NuScale Power

RAIO-0518-59929

May 16, 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 Response to NRC Request for Additional Information No. 350 (eRAI No. 9278) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 350 (eRAI No. 9278)," dated January 29, 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 Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 9278:

- 12.03-31
- 12.03-32
- 12.03-33
- 12.03-34
- 12.03-35
- 12.03-36
- 12.03-37
- 12.03-38
- 12.03-39
- 12.03-40

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

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9278, nonpublic

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9278, public

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9278, nonpublic Security-Related Information - Withhold Under 10 CFR §2.390

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Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9278, public

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-31

Regulatory Basis

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.

10 CFR 50.49 and 10 CFR Part 50, Appendix A, Criterion 4 require that certain components important to safety be designed to withstand environmental conditions, including the effects of radiation, associated with design basis events, including normal operation, anticipated operational occurrences, and design basis accidents.

NUREG-0737 and DSRS section 12.3-12.4 provide additional guidance on acceptable methods of meeting these requirements. These documents indicate that post accident radiation zones 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. RAI-8775 Question 12.03-1 focused on the sample station, sample analysis, and other areas requiring infrequent access. NUREG-0737 provides a list of other areas that should be considered in determining the vital areas and stipulates that if these areas are not considered vital areas, justification should be provided for not including them. Areas specified include the containment isolation reset control area, motor control centers, instrument panels, emergency power supplies, 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.

Background

The applicant's response to RAI-8775 Question 12.03-1, dated June 26h 2017, provided a new proposed DCD Table 12.4-8: "Post- Accident Sampling Operator Dose," which provides a time line and estimated dose for steps of the sampling process. The response provided proposed Figures 12.3-4a-12.3-4c depicting reactor building post-accident radiation zone maps for the 50', 75' and 100' elevations. The response also proposed changes to DCD Section 12.4.1.8 "Post-Accident Actions."

Key Issue 1:

DCD Tier 2 Section 9.3.2.2.3 "System Operation," states that the sample line purge fluid may be collected in a temporary disposal tank if radiation levels are expected to be too high for disposal to the LRWS. Proposed Table 12.4-8: "Post-Accident Sampling Operator Dose," states that a purge collection tank may be used to collect water from the sample line. The table also states that ¼" of temporary lead equivalent shielding material may be staged to reduce dose rates during sampling. Based on need to purge multiple line volumes in order to ensure a representative sample is obtained, and operating experience from Three Mile Island (TMI) regarding the specific activity of reactor coolant system (RCS) fluids following an accident, the temporary surge tank is a significant source of radiation exposure that is not reflected in the post-accident dose calculations.

Question 1

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to estimated radiation exposure for accessing areas following an accident, the staff requests that the applicant:

- As necessary, revise proposed Table 12.4-8: "Post-Accident Sampling Operator Dose," to include the dose from temporary purge collection tank, or provide the specific alternative approaches used and the associated justification.

OR

Provide the specific alternative approaches used and the associated justification

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NuScale Response:

As described in the NuScale response to RAI 9.3.2-8 (eRAI# 9044), the potential use of a temporary purge collection tank is an alternate option requiring the COL applicant to develop and evaluate the use of such a tank for post-accident sampling operations (COL item 9.3-2). The FSAR Section 12.4.1.8 and Table 12.4-8 have been revised to remove all reference to a temporary purge collection tank, because it is not permanent plant equipment, but is a COL item.

Impact on DCA:

FSAR Section 12.4.1.8 andTable 12.4-8 have been revised as described in the response above and as shown in the markup provided with the response to question 12.03-37.

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-32

The Regulatory Basis and Background are in RAI-9278 Question 31017

Question 2
To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to radiation levels in areas requiring access following an accident, the staff requests that the applicant:

- As necessary, revise DCD Section 12.3 to include figures depicting post-accident radiation zone maps of areas containing pipes or components that may contain highly radioactive fluid resulting from sampling activities;

And,

- Explain/Justify any changes needed to support the qualification of equipment (e.g., DCD Table 3C-1),
- As necessary, revise the DCD to include the description of the changes needed to support the qualification of equipment (e.g., DCD Table 3C-1),

Or,

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

NuScale has revised its post-accident radiation zone maps (FSAR Figure 12.3-4a through 12.3-4d) to include post-accident primary coolant source terms in appropriate components and pipes.

The post-accident radiation zone maps represent a composite of maximum dose rates developed by using the highest dose rate in a particular Reactor Building area associated with the range of design basis accidents occurring in the module (or a fuel handling accident) that results in the highest calculated dose rate. FSAR Section 12.4.1.8 has been revised to include this description.

A discussion of impacts to equipment qualification and FSAR Chapter 3 can be found in the NuScale response to RAI 8837.

Impact on DCA:

FSAR Section 12.4.1.8 and FSAR Figures 12.3-4a through 4d have been revised as described in the response above and as shown in the markup provided with the response to question 12.03-37.

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-33

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 3:

Proposed changes to DCD Section 12.4.1.8 “Post-Accident Actions,” states 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. However, the staff was unable to identify any design features on Figure 12.3-4c: “Reactor Building Post-Accident Radiation Zone Map - 100' Elevation,” that would serve to isolate the air space in this area of the facility from the air spaces in other portions of the facility.

Question 3

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to airborne radioactivity levels in areas requiring access following an accident, the staff requests that the applicant:

- Explain/Justify the design features relied upon to provide isolation of the air space above the ultimate heat sink pool from other areas of the Reactor Building (RXB)
- As necessary, revise DCD Section 12.3 to include a description of those design features relied upon to provide isolation of the air space above the ultimate heat sink pool from other areas of the Reactor Building (RXB),

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

The Reactor Building contains concrete walls that separate the airspace above the reactor pool from the other areas of the RXB. These walls use airtight seals on doors and other penetrations between the pool area and the other portions of the building to minimize the migration of airborne contamination. The Reactor Building HVAC system uses smoke dampers to minimize leakage between the pool area and other portions of the building.

Therefore, the operator's exposure to airborne activity is not expected to result in significant doses. The description in FSAR Section 12.4.1.8 has been revised accordingly.

Impact on DCA:

FSAR Section 12.4.1.8 has been revised as described in the response above and as shown in the markup provided with the response to question 12.03-37.

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-34

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 4:
Proposed changes to DCD Section 12.4.1.8 “Post-Accident Actions,” states 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. However, the response does not address potential airborne activity and the resultant personnel exposure, resulting from the purging and sampling activities.

Question 4
To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to airborne radioactivity levels resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used to determine the exposure to the operator from airborne radioactive material during resulting from sample purging and acquisition, following an accident,
- As necessary, revise DCD Section 12.3 to include the assessed dose resulting from airborne activity radioactive material evolving from purging or sampling fluids,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

Post-accident sampling can only be performed if the chemical volume and control system (CVCS) and process sampling system (PSS) are intact and operational. An initial operator action prior to conducting a post-accident sample is to enter the RXB, evaluate the radiological conditions, and determine if the sampling system condition will allow sampling to be conducted. The conditions under which a sampling operator could be exposed to significant airborne contamination result from primary coolant leaks in the 50' elevation gallery. Such conditions would be investigated using appropriate protective equipment in conjunction with existing fixed and mobile radiation monitors and detectors.

Because the temperature and pressure of the primary coolant in the sampling system is controlled, the generation of an airborne source term in the CVCS gallery during sampling is minimized. Also, the use of personnel respiratory equipment and protective clothing is used, if radiological conditions warrant. Therefore, the operator's exposure to airborne activity is not expected to result in significant doses. The description in FSAR Section 12.4.1.8 has been revised accordingly.

Impact on DCA:

FSAR Section 12.4.1.8 has been revised as described in the response above and as shown in the markup provided with the response to question 12.03-37.

Response to Request for Additional Information
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eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-35

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 5:
DCD Tier 2 Revision 0 Section 9.3.2.2.3 “System Operation,” states that the post-accident water level in the reactor vessel for some events may be below the level of the normal CVCS reactor vessel nozzle (for reactor coolant discharge to CVCS). The reactor coolant post-accident sample is collected via the CVCS sample line flow path to the primary sampling system sample panel. DCD Section 9.3.2.2.3 also states that RCS pressure is used, as in normal operation, to force the RCS sample to the sample panel. At temperatures below 200 degrees Fahrenheit with insufficient RCS pressure, a nitrogen overpressure can be established. However, the proposed changes to DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” do not include dose estimates for the activities of adding water to the reactor, or raising the pressure of the containment vessel.

Question 5
To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to personnel exposure to radiation resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used to assess the dose resulting from adding water to the reactor and raising containment pressure, as needed, to support sample acquisition following an accident,
- As necessary, revise DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” to include the assessed dose resulting from adding water to the reactor, and raising containment pressure,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

There is no additional operator dose attributed for these activities because the activities related to adding water to the reactor and raising the pressure of the containment vessel are performed from the main control room.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-36

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 6:
Based on information made available to the staff during the RPAC NuScale Chapter 12 Audit, the staff became aware that the methodology used to develop the photon source strength from the post-accident fluid did not account for some principle radiation emitting isotopes in the fluid stream. For instance, Ba-137m is in secular equilibrium with the parent Cs-137 radionuclide, the specific activity of Ba-137m should be within 94 percent of the Cs-137 specific activity, within 20 minutes. However, the information reviewed by the staff indicated that the source term used by the applicant to perform the analysis of dose resulting from the sample fluid, did not properly account for Ba-137m, thus resulting in an underestimation of the dose rate from the sample fluid.

Question 6

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to personnel exposure to radiation resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used to determine the dose rates from systems, structures and components containing reactor coolant fluids following an accident,
- As necessary, revise DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” to describe the necessary assumptions used to reflect the correct source term specific activity, and the resultant dose rates and the doses to operator(s),

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

NuScale has revised its engineering calculations and analyses to model the decay of radionuclides involving isotopes that are in secular equilibrium, including Ba-137m and Cs-137. The resultant changes to the post-accident sampling operator dose have been incorporated, along with other changes, into FSAR Section 12.4.1.8, Table 12.4-8, and Figures 12.3-4a through 12.3-4d.

Impact on DCA:

FSAR Section 12.4.1.8, Table 12.4-8, and Figures 12.3-4a through 12.3-4d have been revised as described in the response above and as shown in the markup provided with the response to question 12.03-37.

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-37

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 7:
The dose calculations performed to support the specific task dose information provided in proposed DCD Table 12.4-8: “Post- Accident Sampling Operator Dose,” assumed that fluid contained within the CVCS heat exchangers was at the normal RCS fluid specific activity. However, because DCD Tier 2 Revision 0 Figure 9.3.4-1: “Chemical and Volume Control System Diagram,” does not show any design features to prevent flow through the CVCS heat exchanger once the containment isolation valves are opened, and proposed DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” does not include an activity to prevent flow of post-accident radioactive fluid through the CVCS heat exchangers, the staff is unable to determine if the dose estimate should include dose from the CVCS heat exchangers containing post-accident fluids.

Question 7

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to personnel exposure to radiation resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used for determining the radiation exposure to operators from SSCs containing reactor coolant system fluids following an accident,
- As necessary, revise the proposed DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” dose rates and the resultant operator doses, to reflect the activities to prevent flow of post-accident fluids through the CVCS heat exchanger,

OR

- As necessary, revise proposed DCD Table 12.4-8 to include dose from the CVCS heat exchanger containing post- accident fluid,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

To model the estimated dose to an operator taking a post-accident primary sample, the dose rate from the CVCS heat exchangers filled with post-accident source term primary fluid is conservatively treated as equal to the dose rate from a pool of post-accident source term primary water on the floor inside the CVCS heat exchanger room from a small line break. This model is conservative because the assumed pool of water contains a larger amount of radioactivity than the activity that could be contained within the CVCS heat exchangers , the pool of water is closer to the dose receptor, and does not credit shielding from the CVCS heat exchanger steel shell and tubes. The primary coolant post-accident source term is modeled using the Technical Specification coolant activity, plus a coincident iodine spike. Updates to FSAR Table 12.4-8 have incorporated the additional operator dose contribution from the CVCS heat exchangers, as described above.

Impact on DCA:

FSAR Table 12.4-8 has been revised as described in the response above and as shown in the markup provided with this response.

RAI 12.03-1, RAI 12.03-31, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35

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

{{ Withheld - See Part 9 }}

RAI 12.03-1, RAI 12.03-31. RAI 12.03-33, RAI 12.03-34, RAI 12.03-35

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

{{ Withheld - See Part 9 }}

RAI 12.03-1, RAI 12.03-31, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35

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

{{ Withheld - See Part 9 }}

RAI 12.03-1, RAI 12.03-31, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35

Figure 12.3-4d: Reactor Building Post-Accident Radiation Zone Map - 126' 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, RAI 12.03-31, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35

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, operators would enter the RXB at the 100′ elevation, descend to the 50′ elevation using a stairwell, and traverse to 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, operators would use the counting room in the Annex Building. These areas are depicted in Figure 12.3-4a through Figure 12.3-4d. These post-accident radiation zone maps represent a composite of maximum dose rates developed by using the highest dose rate in a particular Reactor Building area resulting from design basis accidents occurring on the module with the highest calculated dose rate. Therefore, the radiation zones depicted in Figure 12.3-4a through Figure 12.3-4d will not occur simultaneously. For post-accident sampling operator activities, the limiting design basis accident is a break of a small line carrying primary coolant, with a coincident iodine spike. This also assumes that the sampling system has not been rendered inoperable due to the accident. For example, a small line break on a CVCS line upstream of the sample line tap causes the sampling system to be non-functional. Consistent with 10 CFR 50.34(f)(2)(vii), post-accident radiological conditions were evaluated and determined that primary coolant sampling activities expose operators to dose rates up to 70 mrem/hr at the sample panel, with much of the collection activities resulting in dose rates less than 13 mrem/hr. Post-accident doses in the counting room and hot lab were determined to be less than 2.5 mrem/hr. To perform containment gas sampling, operators would perform the necessary functions from the main control room. ~~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, operators would use the counting room in the Annex Building. To perform containment gas sampling, 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 the steam gallery on the 100' elevation. If containment gaseous grab samples become necessary, 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 one-quarter inch 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.~~

RAI 12.03-1, RAI 12.03-31, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35

~~In the NuScale design, the potential accident source term from core damage remains confined within the NPM unless the decision is made to open a containment isolation valve. If the decision is made to obtain a sample, temporary shielding can be erected prior to opening a containment isolation valve so workers can 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, RAI 12.03-31, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35

The operator's exposure to airborne activity is considered as part of the dose evaluation, but is not expected to result in significant doses. This is due to the barrier created by the RXB walls around the reactor pool, which include airtight seals for doors and other penetrations, between the reactor pool area and other areas in the RXB. This barrier minimizes the migration of airborne contamination from the airspace above the pool to other areas of the RXB. The RXB HVAC system uses smoke dampers to minimize the leakage between the pool area and other portions of the building. In addition, the generation of airborne contamination is minimized in the CVCS gallery area through the control of the temperature and pressure of the primary coolant during sampling. Also, if radiological conditions warrant, respiratory protection equipment can be provided to post-accident sampling personnel. ~~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, RAI 12.03-31, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35

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. T~~A summary of 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).

RAI 12.03-31, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35

Table 12.4-8: Post-Accident Sampling Operator Dose

Activity Description	Duration (minutes)	Average Dose Rates (mrem/hr)	ORE (man-rem)
Investigative survey and isolation of CVCS down stream of PSS line	12.5	3.16	6.59E-04
Ingress for sampling event	6	2.93	2.93E-04
Reactor coolant sample preparation	7	8.78	1.02E-03
Reactor Coolant Sample Collection	40	15.9	1.06E-02
Reactor coolant sample transport	5	11.1	9.21E-04
Reactor coolant sample processing	60	0.522	5.22E-04
Reactor coolant sample analysis	240	0.522	2.09E-03
Egress following sample	6	3.69	3.96E-04
Total	**377**		**1.65E-02**

RAI 12.03-1, RAI 12.03-31, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35

~~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.~~
~~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 front of sampling panel and drain line to allow reduced 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~~	~~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.~~
~~Close sample panel hand valve~~	~~0.5~~	~~1.77E+03~~	~~1.48E-02~~	~~Dose rates at 2 ft from the sample panel based on the expectation that reactor coolant sample collection will be performed with reach tools (tongs and reach rods for valve actuation) to keep whole-body doses ALARA.~~

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

Activity Description	Duration (minutes)	Dose Rates (mrem/hr)	ORE (man-rem)	Dose Rate Notes
Disconnect the drain hose from sample tap and cap the drain hose	2	1.77E+03	5.90E-02	Dose rates at 2 ft from the sample panel based on the expectation that reactor coolant sample collection will be performed with reach tools (tongs and reach rods for valve actuation) to 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 2 ft from the sample panel based on the expectation that reactor coolant sample collection will be performed with reach tools (tongs and reach rods for valve actuation) to keep whole-body doses ALARA.
Open sample panel hand valve	0.5	1.77E+03	1.48E-02	Dose rates at 2 ft from the sample panel based on the expectation that reactor coolant sample collection will be performed with reach tools (tongs and reach rods for valve actuation) to keep whole-body doses ALARA.
Monitor sample vessel collection volume	2	1.77E+03	5.90E-02	Dose rates at 2 ft from the sample panel based on the expectation that reactor coolant sample collection will be performed with reach tools (tongs and reach rods for valve actuation) to keep whole-body doses ALARA.
Close sample panel hand valve	0.5	1.77E+03	1.48E-02	Dose rates at 2 ft from the sample panel based on the expectation that reactor coolant sample collection will be performed with reach tools (tongs and reach rods for valve actuation) to keep whole-body doses ALARA.
Close the CVCS letdown line isolation valve	MCR	0.00E+00	0.00E+00	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 2 ft from the sample panel based on the expectation that reactor coolant sample collection will be performed with reach tools (tongs and reach rods for valve actuation) to 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.
Perform required analysis	240	0.00E+00	0.00E+00	Radiation counting performed in Annex Building in radiological count room.
Total			**3.89E-01**	

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-38

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 8:
DCD Tier 2 Revision 0 Section 9.3.2.2.3 “System Operation,” and proposed DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” indicate the need for special tools, such as reach rods for remote operation of local sample valves, mechanical disconnects for sample tubing connections and the associated remote disconnect tools, and shielded carts for sample transport. The Acceptance Criteria of DSRS 12.3-12.4 specifically mentions provisions for portable shielding and remote handling tools. Some special tools, like those for handling tubing connectors, are not routinely available, and the use of routinely available remote handling tools, such as tongs or pliers may significantly extend the amount of time in the area. Also, while transport carts for radioactive filters and other large components are expected to be available for use, they are ill suited for the task of transporting samples However, there is no COL Item that directs the COL Applicant to provide these types of items.

Question 8

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to design features and tools expected to be available to reduce personnel exposure to radiation resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the types, number, and any special features, of the tools and shielded transport cart, expected to be used for reducing radiation exposure to operators obtaining samples following an accident,
- As necessary, revise DCD Section 12.4.1.8 “Post-Accident Actions,” to describe the tools and equipment, that were assumed to be readily available and used to take samples of reactor coolant fluids following an accident,
- As necessary, revise DCD Section 12.4.1.8 to add a COL Item to provide the special tools and shielded transport cart needed to support post-accident sampling,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

As a result of clarification discussions with the NRC staff, this question has been resolved.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-39

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 9:
DCD Tier 2 Revision 0 Section 9.3.2.2.3 "System Operation," and proposed DCD Section 12.4.1.8 "Post-Accident Actions," indicate that the post-accident primary coolant sample is collected via the normal CVCS sample line flow path to the primary system sample at a panel located in the CVCS gallery, but appears to indicate that the sample does not flow through the normal Plant Sampling System (PSS) sample panel. Because DCD Section 9.3.2.2.2 "Component Description," discusses first and second stage cooling for samples streams over approximately 100 degrees Fahrenheit but also notes that second stage cooling is provided when the sample stream is directed to an analysis panel. Proposed DCD Table 12.4-8: "Post-Accident Sampling Operator Dose," does not include expected dose from post-accident fluid contained in sample coolers.

Question 9

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to design features protect personnel, and reduce personnel exposure to radiation to personnel, resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the methods and assumptions used to determine the conditions (e.g., temperature, pressure, volatile gas content,) of reactor coolant system fluids present at the sampling point, following an accident,
- As necessary, revise the DCD to include a description of how post-accident sample streams that could potentially be over 100 degrees Fahrenheit will be cooled to a low enough temperature to allow for safe purge volume collection, and sample collection while minimizing airborne radioactive material release from vaporization;

And,

- As necessary, revise the proposed DCD Table 12.4-8: "Post-Accident Sampling Operator Dose," dose rates, and the resultant doses, to reflect the dose from the post-accident fluids in the sample cooler(s),

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

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 area on elevation 50' of the Reactor Building. While the sample panel in the CVCS gallery is used for continuous sampling and online hydrogen and oxygen analysis of reactor coolant during normal operation, it also includes capability to collect reactor coolant grab sample for post-accident sampling. The use of the sample panel in the CVCS gallery area for post-accident sampling instead of the sample panel in the chemistry hot lab (used for grab sample collection during normal operation) minimizes the potential for high radiation in the hot lab area. The flowpath of the post-accident sample to the sample panel in the CVCS gallery area includes routing the sample stream through both the first and second stage sample coolers located upstream of the sample collection point. Therefore, the sample fluid will be sufficiently cooled (i.e., below 100 ºF).

Shielding of the sample panel is provided by routing radioactive components (e.g., sample coolers) behind a 20" thick concrete/steel partition wall. Because the design of the primary sample panel sample coolers has not been finalized, and the specific cooler dimensions are not known at this time, the post-accident dose rate to an operator is conservatively modeled as the dose rate from an additional 2.5" diameter CVCS letdown line containing post-accident primary fluid.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

NUSCALE POWER™

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-40

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 10:
DCD Tier 2 Revision 0 Section 9.3.2.2.3 “System Operation,” and proposed DCD Section 12.4.1.8 “Post-Accident Actions,” states that the sample line purge volume may be directed to the liquid radioactive waste system (LWRS). DCD Figure 9.3.3-1: “Radioactive Waste Drain System Diagram,” shows that the drains in the Reactor Building (RXB) go to sumps 41 A/B – 46 A/B. As noted in DCD Section 9.3.2.2.3, additional operator actions, such as raising containment pressure, may be required to allow sampling. Also, as noted in DCD Section 9.3.2.2.3, sample fluids greater than 100 degrees Fahrenheit may require cooling. There is no discussion in DCD Section 9.3.2.2.3 or the response to RAI-8775 Question 12.03-1 that addresses any actions needed to assure proper operation of the these systems during post-accident conditions. As noted above, NUREG-0737 provides a list of other areas that should be considered in determining the vital areas and stipulates that if these areas are not considered vital areas, justification should be provided for not including them.

Question 10

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to the levels of radiation that may be present in the facility resulting from taking samples of reactor coolant system fluids following an accident, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used to assess the radiological impact on area dose rates, in other areas of the plant (e.g., radioactive waste collection systems,) resulting from acquiring samples of reactor coolant system fluids following an accident,
- As necessary, revise proposed DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” dose rates, and the resultant doses, to reflect the dose from accessing those areas,

Or,

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

There are no areas that require operator access because the NuScale design does not include any credited post-accident operator actions for design-basis accidents. However, to comply with 10 CFR 50.34(f)(2)(viii), NuScale has demonstrated the capability to obtain and analyze post-accident samples from the reactor coolant system and containment without exceeding the stated dose criteria. As part of this demonstration, accessibility to areas of the facility for performing post-accident sampling have been evaluated and included in the operator dose analysis, as described in FSAR Section 12.4.1.8 and Table 12.4-8. The areas potentially impacted by radioactive drains from sampling activities do not require post-accident operator access.

Impact on DCA:

There are no impacts to the DCA as a result of this response.