NuScaleTRRaisPEm Resource

From: Cranston, Gregory
Sent: Thursday, June 27, 2019 2:39 PM
To: Request for Additional Information
Cc: NuScaleTRRaisPEm Resource; Lee, Samuel; Dudek, Michael; Hart, Michelle; Chowdhury, Prosanta; Tesfaye, Getachew
Subject: Request for Additional Information Letter No. 9690 (eRAI No. 9690) Topical Report, Design-Specific Methodology for Determining Appropriate Accidents to be Evaluated , Topical Report, 1.05, RGRB
Attachments: RAI_9690.pdf

Attached please find NRC staff's request for additional information (RAI) concerning review of the NuScale Topical Report.

Please submit your technically correct and complete response by August 26, 2019, 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. 9690 (eRAI No. 9690)

Issue Date: 06/27/2019
Application Title: NuScale Topical Report
Operating Company: NuScale
Docket No. PROJ0769
Review Section: 01.05 - Other Regulatory Considerations
Application Section: TR-0915-17565, Rev. 3

QUESTIONS

01.05-39

Regulatory Basis:

10 CFR 50.49(e)(4) requires that the radiation environment for equipment qualification must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

Background:

On April 21, 2019, NuScale submitted Revision 3 to TR-0915-17565, "Licensing Topical Report Accident Source Term Methodology." The revision included a new design basis iodine spike source term and reclassified the maximum hypothetical accident as a beyond design basis source term (DBST). This resulted in the maximum hypothetical accident no longer being considered for environmental qualification and the iodine spike source term being used for the maximum radiation environment being used for equipment qualification in and around containment.

In TR-0915-17565, Revision 3, Section 3.2.6, the applicant indicates that, "Spiking effects may occur for radionuclides besides iodines. However, any potential spiking of radionuclides besides iodine is implicitly accounted for by conservative treatments of the iodine spike DBST. For example, the assumed instantaneous event time-zero release of the entire primary coolant inventory results in doses expected to be several times larger than a more realistic graduated release of a primary coolant mass less than the entire primary coolant mass." The staff understands that assuming an instantaneous release may be conservative, but TR-0915-17565 does not provide information explaining NuScale's statement that the conservatisms bound the consideration of spiking of other radionuclides.

The applicant also does not provide any additional information or justification of the implicit conservatism to support their position except that the treatment of primary coolant activity, including iodine spiking, is consistent with RG 1.183. However, RG 1.183 assumes that a core melt accident is being considered for the radiation environment for equipment qualification, which typically bounds the dose to equipment inside containment. Since a core melt source term is not being considered for NuScale, additional justification is needed for why it is not necessary to consider the spiking of other radionuclides besides iodine for equipment impacted by the iodine spike design basis source term.

Issue:

Additional information is needed to demonstrate the conservatisms in developing the iodine spike DBST, as the staff is unable to make a determination that the radiation environment associated with the most severe design basis accident is being appropriately considered for environmental qualification.

Request:

Please provide 1) justification that the methodology used for developing the design basis iodine spike reactor coolant source term, described in TR-0915-17565 provides a source term that reasonably conservatively bounds the radiation environment associated with the most severe design basis accident, as required by 10 CFR

50.49(e)(4) or 2) update the topical report, as appropriate, to ensure that the methodology appropriately considers the potential for spiking of other radionuclides besides iodine or bounds the potential spiking of other radionuclides.

01.05-40

Regulatory Basis:

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are completed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; and 10 CFR Part 50, Appendix A, GDC 19, 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(xxviii) for control room radiological habitability. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) Section 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

Background:

On January 31, 2019, NuScale submitted a request for exemption from the 10 CFR 50.34(f)(2)(viii) requirements related to post-accident sampling. In the technical basis for the exemption request, the applicant describes that the capability to continuously monitor hydrogen and oxygen concentration in the containment atmosphere to meet the requirements of 10 CFR 50.44 is accomplished using the process sampling system (PSS) in-line monitors during accident conditions, including beyond-design-basis events with core damage. The path that the highly radioactive post-accident containment atmosphere would take to achieve continuous monitoring of hydrogen and oxygen concentration is outside of the containment, and the PSS is not related to safety. NuScale topical report TR-0915-17565, Revision 3, "Accident Source Term Methodology," was submitted on April 21, 2019. This topical report describes the accident source term and radiological consequence analysis methodology for the core damage event (CDE), which is used to show compliance with the regulatory requirements described above. The description of the CDE radiological consequence analysis in Section 4.2.5 of the topical report does not include discussion of the potential releases from the post-accident combustible gas monitoring pathway outside containment.

RG 1.183, Appendix A provides guidance on modeling of potential pathways to the environment for core damage accidents in the radiological consequence analyses which show compliance with the regulations stated above. Guidance on the modeling of ESF system leakage and containment purging in RG 1.183, although not directly describing the NuScale design post-accident combustible gas monitoring capability, provides indication that potential release pathways to the environment for the accident should be included in the analysis.

Issue:

Additional information is needed to describe the modeling of potential releases to the environment through the systems used in post-accident monitoring of hydrogen and oxygen concentration in the containment atmosphere to demonstrate compliance with the regulatory requirements described above.

Request:

Please describe the methods, models, and assumptions used for calculating the contribution to the dose from a potential release to the environment through leakage from the systems used in post-accident monitoring of the hydrogen and oxygen concentration in the containment atmosphere for the CDE. Additionally, please update the topical report to provide this description and make any necessary conforming changes to the FSAR.

01.05-41

Regulatory Basis:

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are completed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; and 10 CFR Part 50, Appendix A, GDC 19, 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(xxviii) for control room radiological habitability. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) Section 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

10 CFR 50.49(e)(4) requires that the radiation environment for equipment qualification must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

Background:

NuScale topical report TR-0915-17565, Revision 3, "Accident Source Term Methodology," was submitted on April 21, 2019. This topical report describes the accident source term and radiological consequence analysis methodology for the iodine spike design basis source term (iodine spike DBST), which is used to show compliance with the regulatory requirements described above.

Issue:

In order to make the finding on the acceptability of the topical report's methodology, additional information is needed for the staff to understand NuScale's implementation of the methodology and assumptions for the iodine spike DBST and how the topical report methodology is used to provide the source term information in Table 12.2-37 and dose rate information in Table 3C-8.

Request:

1. Please provide additional details in Section 3.2.6 of the topical report on the analysis assumptions for the iodine spike DBST, including bases for the assumptions, to the same level of detail as for the other design basis events. Include details such as the following:

    - Clarify the timing of the release to containment.
        - For example, clarify whether the entire integrated activity (including total coincident iodine spike values) is assumed to be released instantaneously, or is the initial RCS activity released instantaneously at time = 0, with coincident iodine spike activity appearing over 8 hours?
    - Clarify the assumptions on mixing in the containment.
    - For example, clarify the following:
        - Is the release mixed throughout entire containment air volume?
        - What is the assumed containment air volume? Is it the same value for containment air volume used in the CDE dose analysis?
    - Additionally, please revise the text in Section 3.2.6 to clarify that the iodine spike design basis source term includes 2 iodine spiking cases.

2. Provide additional detail in FSAR Section 15.0.3.8.6 on iodine spike DBST assumptions and their bases, similar to the detail for other DBAs in FSAR Section 15.0.3.8. Solely relying on a reference to the topical report does not give the staff enough information to make a safety finding. Include such information as the following:

    - Timing of release.
    - Containment mixing assumptions.
        - For example, please clarify the following:
            - Is the release mixed throughout entire containment air volume?
            - What is the assumed containment air volume?
            - Is it the same value for containment air volume used in the core damage event dose analysis?
    - Assumed mass of the primary coolant.

- Assumptions for the two iodine spiking cases.

3. Provide additional information, including updates in the FSAR and/or topical report, as appropriate, to describe the methods, models, and assumptions used in developing the source term provided in FSAR Table 12.2-37.3.

4. Please describe the methods, models, and assumptions used for calculating the total integrated doses provided in FSAR Table 3C-8. Please ensure that the discussion includes information demonstrating why the maximum design basis accident total integrated dose values provided in FSAR Table 3C-8 represent dose rates for the most severe design basis accident, with appropriate margin, as required by 10 CFR 50.49(e)(4) and 10 CFR 50.49(e)(8). Include updates to the FSAR and/or topical report, as appropriate.

01.05-42

Regulatory Basis:

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are completed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; and 10 CFR Part 50, Appendix A, GDC 19, 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(xxviii) for control room radiological habitability. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) Section 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

Background:

NuScale topical report TR-0915-17565, Revision 3, "Accident Source Term Methodology," was submitted on April 21, 2019. This topical report describes the accident source term and radiological consequence analysis methodology for the core damage event (CDE), which is used to show compliance with the regulatory requirements described above. Changes to FSAR Chapter 15 were submitted by letter dated April 19, 2019, including description of the CDE accident analysis, which implements the topical report methodology for the NuScale design certification application (DCA).

As discussed in NuScale's supplemental response to RAI 9224 dated April 9, 2019, the CDE source term methodology considers a set of five severe-accident scenarios and takes the median value for the release fraction to the containment for each radionuclide group to provide a representative (not bounding) source term. However, it is unclear how uncertainty related to the core damage and release phenomena is accounted for in the CDE source term methodology. In other words, for each of the five severe-accident scenarios, how accurately can the release to the containment be predicted? In its independent confirmatory analysis, the staff noted that for scenario LCC-05T the staff's predictions of iodine releases from the core at 48 hours were 90% as opposed to NuScale's prediction of 55%. The staff's confirmatory analysis is described in the staff document RES/FSCB 2019-01, "Independent MELCOR Confirmatory Analysis for NuScale Small Modular Reactor," ML19114A041 (proprietary), sent to NuScale by encrypted file in May 2019. As shown in numerous past studies for LWRs, and in particular, the staff's confirmatory analysis, there is considerable uncertainty in core heatup and degradation, specifically involving late phase core melt progression. NuScale's analysis shows only partial core damage resulting in release from the fuel of less than half of the volatile radionuclides, whereas larger amounts of core degradation and subsequent radionuclide release could potentially occur, especially in an unrecovered scenario.

Issue:

The staff notes that there is limited margin between the FSAR's CDE dose results and the control room habitability 5-rem dose limit. Because calculated dose results are generally proportional to the assumed release fraction to the containment and because NuScale's volatile radionuclide release fraction from the fuel is low, additional information is needed to clarify the treatment of uncertainty in the release fraction to the containment and its subsequent effect on the FSAR's CDE source term and dose results.

Requests:

1. Please describe in the topical report the basis and justification for assuming that partial core damage involving limited release of volatile fission products is appropriately conservative for developing the CDE release fractions to the containment to show compliance with the dose requirements given in the regulatory basis.

2. Taking into consideration the uncertainty in the release fractions to the containment, would any applicable dose criteria be exceeded for the CDE?