NuScale Power

February 8, 2017

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Correction to Response to NRC Request for Additional Information Letter No. 6 and Letter No. 11 for the Review of Topical Report TR-0915-17565, "Accident Source Term Methodology," Revision 1 (TAC No. RQ6004)

REFERENCES:

1. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "Submittal of Response to NRC Requests for Additional Information Letter No. 6 and Letter No. 11 for the Review of Topical Report TR-0915-17565, "Accident Source Term Methodology," Revision 1 (TAC No. RQ6004), dated February 6, 2017

In a letter dated February 6, 2017 (Reference 1), NuScale Power, LLC (NuScale) submitted the response to NRC Request for Additional Information Letter No. 6 and Letter No. 11 for the Review of Topical Report TR-0915-17565, "Accident Source Term Methodology," Revision 1.

Several typographical errors were subsequently identified in the response to NRC RAI Number 6 Question Number 01.05-14 (d) that require amendment.

The purpose of this letter is to provide the corrected NuScale response to the NRC RAI Number 6, Question Number 01.05-14 in its entirety. The attached pages replace the Enclosure 1 pages 6 and 7 of Reference 1.

NuScale regrets any inconvenience this amended response might cause.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments. Please feel free to contact Jennie Wike at 541-360-0539 or at jwike@nuscalepower.com if you have any questions.

Sincerely,

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NuScale Power, LLC

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Attachment: NuScale Revised Response to NRC Request for Additional Information Letter No. 6 Question Number 01.05-14, for TR-0915-17565, "Accident Source Term Methodology" Revision 1

Attachment:

NuScale Revised Response to NRC Request for Additional Information Letter No. 6 Question Number 01.05-14, for TR-0915-17565, “Accident Source Term Methodology” Revision 1

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NRC RAI Question Number: 01.05-14

NRC RAI Question

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.

NuScale RAI Question Response

a) Design basis accidents were evaluated considering the loss of AC power and concurrent loss of DC power. In addition, the analyses evaluated the limiting single failures for dose, primary and secondary pressure, minimum critical heat flux ratio and other acceptance criteria. The evaluation for the loss of power and worst single failure discussions are described in Chapter 15 of the FSAR. For many accidents and transients, the limiting cases occur when power is available, because the loss of AC or DC power results in an earlier reactor trip and safety system actuation. Transient analysis cases were run to determine the worst single failure for radiological consequence maximization. Outputs from these cases were then used as inputs in the accident radiological analysis, from which the limiting dose consequences are determined.

b) The basis for the specific value of the limit is independent of the methodology that NuScale is seeking approval for in the report. The methodology only seeks to instruct how to utilize the limit when the limit is provided as a given input. For some cases, where the Technical Specification limit is consistent with the design basis limit, such as primary to secondary leakage of 150 gallons per day (gpd), that value was used in the calculations. Some design basis limits do not have corresponding Technical Specification limits. The term “design basis limits” is used in the report because the report design basis limits reflect limits to which the system must be designed. The Technical Specifications and programs established by the Technical Specifications, such as leakage testing for containment isolation valves (CIVs), protect the design basis limits assumed in the safety analysis. Some margin may exist such that the design basis limits and the Technical Specification limits may be different from one another. For example, for events where a single CIV is relied upon to maintain containment integrity (due to single failure of the redundant valve or component), the valve leakage was conservatively assumed to be equal to the maximum allowed Technical Specification leakage limit for the containment.

c) Leakage from the Steam Generator through the Feedwater Isolation Valves and Main Steam Isolation Valves is based on the maximum allowed leakage for the containment of 8.2E-03 cfm for 30 hours after the event. The RCS is assumed to be de-pressurized within 30 hours. The assumed leakage value bounds the acceptance criteria for the leakage testing of the valves such that actual leakage is expected to be less than the analysis assumption.

d) The iodine chemical forms of 57 percent elemental and 43 percent organic were not used in the calculation. Rather, Equation 3-19 of the report calculates an overall effective decontamination factor (DF) based on the assumed organic/inorganic ratio and corresponding decontamination factors. The initial iodine source term is divided by this effective decontamination factor in the modeling for this event. See the response to question 01.05-18(d) for a detailed basis for the use of the assumed organic/inorganic ratio. Equation 3-18 of the report calculates the decontamination factor of inorganic iodine as a function of pool depth. Additionally, as noted in the NRC Regulatory Issue Summary (RIS) 2006-04 (Reference 7.2.11), an overall DF of 200 is achieved when the DF for elemental iodine is 285, instead of 500. With the use of Equation 3-19, this corresponds to an organic fraction of 0.15 percent and an inorganic fraction of 99.85 percent. Section 3.2.2 of the report has been modified to remove the reference to the RG 1.183 iodine chemical forms of 57 percent elemental and 43 percent organic assumption.

e) For the failure of small lines carrying primary coolant event, the design basis limit for primary to secondary leakage (150 gpd) is consistent with the Technical Specification limit. The design basis limit for CIV leakage for this event was conservative with respect to the maximum allowed by Technical Specifications for containment leakage.

<u>Impact of NRC RAI Question Response on TR-0915-17565:</u>

In response to NRC RAI Question 01.05-14(d), the second bulleted item in Section 3.2.2.of the report will be deleted.