RAIO-0617-54590

June 26, 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. 06 (eRAI No. 8775) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 06 (eRAI No. 8775)," dated April 25, 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 Enclosures to this letter contain NuScale's response to the following RAI Question from NRC eRAI No. 8775:

- 12.03-1

NuScale requests that the security-related information in Enclosure 1 be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. Enclosure 2 contains a public version of the NuScale response.

This letter and the enclosed responses 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, TWFN-6E55
Samuel Lee, NRC, TWFN-6C20
Anthony Markley, NRC, TWFN-6E55

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8775, Nonpublic

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8775, Public

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8775, Nonpublic

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 8775, Public

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

eRAI No.: 8775
Date of RAI Issue: 04/25/2017

NRC Question No.: 12.03-1

10 CFR 52.47(a)(8) requires that the final safety analysis report provide the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

10 CFR 50.34(f)(2)(vii) requires that applicants perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.

10 CFR 50.34(f)(2)(viii) requires that applicants provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.

NUREG-0737 and DSRS section 12.3-12.4 provide additional guidance on acceptable methods of meeting these requirements. These documents indicates that post accident radiation zones should be provided based on the guidance of RGs 1.7 and 1.183 and that the analysis for access to vital areas should consider access to, stay time in, and egress from these vital areas. NUREG-0737 specifies that any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is to be designated as a vital area and in addition to the control room and technical support center, the sample station and sample analysis area must be included among those areas where access is considered vital after an accident (This question is focused on the sample station, sample analysis, and other areas requiring infrequent access. Any questions related to the MCR and TSC will be addressed separately). Finally, NUREG-0737 provides a list of other areas that should be considered in determining the vital areas. NUREG-0737 specifies that if these areas are not considered vital areas, justification should be provided for not including them. The areas specified are the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area (if applicable), motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels. In addition, any other areas that may need to be accessed during an accident are to be identified. As specified, the plant should be designed so that the dose to an individual should not exceed the occupational dose criteria to perform the vital missions, including accessing and egressing from the areas.

DCD Section 12.4.1.8, "Post-Accident Actions," provides a discussion of post-accident sampling and analysis for both primary liquid sampling and containment gas sampling. It indicates that access may be required to the CVCS gallery, counting room, and hot lab in the Reactor Building, and Annex building counting room may be required to sample and analyze liquid samples and that access to the utilities area and steam gallery on the 100' elevation of the reactor building may be required for containment gaseous grab sampling. The discussion to perform these activities indicates that doses will be under the 5 rem occupational dose limit, including ingress and egress. However, it indicates that 0.25" lead equivalent temporary shielding is assumed in the analysis for calculating the post-accident doses (in addition to the permanent shielding specified in the application) and there is no discussion of the dose rates received in installing the temporary shielding to the various different areas. In addition, no post-accident radiation zone maps or ingress/egress routes are provided; no information is provided on the time assumed to be spent at each location or the speed of travel assumed; and no information is provided on the assumptions made regarding submersion and inhalation dose. Finally, staff did not find any information regarding any other vital areas in the DCA.

Based on the above, staff requests the following:

1. Provide more information regarding where temporary lead equivalent shielding is assumed to be installed as it relates to taking and analyzing liquid and gaseous samples, including access and egress, and the dose rates that will be received in installing this temporary shielding during accident conditions.
2. Provide post-accident radiation zone maps and ingress and egress routes for all areas associated with performing the vital missions and update the DCD to include this information.
3. Provide information on the time spent at each location and the speed of travel assumed and update the DCD to include this information.
4. Provide information on the assumed post-accident airborne activity concentrations and how these source terms were developed. In addition, provide information regarding what assumptions are made regarding submersion and inhalation dose. Update the DCD accordingly.
5. Identify any other areas that would require or may require access following an accident to permit an operator to aid in the mitigation of or recovery from an accident. For each of these areas, identify the work that may need to be performed in these areas, provide the dose to perform the work and to access and egress from the area. Also, for each of

these areas, provide the information requested in items 1 through 4, as applicable. Update the DCD with this information, as appropriate.

6. If any of the areas that NUREG-0737 specifies should be included for consideration as a vital area (i.e. hydrogen control system, containment isolation reset control area, manual ECCS alignment area, motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels), are not considered as a vital area in the NuScale design, please provide justification for not including them. The justification should include an explanation for why it will not be necessary to access each of the areas following an accident.

NuScale Response:

1. Provide more information regarding where temporary lead equivalent shielding is assumed to be installed as it relates to taking and analyzing liquid and gaseous samples, including access and egress, and the dose rates that will be received in installing this temporary shielding during accident conditions.

The temporary shielding for the liquid and gaseous sampling activities, including access and egress, is assumed to be installed:

- on the short length of exposed sample line penetrating the sample panel shield wall,
- on the drain hose that connects the sample tap to a drain for the short length of exposed sample line penetrating the sample panel shield wall,
- on the drain hose that connects the sample tap to a drain.

The operator dose expected to be received during installation of this temporary shielding during accident conditions is not expected to exceed 5 rem to the whole body or 50 rem to the extremities.

FSAR Section 12.4.1.8 will be revised to provide additional information regarding the location and dose assumptions for temporary shielding installation.

2. Provide post-accident radiation zone maps and ingress and egress routes for all areas associated with performing the vital missions and update the DCD to include this information.

As stated in FSAR Sections 6.4.4, 12.4.1.8, 15.0.0.6.4, and 15.0.4, the NuScale design does not include any credited post-accident operator actions or vital missions.

However, to illustrate the post-accident area dose rates in the areas related to the post-accident sampling contingency action, dose rate maps will be added to the FSAR (Figures 12.3-4a through 12.3-4c) (including ingress and egress) for the time period of the highest post-accident dose rates.

3. Provide information on the time spent at each location and the speed of travel assumed and update the DCD to include this information.

FSAR Section 12.4.1.8 will be revised to provide a more detailed description of the sampling activities, including a new table (Table 12.4-8) that will include the time spent at each location and the dose rates assumed.

4. Provide information on the assumed post-accident airborne activity concentrations and how these source terms were developed. In addition, provide information regarding what assumptions are made regarding submersion and inhalation dose. Update the DCD accordingly.

In the NuScale design, airborne activity in the Reactor Building post-accident would originate from containment leakage and be confined to the air space directly above the reactor pool. Because the air space above the pool is isolated from the rest of the Reactor Building air spaces, the operators in the sampling areas are not subjected to significant airborne activity concentrations.

A clarification statement will be added to FSAR Section 12.4.1.8.

5. Identify any other areas that would require or may require access following an accident to permit an operator to aid in the mitigation of or recovery from an accident. For each of these areas, identify the work that may need to be performed in these areas, provide the dose to perform the work and to access and egress from the area. Also, for each of these areas, provide the information requested in items 1 through 4, as applicable. Update the DCD with this information, as appropriate.

As stated in FSAR Sections 6.4.4, 12.4.1.8, 15.0.0.6.4, and 15.0.4, the NuScale design does not include any credited post-accident operator actions. No additional information is expected to be added in the FSAR.

6. If any of the areas that NUREG-0737 specifies should be included for consideration as a vital area (i.e. hydrogen control system, containment isolation reset control area, manual ECCS alignment area, motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels), are not considered as a vital area in the NuScale design, please provide justification for not including them. The justification should include an explanation for why it will not be necessary to access each of the areas following an accident.

There are no credited post-accident operator actions in the NuScale design. Therefore, none of the areas listed in NUREG-0737 are considered vital areas.

Impact on DCA:

FSAR Section 12.4.1.8 has been revised as described in the response above and as shown in the markups provided with this response.

Additional Information:

A new Table 12.4-8 will be added at the end of Section 12.4 that provides details regarding operator dose assumptions for post-accident sampling activties. New Figures 12.3-4a through 12.3-4c will be added to Section 12.3 depicting the post-accident radiation zones in the Reactor Building in the areas associated with post-accident sampling activities.

RAI 12.03-1

Figure 12.3-4a: Reactor Building Post-Accident Radiation Zone Map - 50' Elevation

Withheld - See Part 9

RAI 12.03-1

Figure 12.3-4b: Reactor Building Post-Accident Radiation Zone Map - 75' Elevation

Withheld - See Part 9

RAI 12.03-1

Figure 12.3-4c: Reactor Building Post-Accident Radiation Zone Map - 100' Elevation

Withheld - See Part 9

Occupational doses are estimated for a single NPM refueling outage and for an entire year, assuming six NPM refueling outages. Table 12.4-7 provides dose estimates for the various refueling activities.

12.4.1.7 Overall Plant Doses

The estimated annual personnel doses associated with the activities discussed above are summarized in Table 12.4-1.

Occupational personnel dose estimates are calculated assuming a 12-NPM site and 24-month fuel cycle for NPM operation, which equates to six refueling outages per year.

12.4.1.8 Post-Accident Actions

RAI 12.03-1

There are no credited post-accident operator actions outside of the main control room for design basis events, as described in Chapter 15. The operator dose assessments for the main control room and the technical support center are provided in Section 15.0.3. As described in Section 9.3.2, the process sampling system may be used as part of a contingency action to obtain post-accident samples, which would potentially expose the operator to post-accident radiation sources. The primary means to detect and monitor fuel damage uses core exit temperature indication and radiation monitors located under the NPM bioshield. The post-accident primary coolant sample is collected via the normal CVCS sample line flow path to the primary system sample panel located in the CVCS gallery. To perform primary liquid sampling, ~~operator access is required to~~operators would access the sample panels in the CVCS gallery and the counting room and hot lab on elevation 50' of the RXB. If the background dose rate in the counting room is too high, ~~operator access to the Annex Building counting room is also required~~operators would use the counting room in the Annex Building. To perform containment gas sampling, ~~operator access may be required to~~operators would access the automatic sample panels in the utilities area on the 100' elevation of the RXB. If calibration of the hydrogen and oxygen analyzers becomes necessary, operators would access ~~and to~~ the steam gallery on the 100' elevation~~, if calibration of the hydrogen and oxygen analyzers becomes necessary~~. If containment gaseous grab samples become necessary, ~~operator access is required to~~operators would access the steam gallery on the 100' elevation of the RXB. These areas are depicted in Figure 12.3-4a through Figure 12.3-4c. Consistent with 10 CFR 50.34(f)(2)(vii), post-accident doses in the CVCS gallery area were evaluated and determined to be less than 4.6 rad/hr, assuming the presence of 0.25" of lead equivalent temporary shielding. Post-accident doses in the counting room and hot lab areas were determined to be less than 100 mrem/hr. The post-accident doses in the steam gallery range were determined to be less than 2.5 mrem/hr. ~~These area post-accident dose rates were evaluated assuming the use of a 0.25" lead equivalent temporary shielding.~~

RAI 12.03-1

In the NuScale design, the potential accident source term from core damage would remain confined within the NPM, unless the decision is made to open a containment isolation valve. If the decision was made to obtain a sample, temporary shielding could be erected prior to opening a containment isolation valve so workers could work in a low dose area. The temporary shielding is assumed to be installed to shield the exposed sample lines near the sample panel and a drain hose that connects the sample tap to a drain.

RAI 12.03-1

The operator's exposure to airborne activity was considered as part of the dose evaluation. Post-accident airborne activity is generated from containment leakage into the Reactor Building atmosphere. The air space above the reactor pool is isolated from the other Reactor Building air spaces, such that the areas accessed by operators to perform sampling activities are not subjected to post-accident airborne contamination.

RAI 12.03-1

The decision to perform post-accident sampling will be determined and initiated by the site staff considering the expected radiological conditions and radiological dose to operating personnel when making the decision to access these areas to perform post-accident sampling. Operators may also utilize temporary shielding during post-accident sampling activities to reduce exposure. The assumed parameters for the post-accident sampling operator dose evaluation are provided in Table 12.4-8. Post-accident sampling performed consistent with approved procedures prevents radiation exposures to individuals from exceeding 5 rem to the whole body or 50 rem to the extremities, consistent with 10 CFR 50.34(f)(2)(viii).

12.4.1.9 Construction Activities

For the construction of an additional NuScale Power Plant adjacent to an existing NuScale Power Plant, the estimated annual radiation exposure to a construction worker is estimated based upon a construction staffing plan over the estimated construction period. It is estimated that the annual dose for a construction worker is 1.64 mrem/year.

COL Item 12.4-1: A COL applicant that references the NuScale Power Plant design certification will estimate doses to construction personnel from a co-located existing operating nuclear power plant that is not a NuScale Power Plant.

12.4.2 Radiation Exposure at the Site Boundary

The direct radiation to the site boundary from on-site sources, such as buildings, is negligible.

RAI 12.03-1

Table 12.4-8: Post-Accident Sampling Operator Dose

Activity Description	Duration (minutes)	Dose Rates (mrem/hr)	ORE (man-rem)	Dose Rate Notes
Post-Accident Containment Sampling				
hydrogen analyzer calibration	20	1.55	5.17E-04	Dose rate from CNV vapor source term under bioshield attenuated through pool wall at 100' elevation.
oxygen analyzer calibration	20	1.55	5.17E-04	Dose rate from CNV vapor source term under bioshield attenuated through pool wall at 100' elevation.
Containment Gas Sampling & Analysis	MCR	0.00E+00	0.00E+00	The hydrogen oxygen analyzers are connected to the module control system providing readout in the main control room, with no local operator action required for the analysis.
Total	**40**	**1.55**	**1.03E-03**	
Post-Accident Reactor Coolant Sampling				
Stage sample purge collection tank (if required) and tank shielding	60	3.01E-02	3.01E-05	Center line dose rate from the CVCS heat exchangers from normal operations RXB dose rates.
Stage sample vessel with shielding cask and transport cart	5	9.32E+01	7.77E-03	Retrieved from Hotlab.
Connect sample line drain hose to sample tap, and connect drain hose to either a collection tank or the liquid radioactive waste system	2	3.01E-02	1.00E-06	Center line dose rate from the CVCS heat exchangers from normal operations RXB dose rates.
Place temporary shielding blankets on sample line and drain hose	20	3.01E-02	1.00E-05	Center line dose rate from the CVCS heat exchangers from normal operations RXB dose rates.
Build temporary shield wall in from of sampling panel and drain line to allow reduce exposure for ingress and egress to the sample panel.	60	3.01E-02	3.01E-05	Center line dose rate from the CVCS heat exchangers from normal operations RXB dose rates.
Open the sample panel hand valve	0.5	3.01E-02	2.50E-07	Center line dose rate from the CVCS heat exchangers from normal operations RXB dose rates.
Open the CVCS letdown line isolation valve	MCR	0.00E+00	0.00E+00	There is expected to no additional dose from this action, as it is expected to be performed from the main control room by operators monitoring and controlling plant activities.
Purge the sample line	Remote	0.00E+00	0.00E+00	This action is not expected to require local supervision; therefore it is assumed that this will not result in additional dose.

Table 12.4-8: Post-Accident Sampling Operator Dose (Continued)

Activity Description	Duration (minutes)	Dose Rates (mrem/hr)	ORE (man-rem)	Dose Rate Notes
Close sample panel hand valve	0.5	1.77E+03	1.48E-02	Dose rates at 2ft from the sample panel based on the expectation that all actions for the reactor coolant sample collection to be performed with some form of reach tool (tongs, and reach rods for valve actuation), in an effort the keep whole body doses ALARA.
Disconnect the drain hose from sample tap and cap the drain hose	2	1.77E+03	5.90E-02	Dose rates at 2ft from the sample panel based on the expectation that all actions for the reactor coolant sample collection to be performed with some form of reach tool (tongs, and reach rods for valve actuation), in an effort the keep whole body doses ALARA.
Connect sample vessel to sample tap at the sample panel	2	1.77E+03	5.90E-02	Dose rates at 2ft from the sample panel based on the expectation that all actions for the reactor coolant sample collection to be performed with some form of reach tool (tongs, and reach rods for valve actuation), in an effort the keep whole body doses ALARA.
Open sample panel hand valve	0.5	1.77E+03	1.48E-02	Dose rates at 2ft from the sample panel based on the expectation that all actions for the reactor coolant sample collection to be performed with some form of reach tool (tongs, and reach rods for valve actuation), in an effort the keep whole body doses ALARA.
Monitor sample vessel collection volume	2	1.77E+03	5.90E-02	Dose rates at 2ft from the sample panel based on the expectation that all actions for the reactor coolant sample collection to be performed with some form of reach tool (tongs, and reach rods for valve actuation), in an effort the keep whole body doses ALARA.
Close sample panel hand valve	0.5	1.77E+03	1.48E-02	Dose rates at 2ft from the sample panel based on the expectation that all actions for the reactor coolant sample collection to be performed with some form of reach tool (tongs, and reach rods for valve actuation), in an effort the keep whole body doses ALARA.
Close the CVCS letdown line isolation valve	MCR	0.00E+00	0.00E+00	There is expected to no additional dose from this action, as it is expected to be performed from the main control room by operators monitoring and controlling plant activities.
Place sample vessel in shielded sample cask on transport cart	2	1.77E+03	5.90E-02	Dose rates at 2ft from the sample panel based on the expectation that all actions for the reactor coolant sample collection to be performed with some form of reach tool (tongs, and reach rods for valve actuation), in an effort the keep whole body doses ALARA.
Transport sample to the hot lab	5	9.32E+01	7.77E-03	This dose rate is from the CNV vapor phase photon shine through the pool wall into the hot lab.
Process sample	60	9.32E+01	9.32E-02	This dose rate is from the CNV vapor phase photon shine through the pool wall into the hot lab.

Table 12.4-8: Post-Accident Sampling Operator Dose (Continued)

Activity Description	Duration (minutes)	Dose Rates (mrem/hr)	ORE (man-rem)	Dose Rate Notes
Perform required analysis	240	0.00E+00	0.00E+00	Radiation counting performed in Annex Building in radiological count room.
Total			**3.89E-01**	