01-14, RAI 06.04-1, RAI 09.01.02-4, RAI 09.01.05-3, RAI 09.01.05-6, RAI 09.03.02-3, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8, RAI 10.02-1, RAI 10.02-2, RAI 10.03.06-1, RAI 10.03.06-5, RAI 10.04.06-1, RAI 10.04.06-2, RAI 10.04.06-3, RAI 10.04.10-2, RAI 13.01.01-1, RAI 13.01.01-1S1, RAI 13.02.02-1, RAI 13.03-4, RAI 13.05.02.01-2, RAI 13.05.02.01-2S1, RAI 13.05.02.01-3, RAI 13.05.02.01-3S1, RAI 13.05.02.01-4, RAI 13.05.02.01-4S1, RAI 14.02-7, RAI 19-31, RAI 19-31S1, RAI 19-38

Table 1.8-2: Combined License Information Items

Item No.	Description of COL Information Item	Section
COL Item 1.1-1:	A COL applicant that references the NuScale Power Plant design certification will identify the site-specific plant location.	1.1
COL Item 1.1-2:	A COL applicant that references the NuScale Power Plant design certification will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1:	A COL applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1:	A COL applicant that references the NuScale Power Plant design certification will provide site-specific diagrams and legends, as applicable.	1.7
COL Item 1.7-2:	A COL applicant that references the NuScale Power Plant design certification will list additional site-specific piping and instrumentation diagrams and legends as applicable.	1.7
COL Item 1.8-1:	A COL applicant that references the NuScale Power Plant design certification will provide a list of departures from the certified design.	1.8
COL Item 1.9-1:	A COL applicant that references the NuScale Power Plant design certification will review and address the conformance with regulatory criteria in effect six months before the docket date of the COL application for the site-specific portions and operational aspects of the facility design.	1.9
COL Item 1.10-1:	A COL applicant that references the NuScale Power Plant design certification will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk significant structures, systems, and components of existing operating unit(s) and newly constructed operating unit(s) at the co-located site per 10 CFR 52.79(a)(31). The evaluation will include identification of management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the limiting conditions for operation of an operating unit would not be exceeded. This COL item is not applicable for construction activities (build-out of the facility) at an individual NuScale Power Plant with operating NuScale Power Modules.	1.10
COL Item 2.0-1:	A COL applicant that references the NuScale Power Plant design certification will demonstrate that site-specific characteristics are bounded by the design parameters specified in Table 2.0-1. If site-specific values are not bounded by the values in Table 2.0-1, the COL applicant will demonstrate the acceptability of the site-specific values in the appropriate sections of its combined license application.	2.0
COL Item 2.1-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site geographic and demographic characteristics.	2.1
COL Item 2.2-1:	A COL applicant that references the NuScale Power Plant design certification will describe nearby industrial, transportation, and military facilities. The COL applicant will demonstrate that the design is acceptable for each potential accident, or provide site-specific design alternatives.	2.2
COL Item 2.3-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific meteorological characteristics for Section 2.3.1 through Section 2.3.5, as applicable.	2.3
COL Item 2.4-1:	A COL applicant that references the NuScale Power Plant design certification will investigate and describe the site-specific hydrologic characteristics for Section 2.4.1 through Section 2.4.14, as applicable.	2.4
COL Item 2.5-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific geology, seismology, and geotechnical characteristics for Section 2.5.1 through Section 2.5.5, below.	2.5
COL Item 3.2-1:	A COL applicant that references the NuScale Power Plant design certification will update Table 3.2-1 to identify the classification of site-specific structures, systems, and components.	3.2

Table 1.8-2: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 5.2-6:	A COL applicant that references the NuScale Power Plant design certification will develop site-specific preservice examination, inservice inspection, and inservice testing program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and will establish implementation milestones. If applicable, a COL applicant that references the NuScale Power Plant design certification will identify the implementation milestone for the augmented inservice inspection program. The COL applicant will identify the applicable edition of the American Society of Mechanical Engineers Code utilized in the program plans consistent with the requirements of 10 CFR 50.55a.	5.2
COL Item 5.2-7:	A COL applicant that references the NuScale Power Plant design certification will establish plant-specific procedures that specify operator actions for identifying, monitoring, and trending~~, and locating~~ reactor coolant system leakage in response to prolonged low leakage conditions that exist above normal leakage rates and below the technical specification limits. The objective of the methods of detecting and ~~locating~~trending the reactor coolant pressure boundary leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached.	5.2
COL Item 5.3-1:	A COL applicant that references the NuScale Power Plant design certification will establish measures to control the onsite cleaning of the reactor pressure vessel during construction in accordance with Regulatory Guide 1.28.	5.3
COL Item 5.3-2:	A COL applicant that references the NuScale Power Plant design certification will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy had been demonstrated.	5.3
COL Item 5.3-3	A COL applicant that references the NuScale Power Plant design certification will describe their reactor vessel material surveillance program consistent with NUREG 0800, Section 5.3.1.	5.3
COL Item 5.4-1:	A COL applicant that references the NuScale Power Plant design certification will develop and implement a Steam Generator Program for periodic monitoring of the degradation of steam generator components to ensure that steam generator tube integrity is maintained. The Steam Generator Program will be based on the latest revision of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," and applicable Electric Power Research Institute steam generator guidelines at the time of the COL application. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, shell side integrity and accessibility assessment, steam plant corrosion product deposition assessment, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting.	5.4
COL Item 6.2-1:	A COL applicant that references the NuScale Power Plant design certification will develop a containment leakage rate testing program that will identify which option is to be implemented under 10 CFR 50, Appendix J. Option A defines a prescriptive-based testing approach whereas Option B defines a performance-based testing program.	6.2
COL Item 6.3-1:	A COL applicant that references the NuScale Power Plant design certification will describe a containment cleanliness program that limits debris within containment. The program should contain the following elements: • Foreign material exclusion controls to limit the introduction of foreign material and debris sources into containment. • Maintenance activity controls, including temporary changes, that confirm the emergency core cooling system function is not reduced by changes to analytical inputs or assumptions or other activities that could introduce debris or potential debris sources into containment. • Controls that limit the introduction of coating materials into containment. • An inspection program to confirm containment vessel cleanliness prior to closing for normal power operation.	6.3
COL Item 6.4-1:	A COL applicant that references the NuScale Power Plant design certification will comply with Regulatory Guide 1.78 Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."	6.4
COL Item 6.4-2:	Not used.	6.4

portion of the CES, at which point containment pressure would increase and isolate the containment.

The CES inlet pressure instrumentation is designed to Seismic Category I and ensures that these components maintain the capability to perform their safety leak monitoring function during and after a safe shutdown earthquake. Therefore, the CES inlet pressure instrumentation is also capable of detecting changes in the containment atmospheric conditions, including leakage from the RCPB, during a seismic event that does not result in an NPM shutdown.

RAI 05.02.05-8

Radioactivity Monitoring & Chemistry Analysis

RAI 05.02.05-8

The CES has the ability to monitor both gaseous and liquid effluent removed from containment for radioactivity using remote radioactivity instrumentation (See Figure 9.3.6-1).

RAI 05.02.05-8

Gaseous discharge of the CES is directed to the process sampling system sample panel for continuous analysis or collection of grab samples if needed. A grab sample location is also provided on the CES sample vessel, allowing for samples to be collected and analyzed in the laboratory.

RAI 05.02.05-8

Using the different radioisotope characteristics of systems (including the use of tracer gases), the source of a leak could be determined. FSAR Section 11.5 provides discussion on the use of radioisotopes to aid in determining the source of leakage.

RAI 05.02.05-8

RCS Inventory Mass Balance

RAI 05.02.05-8

The licensee will regularly perform a reactor coolant system inventory balance to determine RCS leakage quantity as part of technical specification surveillance requirement 3.4.5.1.

RAI 05.02.05-8

Using the inventory mass balance will augment radiation instrumentation, chemistry analysis, and CES leak detection methods to determine the source of a leak inside containment. To the extent practical, the use of these diverse methods provides identification of the location of the source of reactor coolant leakage.

- letdown to the liquid radioactive waste system.

Intersystem leakage is identified by:

- increasing level, temperature, flow or pressure.
- relief valve actuation.
- increasing radioactivity.

Refer to Sections 9.3.3, 9.3.4, and 11.5 for further discussion related to the CVCS intersystem leakage detection and monitoring capabilities.

5.2.5.6 Reactor Component Cooling Water System Leakage Monitoring

Leakage detection for the RCCWS is provided by monitoring expansion tank level and an alarm is provided in the control room. In the event radioactivity is introduced into the RCCWS piping, radiation elements and transmitters located downstream of non-regenerative heat exchanger, the process sampling system cooler lines, and the CES condenser for each NPM detect the radiation and alarm in the control room. For additional information on RCCW, see Section 9.2.2.

5.2.5.7 Primary to Secondary Leakage Monitoring

Radiation monitoring of the gaseous effluent from the condenser air removal system is provided to detect primary to secondary leakage. Radiation monitoring is also provided on the main steam lines condenser air removal system, and turbine sealing steam system. The capability to attain grab samples of steam and feedwater to analyze for indications of primary to secondary leakage is also provided. Additional detail of gaseous and liquid effluent radioactivity monitoring is provided in Section 11.5.

RAI 05.02.05-8

COL Item 5.2-7: A COL applicant that references the NuScale Power Plant design certification will establish plant-specific procedures that specify operator actions for identifying, monitoring, and trending~~, and locating~~ reactor coolant system leakage in response to prolonged low leakage conditions that exist above normal leakage rates and below the technical specification limits. The objective of the methods of detecting and ~~locating~~trending the reactor coolant pressure boundary leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached.

5.2.6 References

5.2-1 NuScale Power, LLC, "Non-LOCA Transient Analysis Methodology," Topical Report TR-0516-49416, Revision 0.

5.2-2 NuScale Power, LLC, "LOCA Evaluation Model," Topical Report TR-0516-49422, Revision 0.