RAIO-1017-56676

October 18, 2017

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. 193 (eRAI No. 9079) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 193 (eRAI No. 9079)," dated August 19, 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. 9079:

- 06.04-1

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

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9079

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9079

NuScale Power

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

eRAI No.: 9079
Date of RAI Issue: 08/19/2017

NRC Question No.: 06.04-1

10 CFR 52.47(a)(2) requires that a standard design certification application include a final safety analysis report (FSAR) that describes the design of the facility including the principal design criteria for the facility, for which NuScale used the 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." General Design Criterion (GDC) 19 requires that a control room be provided with adequate radiation protection to permit access and occupancy of the control room under accident conditions without the personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. NuScale design-specific review standard (DSRS) section 15.0.3 provides additional guidance on the evaluation of control room radiological habitability.

NuScale FSAR Tier 2, Section 6.4, “Control Room Habitability,” and Section 9.4.1, "Control Room Area Ventilation System," discuss the two non-safety-related ventilation systems that are to be relied upon to meet the requirements of GDC 19, including protection to the control room occupants from radioactive releases during accident conditions. These heating, ventilation and air conditioning (HVAC) systems are the control room habitability system (CRHS) and the normal control room HVAC system (CRVS). FSAR Tier 2, Section 15.0.3 provides a discussion of the design basis accident (DBA) radiological consequence analyses, which includes dose results in the control room.

The NuScale DBA radiological consequence analyses assume that after a 10-minute loss of ac power, or based on reaching the appropriate control room HVAC intake radiation monitor setpoint, the CRHS system is actuated to provide clean air to the control room envelope for 72 hours, after which the CRVS is available to provide filtered outside air for the duration of the accident. When power is available and the CRHS initiation radiation monitor setpoint has not been reached, the CRVS remains in operation and the filtration function is initiated based on reaching a related control room HVAC intake radiation monitor setpoint. The staff has previously audited NuScale’s DBA radiological consequence analyses (report not yet issued) and noted that the control room doses were calculated for both the case where the CRHS operates for the first 72 hours and the case where the CRVS remains in operation to provide filtration for the duration of the accident.

FSAR Tier 2, subsection 6.4.4, “Design Evaluation,” first paragraph states:

As noted in Section 15.0.0, no operator actions are required or credited to mitigate the consequences of design basis events. As such, the operators perform no safety-related functions, consistent with the definition in 10 CFR 50.2. Therefore, although a habitable control room is provided for the operators, consistent with GDC 19, to perform other important non-safety related functions, the control room envelope and supporting habitability systems and components, including the CRHS, are not safety-related.

FSAR Tier 2, Section 15.0.3 states that the DBAs are analyzed using RG 1.183 guidance, with necessary non-conformances as necessary due to differences in the NuScale design from large light-water reactors. Position 4.2.2 of RG 1.183 states that "[c]redit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed."

Position 5.1.2, "Credit for Engineered Safeguard Features," states:

Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.

Because the NuScale design does not classify either the CRHS or the CRVS as safety-related, the staff requests the following additional information to conduct its review of the NuScale control room radiological habitability and compliance with GDC 19:

a. Considering that the CRHS is not a safety-related system and therefore does not necessarily meet all the engineered safety feature requirements but is relied upon to provide a safe environment for control room operators under accident conditions, how is the reliability of the system ensured to be able to assume that the system will operate as expected and needed under accident conditions?
b. Considering that the CRVS is not a safety-related system and therefore does not necessarily meet all the engineered safety feature requirements, but is relied upon to provide a safe environment for control room operators under accident conditions how is the reliability of the system ensured to be able to assume that the system will operate as expected and needed under accident conditions?
c. What are the effects on the control room radiological habitability in the event that the non-safety-related CRHS fails to operate during a DBA?
d. What are the effects on the control room radiological habitability in the event that the non-safety-related CRVS fails to operate during a DBA?
e. What are the effects on the control room radiological habitability in the event that both the CRHS and the CRVS fail to operate during a DBA?

NuScale Response:

Questions (a) and (b)

The set of questions in this RAI are concerned with the radiological protection of control room personnel during accident conditions. Questions (a) and (b) ask how the reliability of the CRVS and CRHS, both non safety-related systems, are assured so that these systems may be assumed to operate under accident conditions.

The operation of the CRVS and CRHS are described in the NuScale FSAR sections 9.4.1 and 6.4 respectively. Regulatory positions 4.2.4 and 5.1.2 of Regulatory Guide 1.183 do not apply to these systems because the CRVS and CRHS do not perform engineered safety functions. FSAR Section 9.4.5 states that the NuScale design does not utilize engineered safety features for ventilation systems to mitigate the consequences of a design basis accident, as there are no design basis events that result in significant core damage.

As noted in FSAR Section 15.0.0, no operator actions are required or credited to mitigate the consequences of design basis events. As such, the operators perform no safety-related functions as defined in 10 CFR 50.2 either during or after the 72-hour period following a design basis accident. Operator actions are not required to protect the integrity of the reactor coolant pressure boundary, or to shut the reactors down or maintain them in a safe shutdown condition. For the NuScale design, no operator actions are credited in demonstrating that no design basis events could result in offsite exposures comparable to the applicable guideline exposures in 10 CFR 50.34(a)(1) or 10 CFR 100.11, therefore operator actions are not relied upon to prevent or mitigate the consequences of such events. Therefore, although a habitable control room is provided for the operators, consistent with GDC 19, to perform other important non-safety related functions, the control room envelope and supporting habitability systems and components, including the CRHS, are not safety-related.

The following measures enhance CRVS and CRHS reliability:

1. All components relied upon for isolation of the control room and activation of the CRHS satisfy the single failure criterion, and are Seismic Category I.
2. CRVS components relied upon for radiation control are designed to ASME AG-1 (Code on Nuclear Air and Gas treatment), ASME N509, ASME N510, and RG 1.140, as appropriate (see FSAR Section 9.4.1 for details).
3. Reliability of air supply and radiation protection to the control room staff is assured by redundant features in the design of the CRVS and CRHS:
    a. Redundant radiation monitors at the outside air intake and downstream of the CRVS filters.
    b. Two sets of control room envelope isolation dampers.
    c. The CRVS, in conjunction with a subsystem of the chilled water system, can

provide cooled and filtered air to the control room using power from the backup diesel generators in the event that normal power is unavailable.

4. COL Items 6.4-5 and 9.4-1 require that the applicant will specify testing and inspection requirements for the CRHS, the CRVS, and for control room envelope integrity.
5. As identified in FSAR Table 3.2-1, components of the CRHS and CRVS that are necessary for radiological protection of control room personnel are designated as augmented quality. In addition to the Seismic Category I designations mentioned above, these augmented quality provisions include
    a. The calibration program for detectors that assure control room envelope isolation.
    b. Separation and isolation criteria in accordance with RG 1.189.
    c. Inclusion of these components in the quality assurance program.

These design specifications and COL items provide a high degree of system reliability and radiation protection for control room personnel under accident conditions.

Question (c)

Questions (c), (d), and (e) are concerned with the effects of control room radiological habitability in the event of failure of the CRHS, the CRVS, or both simultaneously. With respect to the assumption of CRHS and CRVS system reliability justified above, radiological effects of these failures were not specifically analyzed.

The CRHS is a passive system that supplies air from a bottled air supply to the control room envelope (CRE) for 72 hours. As described in FSAR Section 6.4 and shown in FSAR Figure 6.4-1, the CRHS includes a normal and a backup flow path to the CRE, and multiple points of delivery within the CRE. All CRHS components needed to deliver the 72-hour air supply are Seismic Category 1, as shown in FSAR Table 3.2-1. As described in FSAR Section 9.4.1, all CRE isolation dampers are paired (redundant design) and Seismic Category 1. For these reasons, a failure of the CRHS coincident with a design basis event is not considered credible.

Question (d)

For each design basis event and for the maximum hypothetical accident (MHA), two scenarios were analyzed to determine the dose to control room personnel. Results for the most limiting cases are summarized in FSAR Table 15.0-12. In one scenario, the filtered path remains in service for the duration of the event. In the second scenario, the CRHS is placed in service when any of the redundant post-filter radiation monitors reaches its setpoint. CRHS remains in service for the following 72 hours, and then the CRVS provides filtered airflow for the remainder of the event.

In the event of a failure of the CRVS coincident with a design basis event, the redundant CRE isolation dampers would close and CRHS would be placed into service. As described in FSAR

Section 9.4.1, this action would be automatic if radiation exceeded the high setpoint of the radiation monitors downstream of the CRVS air filtration unit, or if power was lost to both CRVS air handler units for 10 minutes. The CRHS would then provide air to the CRE for 72 hours, with radiological consequences as analyzed for the second scenario described above. Therefore, failure of the CRVS does not result in radiation doses in excess of those reported in FSAR Table 15.0-12.

Question (e)

As explained in the response to Question (c), a failure of the CRHS coincident with a design basis event is not considered credible because of the redundancy and seismic qualification of the components. Therefore, the simultaneous failure of CRVS and CRHS coincident with a design basis event is not considered credible.

As added protection against overexposure to radiation, the control room is equipped with area radiation monitors. If the radiation level should exceed preset limits, which will be determined by the licensee in accordance with their radiation protection program, the operators would trip any operating reactors, initiate decay heat removal and containment isolation, and vacate the control room. Each reactor module would then reach safe shutdown conditions without operator action.

Although unrelated to this RAI, NuScale is also communicating the following changes to FSAR Section 6.4:

> COL Items 6.4-3 and 6.4-4 are being deleted because they are redundant to information in the text. Text under the heading of Toxic Gas Protection and COL Item 6.4-2 under the same heading are being deleted because they are redundant to COL Item 6.4-1.

An FSAR markup is included with this response to show these deletions.

Impact on DCA:

FSAR Section 6.4 has been revised as described in the response above and as shown in the markup provided in this response.

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 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.09.02-15, RAI 03.09.06-6, RAI 03.09.06-27, RAI 03.11-8, RAI 03.11-14, RAI 06.04-1, RAI 09.01.05-3, RAI 09.01.05-6, RAI 10.02-1, RAI 10.02-2, 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

Table 1.8-2: Combined License Information Items

Item No.	Description of COL Information Item	Section
COL Item 1.1-1:	A COL ~~Applicant~~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~~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~~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~~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~~applicant that references the NuScale Power Plant design certification will list additional site-specific P&IDs and legends as applicable.	1.7
COL Item 1.8-1:	A COL ~~Applicant~~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~~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~~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 any 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~~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~~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~~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~~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~~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~~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

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

Item No.	Description of COL Information Item	Section
COL Item 5.3-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will establish measures to control the onsite cleaning of the RPV during construction in accordance with RG 1.28.	5.3
COL Item 5.3-2:	A COL ~~Applicant~~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~~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 NEI 97-06, "Steam Generator Program Guidelines," Revision 3 and applicable EPRI steam generator guidelines. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, 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~~applicant that references the NuScale Power Plant design certification will develop a "Containment Leakage Rate Testing Program" which 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~~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:	6.3
COL Item 6.4-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will comply with RG 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:	~~A COL Applicant that references the NuScale Power Plant design certification will specify operator training and qualification in the use of self-contained portable breathing apparatus.~~Not used.	6.4
COL Item 6.4-3:	~~A COL Applicant that references the NuScale Power Plant design certification will specify the technical resources to be stored within the CRE.~~Not used.	6.4
COL Item 6.4-4:	~~A COL Applicant that references the NuScale Power Plant design certification will specify food, water, and medical supplies to be stored within the CRE.~~Not used.	6.4
COL Item 6.4-5:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will specify testing and inspection requirements for the CRHS, including CRE integrity testing.	6.4
COL Item 6.6-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will implement an Inservice InspectionTesting Program in accordance with 10 CFR 50.55a(f).	6.6
COL Item 6.6-2:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will develop a preservice inspection and Inservice Inspection Program plans in accordance with Section XI of the ASME Code, and will establish the implementation milestones for the program. The COL applicant will identify the applicable edition of the ASME Code utilized in the program plan consistent with the requirements of 10 CFR 50.55a. The COL applicant will, if needed, address the use of a single ISI Program for multiple NPMs, including any Alternative to the Code that may be necessary to implement such an ISI Program.	6.6
COL Item 7.2-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification is responsible for the implementation of the life cycle processes for the operation phase for the instrumentation and controls systems, as defined in IEEE Standard 1074-2006 and IEEE Standard 1012-2004.	7.2
COL Item 7.2-2:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification is responsible for the implementation of the life cycle processes for the maintenance phase for the instrumentation and controls systems, as defined in IEEE Standard 1074-2006 and IEEE Standard 1012-2004.	7.2

The CRHS provides bottled air to the CRE for 72 hours following CRE isolation and maintains the CRE at a higher pressure than its surroundings. After 72 hours, the CRHS bottled air supply is depleted and the CRVS (Section 9.4.1) is returned to operation, providing the CRE and the rest of the CRB with filtered air.

With the CRHS in operation, the CRE is maintained at a positive pressure of one-eighth inch water column with respect to its surroundings. The only source of unfiltered leakage into the CRE is due to ingress and egress; up to 5 cfm is considered for this type of inleakage. The dose analysis conservatively includes an additional 10 cfm of inleakage. Before the isolation dampers have closed in response to the high radiation signal, a certain amount of radioactive material will have entered the CRE. This material is gradually diluted over the duration of the accident as air enters and an equal amount of air leaves the CRE. Finally, control room operators will receive a small amount of radiation dose due to airborne radiation outside the CRE (sky shine and direct shine) and from the filters in the CRVS (filter shine).

For the purposes of radiation dose, accident duration is considered to be 30 days in accordance with RG 1.183. Analysis shows that the sum of radiation doses to control room personnel from all sources is less than 5 rem for the duration of any postulated accident. These design features provide compliance with 10 CFR 50.34(f)(2)(xxviii). Radiological dose to control room operators is further addressed in Section 15.0.3.

Toxic Gas Protection

RAI 06.04-1

~~In accordance with RG 1.78, the station maintains a supply of self-contained portable breathing apparatus with air bottles stored onsite. These bottles are used if hazardous chemicals are suspected to be present. The amount of stored air is sufficient to provide a 6-hour supply of breathable air for the number of main control room occupants identified in Technical Specifications Section 5.2.~~

COL Item 6.4-1: A COL applicant that references the NuScale Power Plant design certification will comply with RG 1.78 Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

RAI 06.04-1

COL Item 6.4-2: ~~A COL applicant that references the NuScale Power Plant design certification will specify operator training and qualification in the use of self-contained portable breathing apparatus.~~Not used.

Other Habitability Considerations

When normal air conditioning from the CRVS is not available, the thermal mass of the CRB and its contents limit the temperature increase as shown in Table 6.4-3 for the first 72 hours following an accident. The peak temperature at three hours is the result of a conservative assumption in the analysis, that control room equipment powered by the normal DC power system remains powered for three hours. After 72 hours, the CRVS (Section 9.4.1) provides cooling to the CRE.

The MCR is provided with adequate lighting for safe operation of the plant during accidents and all modes of operation. Refer to Section 9.5.3 for more information.

The CRE includes access to procedures, drawings, and other technical resources useful for mitigating accident conditions. The CRE includes a lavatory, kitchen facilities, and a dining area. Food, water, and first-aid medical supplies are stored in an accessible location within the CRE. The food and water supply is adequate for the minimum needs of the number of main control room occupants identified in Technical Specifications Section 5.2 for 72 hours.

RAI 06.04-1

COL Item 6.4-3: ~~A COL applicant that references the NuScale Power Plant design certification will specify the technical resources to be stored within the CRE.~~Not used.

RAI 06.04-1

COL Item 6.4-4: ~~A COL applicant that references the NuScale Power Plant design certification will specify food, water, and medical supplies to be stored within the CRE.~~Not used.

6.4.5 Testing and Inspection

Refer to Section 14.2 for information regarding preoperational testing.

Inservice testing of the CRHS is conducted in accordance with the surveillance requirements specified in the technical specifications. Leak tightness testing of the CRE pressure boundary is conducted in accordance with the frequency specified in the technical specifications. Inservice testing includes demonstration of the integrity of the CRE in accordance with RG 1.197.

In addition to periodic tests, CRE testing is performed when changes are made to structures, systems, and components that could impact CRE integrity, including systems internal and external to the CRE. These tests are commensurate with the types and degrees of modifications and repairs and the potential impact upon integrity. Additional CRE testing is also performed if a new limiting condition or alignment arises for which no in-leakage data is available (e.g., a hazardous chemical source appears where previously there was none). In-leakage in excess of the licensing basis value for the particular challenge to CRE integrity constitutes test failure.

COL Item 6.4-5: A COL applicant that references the NuScale Power Plant design certification will specify testing and inspection requirements for the CRHS, including CRE integrity testing.

6.4.6 Instrumentation Requirements

The CRHS design includes instrumentation and controls to ensure safe, efficient, and reliable operation. Instrumentation and controls for the CRHS are part of the plant protection system, which is described in Chapter 7.