NuScaleTRRaisPEm Resource

From:	Foster, Rocky
Sent:	Tuesday, November 01, 2016 7:45 AM
To:	Bergman, Tom
Cc:	'smirsky@nuscalepower.com'; Pope, Steven (spope@nuscalepower.com); Unikewicz, Steve; NuScaleTRRaisPEm Resource; NuScaleTRRaisPEm Resource; Tonacci, Mark; Hart, Michelle; Burkhart, Lawrence; Compton, Keith; Cranston, Gregory; Foster, Rocky
Subject:	Request for Additional Information Letter No. 6 for the Review of NuScale Topical Report, TR-0915-17565, "Accident Source term Methodology," Revision 1 (CAC NO. RQ6004)
Attachments:	Final RAI 06 eRAI_8692 AST TR.pdf

Tom,

Attached please find NRC staff's request for additional information concerning NuScale topical report entitled, "Accident Source Term Methodology," Revision 1.

Please submit your response by March 1, 2017, to the NRC Document Control Desk. If you have any questions, please feel free to contact me.

Thank you,

Rocky D. Foster
Project Manager
US Nuclear Regulatory Commission
Office of New Reactors
Division of New Reactor Licensing
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November 1, 2016

Mr. Thomas Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC
1100 NE Circle Boulevard, Suite 200
Corvallis, OR 97330

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 6 FOR THE REVIEW OF NUSCALE TOPICAL REPORT, TR-0915-17565, "ACCIDENT SOURCE TERM METHODOLOGY, REVISION1 (CAC NO. RQ6004)

Dear Mr. Bergman:

In an April 8, 2016, letter, NuScale Power, LLC, submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) TR-0915-17565, "Accident Source Term Methodology," Revision 1. The NRC staff is performing a detailed review of this topical report to enable the staff to reach a conclusion on the safety of the proposed application. The NRC staff has identified that additional information is needed to continue portions of the review. The NRC staff's request for additional information (RAI) is contained in the enclosure to this letter.

To support the review schedule, NuScale is requested to respond within 120 calendar days of the date of this letter. If changes are needed to the topical report, the NRC staff requests that the RAI response include the proposed wording changes.

If you have any questions or comments concerning this matter, you may contact me at 301-415-5787.

Sincerely,

/**RA**/

Rocky D. Foster
Project Manager
Licensing Branch 1
Division of New Reactor Licensing
Office of New Reactors

Docket No. PROJ0769
eRAI Tracking No. 8692

Enclosure: Request for Additional Information

Request for Additional Information 6
Issue Date: 11/01/2016
Application Title: NuScale Topical Report
Operating Company: NuScale
Docket No. PROJ0769
Review Section: 01.05 - Other Regulatory Considerations
Application Section: TR-0915-17565-P, Rev.1 Accident Source Term Methodology

QUESTIONS

01.05-11

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

Are there other possible design basis accidents (DBAs) to assess for radiological consequences that were screened out or not deemed credible in the development of the accident source term methodology topical report? If during the design certification application review additional DBAs are identified, please provide a description of what would be the applicant's method for how the topical report be applied or be used in these cases?

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.

ENCLOSURE

01.05-12

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

Considering the 10 CFR 52.47 requirement to assess the radiological consequences of an accident for purposes of site analysis, how are multi-module considerations taken into account to determine what NuScale is calling the “design basis source term” accident?

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.

01.05-13

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

On pages 13-16 in Section 3.1 of the topical report provides information on applicable software. Does the proposed methodology require the use of these specific computer codes, or may NuScale (or a subsequent COL applicant or licensee) use a different code when implementing the methodology? If a specific code is required, the specific code version should be listed.

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and

Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.

01.05-14

The staff requires the following information related to the discussion of design basis accidents (DBAs) in the topical report in order to complete its review:

a) Does each DBA assume worst single failure and loss of offsite power (LOOP)? If not, justify why not.

b) In the descriptions of the rod ejection accident (Section 3.2.1), main steam line break outside containment (Section 3.2.3) and steam generator tube failure (Section 3.2.4) analyses, the topical report states that primary coolant leaks into both Steam Generators at the maximum leak rate allowed by design basis limits. Does this refer to technical specification (TS) limits for operational leakage, accident-induced primary-to-secondary leakage as discussed in the TS steam generator program, or some other basis? Please provide the basis.

c) For the rod ejection accident description in Section 3.2.1, what is the basis for the value for leakage from secondary system isolation valves (e.g., TS limit, other)?

d) In the Section 3.2.2 description of the fuel handling accident, the referenced guidance from RG 1.183, Appendix B for the assumption that the iodine chemical forms are equal to 57% elemental and 43% organic for releases from the pool water is dependent upon the modeling of the pool iodine decontamination factors. Considering the proposed change in pool iodine decontamination factors from those provided in RG 1.183 (500 for elemental iodine and 1 for organic iodine), how is the assumed iodine chemical form of the release from the pool affected?

e) With respect to the description of the failure of small lines carrying primary coolant outside containment (Section 3.2.5); The containment isolation valve leakage is assumed to be based on design basis limits – does this refer to TS limits on unidentified reactor coolant system operational leakage or some other basis? Please provide the basis.

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the

requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.

01.05-15

The staff requires the following information with respect to topical report Section 3.2.2, primary coolant radionuclide inventory:

a) Is NuScale requesting approval of the method for determining the primary coolant radionuclide inventory as part of the accident source term methodology topical report?

b) How are the initial (equilibrium) primary coolant activity concentrations determined permitted by the design basis? What is the basis and justification for calculation of the values? Are the design basis values in the technical specifications?

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.

01.05-16

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

On page 28, Section 3.3.4.4, the topical report states that pre-accident coolant radiation levels are neglected for any accident with damaged fuel. Since the fuel releases are relatively small for the NuScale design, have you completed analyses to confirm that any radioactive release from coolant would be negligible? Please provide the analyses.

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.

01.05-17

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

On page 28 of the topical report, Section 3.3.4.6, with respect to the radiation shine radiological consequences; are the shine doses calculated for the event with the largest activity release to be applied to each of the DBAs? Please provide a description of the evaluation of shine dose for each DBA.

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.

01.05-18

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident

radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

On page 32, Section 3.3.8, the topical report proposed use of the exponential function from the Burley paper (U.S. Nuclear Commission, “Evaluation of Fission Product Release and Transport for a Fuel Handling Accident,” Oct. 1971) applied to greater pool depth than considered in the reference. The staff requires the following information to complete its review of this topic:

a) Is there an upper limit on iodine decontamination factors related to capability of water to remove either inorganic or organic iodine in regards to use of the reference Burley paper, including any other reference on the topic considered by the applicant?

b) Please provide the analyses for determination of the pH of the reactor building pool for the NuScale design and indicate how the reactor building pool pH affects decontamination.

c) Justify how the NuScale fuel design meets the release pressure assumption from the Burley paper of 1200 psig or less.

d) What is the basis for the organic/inorganic iodine ratio assumption for NuScale? Would the ratio be different than that assumed for large light water reactors?

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.

01.05-19

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

On page 58, Section 4.2.5, for the source term design basis accident the topical report states that the chemical form of iodine released to containment atmosphere is assumed to be the same as in RG 1.183, Appendix A. What is the basis for applicability of this assumption to the NuScale design?

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.

01.05-20

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

In Section 4.3.5 the topical report proposes that there is not a maximum limit on elemental iodine decontamination factor (DF) because it is a natural process. Provide additional technical justification for the lack of a maximum limit on the assumed elemental iodine DF. The referenced information from Position 3.3 of Appendix A to RG 1.183 applies to the assumptions for particulate (aerosol) removal DF, not elemental iodine DF.

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.

01.05-21

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

Section 4.5.6 of the topical report states that the amount of iodine re-evolution that could occur between pH_T values of 6.0 and 7.0 is negligible. Please provide the basis for this statement in the topical report and any calculations used to confirm that the iodine re-evolution is negligible based on the calculated pH_T for the NuScale design.

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, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. 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.