NuScaleDCRaisPEm Resource

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Subject: Request for Additional Information No. 205, RAI 9044 (9.3.2)
Attachments: Request for Additional Information No. 205 (eRAI No. 9044).pdf

Attached please find NRC staff's request for additional information concerning review of the NuScale Design Certification Application.

Please submit your technically correct and complete response within 60 days of the date of this RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

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Request for Additional Information No. 205 (eRAI No. 9044)

Issue Date: 09/01/2017
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Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 09.03.02 - Process and Post-Accident Sampling Systems
Application Section: 9.3.2

QUESTIONS

09.03.02-1

Regulatory Requirements:

10 CFR 50.34(f)(2)(xxvi) requires that applicants "provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency." DSRS Section 9.3.2 specifies that to prevent unnecessarily high exposures to workers and the public and to maintain control and use of the systems during an accident, a program should be implemented to minimize leakage from the sampling system to be as low as practical levels. The DSRS also specifies that excessive leakage will not prevent the use of the system under accident conditions and that the leakage control program will include measures to minimize the leakage from the systems.

Key Issue: The application does not provide sufficient details and clarity to determine if the potential leakage rate is minimized to be as low as practical

NuScale COL Item 9.3-1 specifies that a COL applicant will provide a leakage control program, including an initial test program, to satisfy the programmatic aspects of this requirement. It is acceptable to leave this programmatic responsibility to the COL applicant; however, the staff notes that most of the equipment associated with the sampling system (including most portions of the CVCS and CES) are non-safety related and non-seismically qualified (See DCD Table 3.2-1, "Classification of Structures, Systems, and Components"). In addition, design aspects that facilitate meeting this requirement are generic and appropriate to be addressed as part of a standard design certification. In order to take samples following an accident, override of the containment isolation valves in the CVCS and CES systems is required to open the valves and take samples. If these valves are opened and there is a leak or failure in these non-safety related and non-seismically qualified systems, accident source term could be released outside of the containment into the reactor building and, potentially the environment.

Requested Additional Information:

1. Explain the leakage control and detection capabilities in the design of these systems as required by 10 CFR 50.34(f)(2)(xxvi). Including how the design ensures that leakage is controlled and quickly detected and mitigated so that doses to workers and the public are acceptable.

2. Please update the DCD to provide additional information regarding how the design minimizes a potential radioactive release and worker exposure in this regard and how it adequately allows for the use of systems needed in an emergency in accordance with the requirements of 10 CFR 50.34(f)(2)(xxvi).

09.03.02-2

Regulatory Requirements and Guidance:

10 CFR Part 50, Appendix A, General Design Criterion 64, requires that means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

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. In addition, NUREG-0737 recommends prompt sampling under accident conditions.

DSRS Section 9.3.2 states that the primary review organization and the organization responsible for the review of radiation protection verify that provisions are made for purging sampling lines and for reducing plateout in sample lines (e.g., with heat tracing).

Key Issue: There is not enough information how heat tracing will be implemented in the design.

DCD Section 9.3.2.2.3, under "Containment Gas Post-Accident Monitoring Sampling," the applicant indicates that the PSS piping is heat traced to prevent the build-up of condensate within the containment gas monitoring lines and analyzer to ensure monitoring capability under accident conditions.

Requested Additional Information:

It is unclear how heat tracing will be implemented in the design, e.g., what power supply will be used to heat trace the sample lines. If the heat tracing is not supplied from a reliable power supply and the power supply is non-functional following a design basis accident, it is unclear how it can be ensured that reliable samples will be able to be taken. Please provide additional information regarding which power supply will be used to perform heat tracing and how it can be assured that heat tracing can be performed following a design basis accident.

09.03.02-3

Regulatory Requirements:

10 CFR Part 50, Appendix A, General Design Criterion 64, requires that "means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents."

10 CFR 50.34(f)(2)(vii) requires the performance of radiation shielding design reviews to ensure the design permits adequate access to important areas and provides for protection of safety equipment from radiation, following an accident. DSRS Section 12.3-12.4, references this requirement and the associated NUREG-0737, Section II.B.2, which provides additional guidance on meeting this requirement.

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." In addition, NUREG-0737 recommends prompt sampling under accident conditions.

Key Issues: The application does not have sufficient detail and clarity to determine if and how gaseous samples will be obtained post-accident and that applicable requirements will be met.

DCD Section 9.3.2 indicates that post-accident sampling of containment gas is possible in the NuScale design and "would be used for post-accident sampling only if the information sought is essential and cannot be determined or estimated by other means". However, the DCD is unclear and inconsistent regarding how post accident gaseous samples of the containment atmosphere will be obtained.

In DCD Section 9.3.2.2.3, under "Off-Normal Operations", it states, "The CES (containment evacuation system) is a low pressure system not designed for full containment design pressure and has not been provided with override capability. Accident simulations project that in approximately 24 hours following a containment isolation initiation, RCS temperatures will fall below 200 degrees Fahrenheit, permitting the opening of the containment evacuation system CIVs to support sampling at that time, if necessary."

While in Section 9.3.2.2.3, under "Containment Gas Post-Accident Monitoring and Sampling" it states, "Plant conditions amenable to plant sampling exist within 2 hours of the most limiting design basis event, and will require override of the CNV (containment vessel) containment isolation valves for the CES and CFDS (containment flooding and drain system)." Furthermore, it also states, "the CNV isolation valves for CES and CFDS are opened to establish the monitoring and sampling flow paths. A manual logic override is required to open the CNV isolation valves if RCS temperature is greater than 200 degrees F and containment parameters are greater than the containment isolation setpoints."

Requested Additional information:

Based on the above information and apparent inconsistencies, please address the following.

1. It is unclear to staff at what time after an accident and under what conditions, containment gaseous samples are capable of being taken. It is also unclear if the isolation valves for the CES are provided with override capability or not. Please provide this information and update the DCD as appropriate to correct any inconsistencies.

2. It is unclear which valves are required to be opened to take gaseous samples (only the CES or both the CES and CFDS?). Please clarify which valves need to be opened. If both CES and CFDS valves need to be opened to take gaseous samples, please clarify why the isolation valves for the CFDS (which goes to a part of the containment vessel that is expected to be submerged following an accident), needs to be opened to obtain a gas sample. Update the DCD as appropriate.

3. Likewise, it is unclear if the systems are appropriately designed to handle the temperatures and pressures that will be present. DCD Section 9.3.6, "Containment Evacuation System and Containment Flooding and Drain System," does not specify the design limitations of the system. It is not clear if any relief valves are provided and at what pressure such relief valves would actuate (a significant release into the Reactor Building could occur, even if the piping were still intact, if a relief valve lifted, or a seal was damaged by heat). Please clarify the design limitations of the CES and CFDS systems and if the CES and CFDS systems downstream of the containment isolation valves are capable of withstanding the temperatures and pressures present 2 hours after an accident or if approximately 24 hours and less than 200 degrees Fahrenheit is required to open these valves. Update the DCD as appropriate.

4. It is unclear if appropriate equipment and power will be available to manually override and open valves to take samples during accident condition. Please describe the process and equipment that will be needed to re-open these valves and if this equipment is ensured to be operational following a design basis accident. Update the DCD as appropriate. Is this equipment operable from the Main Control Room, or is operator action in the field required? Is AC electrical power required to open these valves? How is it ensured that the required equipment can be appropriately operated following a design basis accident?

09.03.02-4

Regulatory Requirements:

10 CFR Part 50, Appendix A, General Design Criterion 61, requires, in part, that "systems which contain radioactivity shall be designed to assure adequate safety under postulated accident conditions, including with appropriate containment, confinement, and filtering systems."

10 CFR Part 50, Appendix A, General Design Criterion 64, requires that "means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents."

10 CFR 50.34(f)(2)(vii) requires the performance of radiation shielding design reviews to ensure the design permits adequate access to important areas and provides for protection of safety equipment from radiation, following an accident. DSRS Section 12.3-12.4, references this requirement and the associated NUREG-0737, Section II.B.2, which provides additional guidance on meeting this requirement.

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." In addition, NUREG-0737 recommends prompt sampling under accident conditions.

Key Issues: The application doesn't have sufficient details and clarity with regard to sampling RCS fluids following an accident while meeting the dose limits to workers.

DCD Section 9.3.2.2.3, under "Reactor Coolant Post-Accident Sampling," states that at temperatures below 200 degrees Fahrenheit with insufficient RCS pressure, a nitrogen overpressure can be established to take reactor coolant samples.

DCD Tier 2 Revision 0, Figure 9.3.1-1, "Instrument Air and Service Air System Diagram," shows that the compressed air system (in addition to the nitrogen gas system) can provide gas to the Containment Evacuation System. It is unclear if this is where the nitrogen is being supplied from or if it is being supplied from elsewhere and how the design is adequate to ensure that nitrogen will push RCS fluid through the CVCS system and sampling line to the sampling point.

Requested Additional Information:

1. Please provide information sufficient to determine the source of the nitrogen and ensure that the pressure is sufficient to push the RCS fluid to the sampling point. As part of the response, provide information on minimum water levels in the reactor following design basis accidents, the CVCS system piping connection heights, and nitrogen injection point(s). Update the DCD to discuss how nitrogen injection is adequate to perform this task.

2. Injecting nitrogen, or other gases, into the RCS will impact future samples of the containment atmosphere. Not only does nitrogen have the potential to dilute the containment gases, but also it could potentially result in backflow contamination of the instrument air system (See Bulletin 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment").

    a. Please discuss the impacts of injecting nitrogen, or other gases, into the RCS on the ability to collect accurate and useful containment gaseous samples and how the design ensures that useful containment atmosphere samples can still be taken following gas injection.

    b. Please describe in appropriate detail how the design prevents backflow into the instrument air system or how any backflow will not result in additional exposure to workers or additional release beyond what is accounted for in the accident analysis.

    c. Update the DCD as appropriate to address the above issues.

09.03.02-5

Regulatory Requirements:

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."

10 CFR Part 50, General Design Criteria (GDC) 61, requires in part that systems that contain radioactivity shall be designed with adequate safety under normal and postulated accident conditions be designed with suitable shielding for radiation protection.

10 CFR 20.1101(b) requires that applicants provide engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public as low as is reasonably achievable (ALARA).

Key Issue: The application does not have sufficient details and clarity to determine if shielding is adequate to meet dose requirements for workers.

DCD Section 9.3.2 discusses the option of providing temporary shielding around the sample lines during post-accident conditions.

DCD Section 12.3.1.1.12 indicates that, "Sample stations are designed with appropriate ventilation and shielding to minimize occupational exposures." However, the specifics of the shielding are not provided and it is unclear what is considered "appropriate" shielding.

Requested Additional Information:

In order for the staff to reach its safety and compliance conclusions regarding the regulatory requirements cited above, the staff needs to understand the details of radiation protection measure being taken to ensure exposures are within limits and ALARA. The DCD does not contain sufficient detail in order for the staff to conclude that the applicable regulatory requirements are met. Consequently,

Provide a more detailed description of the shielding for the sample racks (including materials and thicknesses) and update the DCD accordingly. Ensure that shielding is sufficient to meet applicable dose criteria and radiation zoning.

09.03.02-6

Regulatory Requirements:

10 CFR Part 50, General Design Criterion (GDC) 64, requires, in part, that means shall be provided for monitoring the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

10 CFR 20.1501 requires, in part, radiation surveys necessary to comply with the regulations and to evaluate potential radiological hazards. DSRS Section 12.3-12.4 indicates that the area radiation monitoring system will be acceptable if it meets several requirements, including the requirements of 10 CFR 20.1501

Key Issues: The application must have sufficient details and clarity to identify the radiation monitoring systems used for normal and accident releases and to demonstrate compliance with the applicable requirements.

In the NuScale DCD, Tier 2, Section 9.3.2.3 it specifies that a break in a sample line would result in activity release and a resulting actuation of area radiation monitors. However, radiation monitors are not specifically identified with numbers or by name, so it is unclear if the applicant is relying on the post-accident area monitors for the primary sampling equipment and containment sampling system equipment areas in DCD Table 12.3-10, "Fixed Area and Airborne Radiation Monitors Post-Accident Monitoring Variables." However, during normal operations, the amount of radioactivity in the sampling system fluids may not be high enough to trip the room area radiation monitor alarm and therefore, there could be an undetected leak and possible release. Such a release may be more easily detected by a process, effluent, or airborne radiation monitor.

In addition, applicants typically include identification numbers for the individual radiation monitors in DCD Chapters 11 and 12. The numbers are then used to identify the monitors when they are discussed in the DCD text and when they are shown in DCD figures. This allows the staff to identify which monitor is being referred to and allows staff to conclusively evaluate wether the radiation monitor type, range, location, function, etc, are adequate for all of the intended functions. In any event, it is necessary for the staff to understand the design, including the location and function, of necessary radiation monitors, with sufficient detail to determine compliance with GDC 64, 10 CFR 20.1501, and other applicable regulatory requirements.

Based on the above, staff requests the following, additional information:

1. Verify which radiation monitors are being referred to in the Section 9.3.2.3 text discussed above.

2. If area radiation monitors are being credited to alarm during a normal operation release from the sample systems, please provide additional information describing the area radiation monitors' adequacy to perform this function such as the monitors' sensitivity to adequately identify sampling system releases, recognizing that the monitors will likely also be exposed to other radiation sources (e.g. other reactor building sources, which may be operating at the design basis failed fuel percentage), which could interfere with the monitors ability to adequately detect a sampling system leak.

3. Please consistently identify radiation monitors throughout the DCD (For example, each monitor could be provided with unique numbers in the Chapter 11 and 12 tables, which could then be used when referencing specific monitors throughout the DCD text, tables, and figures).3. Please consistently identify radiation monitors throughout the DCD (For example, each monitor could be provided with unique numbers in the Chapter 11 and 12 tables, which could then be used when referencing specific monitors throughout the DCD text, tables, and figures).

09.03.02-7

Regulatory Requirements:

10 CFR Part 50, Appendix A, General Design Criterion 64, requires that "means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents."

10 CFR 20.1101(b) requires that licensees use to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

10 CFR 20.1701 requires the use, to the extent practical, of engineering controls (e.g., ventilation) to control the concentration of airborne radioactive material.

Key Issue: The application does not provide sufficient detail and clarity to determine if the sampling system will be able to meet ALARA and applicable OSHA requirements.

DSRS Section 12.5 specifies that Regulatory Guide (RG) 8.15 "Acceptable Programs for Respiratory Protection" provides elements of an acceptable respiratory protection program. RG 8.15, notes that in 1988, the NRC and the Occupational Safety and Health Administration (OSHA) signed a Memorandum of Understanding (MOU) to clarify jurisdictional responsibilities at NRC-licensed facilities (the MOU was renewed in 2013). The MOU makes it clear that if an NRC licensee is using respiratory protection to protect workers against non-radiological hazards (such as the inhalation of chemicals which could be hazardous to human health), the OSHA requirements apply (including 29 CFR 1910.134, as discussed in RG 8.15). Although NuScale DCD Tier 2, Table 1.9-2, "Conformance with Regulatory Guides" indicates that RG 8.15 is applicable to a COL applicant and not the design. However, design aspects of respiratory protection discussed in RG 8.15 would be applicable to the DCD.

DCD Section 9.3.2.2.1, under "Secondary Sampling System" states that the secondary sampling system provides a means for monitoring and collecting fluid samples in the steam cycle systems, which includes grab samples. DCD Section 9.3.2.2.2, under "Sample Panels," indicates that the secondary sampling system does not contain a vent hood enclosure because secondary samples are considered non-hazardous. However, the secondary system could contain radioactivity, primarily as a result of potential primary to secondary leakage. In addition, the secondary system could contain other chemicals which could be hazardous to human health, if inhaled. The use of a vent hood would significantly reduce worker inhalation of radioactive material if there is radioactive material in the secondary system, hence resulting in a potentially lower dose to the worker. It would also reduce worker inhalation to other hazardous chemicals which may be used to inhibit corrosion (such as hydrazine, morpholine, and ammonia) which may be in the secondary system.

Requested Additional Information:

Please explain how the design meets regulatory requirements and guidance as cited above, including maintaining exposures ALARA. Please provide additional justification for why secondary samples are considered non-hazardous and why a vent hood is determined to not be necessary considering the potential for radiological and chemical hazards in the secondary coolant.

09.03.02-8

Regulatory Requirements:

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.

Key Issue: The application does not contain sufficient clarity and detail for information regarding the ability to obtain a representative sample of the primary coolant post-accident and to be able to conclude it meets regulatory requirements.

DSRS Section 9.3.2 specifies that, the primary review organization and the organization responsible for the review of radiation protection compare the capability of the system to obtain representative samples of process fluids and the locations of sampling points with the guidelines for obtaining representative samples of fluids contained Regulatory Guide (RG) 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste."

RG 1.21 states that sample lines should be flushed for a sufficient period of time before sample extraction to remove sediment deposits, air, and gas pockets and that generally, three line volumes should be purged before withdrawing a sample, unless a technical evaluation or other justification is provided.

While DCD Section 9.3.2 specifies that sample lines are purged for enough time to ensure a representative sample, it does not specify how much time or how many line volumes are considered representative. While COL item 9.3-2 specifies that the COL applicant will develop a post-accident sampling contingency plans for using the process sampling system, it is not clear that this includes describing how a representative sample will be retained.

Based on the Three Mile Island experience, the purge fluid may be highly radioactive, so the process for obtaining representative samples will likely also affect the dose to the worker collecting samples. This is especially true since the design includes the option of using a temporary disposal tank to collect purged sample fluid (as specified in DCD Section 9.3.2.2.3, purged samples "may be

disposed to the liquid radioactive waste system (LRWS) or collected in a temporary disposal tank if radiation levels are expected to be too high for disposal to the LRWS").

Requested Additional Information:

Based on the above, please address the following comments/questions:

1. Please update the DCD to specify that the COL applicant will provide information regarding the process for collecting representative samples or provide additional information regarding what constitutes sufficient purge time to ensure that a representative primary coolant sample is being taken following an accident (e.g. specify the number of line volumes or the time it will take and how many line volumes that corresponds to, expected flow rate, etc.).

2. As discussed above, DCD Section 9.3.2.2.3 specifies that the purged sample may be disposed to the LRWS or collected in a temporary disposal tank if radiation levels are expected to be too high for disposal to the LRWS. Post-accident samples would be anticipated to be highly radioactive. If the radiation levels are too high for disposal to the LRWS, it is unclear why it would be appropriate to send the fluid to a temporary tank located within the same room as the workers taking the samples. While the response to RAI 8775, Question 12.03-1 specifies that shielding would be installed around the collection tank, it does not specify what source term is being assumed for the tank (which would be dependent on the volume of sample lines, including associated CVCS lines, would need to be purged before obtaining a representative sample). In addition, it does not appear that the dose from the tank is considered in the post-accident shielding analysis.

    a. Please provide information regarding what radiation dose limits are considered too high for disposal to the LRWS and the basis for such limits as well as the resulting dose rates to operators in the immediate area and adjacent areas. (e.g. would high dose rates in the LRWS result in high dose rates to other areas of the plant expected to be occupied or traversed following an accident?).

    b. Specify under what conditions the temporary purge collection tank is used. Provide additional information regarding the temporary purge collection tank and associated source term and how the dose from the temporary purge collection tank would potentially effect worker dose when collecting a sample. For example, provide the tank size, maximum source term (including volume and dimensions), location of the tank relative to worker areas when taking samples, amount of shielding assumed, and how it is ensured that this shielding will be available following an accident. Also indicate if there are specific design requirements (e.g., airborne activity control, temperature limitations, hose connectors, etc.) for the temporary purge tank. As part of the response, provide justification for the source term and shielding information used.

3. COL Item 9.3-2 states that "A COL applicant that references the NuScale Power Plant design certification will develop the post-accident sampling contingency plans for using the process sampling system and the CES off-line radiation monitor to obtain reactor coolant and containment atmosphere samples." While this COL item addresses programmatic aspects, it does not specify what the COL applicant will need to have for temporary equipment staged if required during an accident (e.g. adequately sized collection tank, adequate temporary shielding, sample transport cask, etc.). Please update the COL item to ensure that it is clear that the COL applicant will need to identify and procure any necessary temporary equipment and have it available and ready for use following an accident