February 6, 2017 Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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**SUBJECT:** NuScale Power, LLC 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)

**REFERENCES:**

1. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "Accident Source Term Methodology," Revision 1, TR-0915-17565, dated April 8, 2016 (ML 16099A394)

2. NuScale Topical Report "Accident Source Term Methodology," TR-0915-17565, Revision 1, dated April 2015 (ML 16099A394)

3. Letter from U.S. Nuclear Regulatory Commission to NuScale Power, LLC, "Request for Additional Information Letter No. 6 for the Review of NuScale Topical Report TR-0915-17565, "Accident Source Term Methodology," Revision 1 (TAC No. RQ6004) dated November 1, 2016 (ML 16306A008)

4. Letter from U.S. Nuclear Regulatory Commission to NuScale Power, LLC, "Request for Additional Information Letter No. 11 for the Topical Report 0915-17565, "Accident Source Term Methodology," Revision 1 (TAC No. RQ6004) dated December 23, 2016 (ML 16358A422)

In a letter dated April 8, 2016 (Reference 1), NuScale Power, LLC (NuScale) submitted the topical report entitled "Accident Source Term Methodology," Revision 1 (Reference 2). In letters dated November 1, 2016 (Reference 3) and December 23, 2016 (Reference 4), the NRC Staff provided Requests for Additional Information (RAIs) regarding the subject topical report.

The purpose of this letter is to provide NuScale responses to the NRC RAIs. Enclosure 1 provides the NuScale response to RAI Letter No. 6 and RAI No. 11. The information provided in Enclosure 1 is nonproprietary.

As an assist for NRC review of these RAI responses, when a response necessitated a revision or change to the report, redline/strikeout pages of the TR-0915-17565, Revision 1, were generated for the affected pages. Enclosure 2 provides the proprietary redline/strikeout pages of TR-0915-17565, Revision 1. Enclosure 3 provides the nonproprietary redline/strikeout pages of TR-0915-17565, Revision 1.

NuScale requests that the proprietary Enclosure 2 be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 4) supports this request.

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,

Thomas A. Bergman
Vice President, Regulatory Affairs
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Enclosure 1: NuScale Response to NRC Request for Additional Information Letter No. 6 and Letter No. 11, for TR-0915-17565, "Accident Source Term Methodology" Revision 1
Enclosure 2: Redline/strikeout pages of NuScale Topical Report TR-0915-17565, "Accident Source Term Methodology," Revision 1, proprietary version
Enclosure 3: Redline/strikeout pages of NuScale Topical Report TR-0915-17565, "Accident Source Term Methodology," Revision 1, nonproprietary version
Enclosure 4: Affidavit of Thomas A. Bergman, AF-0217-52946

## Enclosure 1:

NuScale Response to NRC Request for Additional Information Letter No. 6 and Letter No. 11, for TR-0915-17565, “Accident Source Term Methodology” Revision 1

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NRC RAI Number: 06 NRC RAI Date: November 1, 2016

NRC Review of: Accident Source Term Methodology, TR-0915-17565-P, Revision 1

NRC RAI Question Number: 01.05-11

NRC RAI Question

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?

NuScale RAI Question Response

One DBA that was deemed not credible is the drop of the NuScale Power Module (NPM). The NPM is moved with the Reactor Building Crane, which is single-failure-proof and meets the criteria of NUREG-0554 and ASME NOG-1 for Type I cranes. Consideration of the NPM drop event is addressed in the Final Safety Analysis Report (FSAR) Chapter 19 as a beyond design basis event.

Chapter 15 of the NuScale Final Safety Analysis Report (FSAR) presents the spectrum of DBAs for the NuScale design. The establishment of the design basis accident spectrum analyzed in Chapter 15 of the FSAR was outside of the scope of TR-0915-17565. As stated in the abstract and implied elsewhere in the report, this report describes the methodology used for establishing the source terms and radiological consequences for a spectrum of design basis accidents typical for Chapter 15 analyses.

If new DBAs with radiological consequences are postulated, NuScale will determine whether a revision to this report is required, using the applicable change control processes.

Impact of NRC RAI Question Response on TR-0915-17565:

This RAI response does not require a revision to the report.

NRC RAI Question Number: 01.05-12

NRC RAI Question

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?

NuScale RAI Question Response

NuScale follows the approach of the 2012 NEI position paper on SMR source terms (Reference 7.2.20 of the report) by referring to the scenario described in footnote 3 of 10 CFR 52.47(a)(2)(iv) as the maximum hypothetical accident (MHA). A source term design basis accident (STDBA) is a postulated accident scenario, meant as a surrogate to the large break LOCA typically evaluated by LWRs, to meet the regulatory intent of addressing the MHA. NuScale’s methodology utilizes the design basis source term (DBST) for the MHA scenario to meet the intent of 10 CFR 52.47(a)(2)(iv). As discussed in Section 4.2 of the report, the DBST is composed of a blend of key parameters, such as fuel release fractions and timing, taken from a spectrum of STDBAs.

For the NuScale design, the STDBAs are single module events. The DBST is therefore a single module event. A single-module DBST is consistent with 10 CFR 52.47(a)(2)(iv), which requires an assessment of plant design features to mitigate the radiological consequences of accidents. Assessment is based upon an assumed fission product release from the core into the containment, and considers postulated site parameters to evaluate offsite radiological consequences.

Both the NuScale radiological consequences assessment, required by 10 CFR 52.47(a)(2)(iv) and a license applicant’s site analysis performed under 10 CFR 52.79(a)(1)(vi)) or 10 CFR 50.34(a)(1)(ii) evaluate the capabilities of “the facility” to limit radiological consequences to prescribed offsite dose limits. A “utilization facility” is defined in 10 CFR 50.2 (applicable by way of 10 CFR 52.1(b)) as “any nuclear reactor…,” and a “nuclear reactor” is in turn defined as “an apparatus…designed or used to sustain nuclear fission.” Thus, NuScale understands “the facility” subject to assessment under 10 CFR 52.47(a)(2)(iv) to refer to a single reactor, comparable to a nuclear power unit as that term is defined in 10 CFR Part 50 Appendix A. Accordingly, it is consistent with 10 CFR 52.47(a)(2)(iv) to assess the radiological consequences from a single unit’s fission product release.

The assessment of radiological consequences on a single-unit basis is also consistent with precedent. Multi-unit sites have not previously assessed the radiological consequences of multiple reactor fission product releases when determining the design basis accident source term. Although multi-unit sites sometimes share SSCs that perform important to safety functions (e.g., ultimate heat sink, emergency power), application of the General Design Criteria (GDCs), particularly GDCs 2, 3, 4, and 5, adequately protects against a multi-unit accident.

In an October 25, 2016 letter entitled, “Response to NuScale Power, LLC Key Issue Resolution Letter, Supplemental Response Regarding Multi-Module Questions,” (ML16229A522), NRC

acknowledged the role of appropriate design provisions in preventing multi-unit accidents. As stated therein:

> NRC regulations in Appendix A to 10 CFR Part 50 state requirements that address hazards within the design basis - internal and external to a plant. These requirements are stated most notably in Appendix A, General Design Criterion 2 – *Design bases for protection against natural phenomena,* Criterion 3 – *Fire Protection,* and Criterion 4 – *Environmental and dynamic effects design bases.* These requirements address assurance that a single hazard would not render SSCs important to safety incapable of performing their intended safety function. Criterion 5 – *Sharing of structures, systems and components* states that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair the SSC's ability to perform the safety function. For example, the NRC staff expects that provisions will be included in the station design to prevent fires or floods initiated in one nuclear power unit from spreading to another nuclear power unit and causing the failure of safety-related equipment associated with that nuclear power unit. Typical hazards include internal fires, internal flooding, and natural phenomena (e.g., seismic events and high winds).

Specific treatment of these GDCs in the NuScale design is beyond the scope of this report. An applicant referencing the report would need to consider the impact on DBST if GDCs 2, 3, 4, or 5 were not met for a multi-module design. As to the NuScale design in particular, these GDCs are addressed in the applicable system design descriptions in the NuScale FSAR. All safety-related systems are module (unit) specific, except for the ultimate heat sink, which is capable of performing its safety function for a limiting 12 module heat load. The safety-related systems are protected from internal events and external events, and adverse system interactions between units have been considered. Unlike other multi-unit designs, NuScale further addresses 10 CFR 52.47(c)(3), to ensure that no restrictions in operating configurations or interface requirements are necessary to ensure the safe operation of any operating NPMs during installation, testing, or startup of subsequent NuScale Power Modules (NPMs). See Chapter 21 of the NuScale FSAR for additional information.

The NRC's October 25, 2016 letter further states (underline added for emphasis):

> [T]he NRC staff's expectation is that applicants with modular plant designs will use information related to the interaction between plant SSCs in conjunction with risk insights from consideration of the modular design and operation (refer to Standard Review Plan (SRP) Chapter 19.0) to <u>appropriately categorize events and determine the design basis accidents (DBAs) for the plant design</u>, including any credible nuclear power unit events or accidents resulting from the modular design. This would include a description of the scenarios considered and determination of the event category for any potential DBAs that are unique to the design or may not have been previously described in the SRP as applicable to a single-reactor unit, <u>including the DBAs used to evaluate potential radiological consequences for the safety and siting assessment</u>.

The NuScale design basis applies GDCs 2, 3, 4, and 5 consistent with other multi-unit designs in order to prevent accidents in multiple units. Risk insights from consideration of the modular

design and operation were used to appropriately categorize events. Probabilistic Risk Assessment (PRA) was used to evaluate scenarios, excluding those initiated by external events, in which there is a potential for the STDBAs to involve multiple modules. The results of the multi-module PRA presented in Chapter 19.1.7 of the FSAR show that the risk from internal events to multiple modules at full power is extremely low. The frequency of scenarios that result in core damage to more than one reactor module was found to be much less than the frequency of the STDBAs developed for a single module that experiences core damage. Accordingly, the existing single-unit radiological consequence assessment is used for the NuScale design and STDBAs involving core damage in multiple modules were not considered in the determination of the DBST.

Note that emergency planning and the emergency planning zone (EPZ) are outside the scope of this report. Relative to establishment of the EPZ, NRC stated the following in the October 25, 2016 letter, “consideration of more than a single module in development of the source terms and technical basis to evaluate EPZ size is appropriate.”

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

This RAI response does not require a revision to the report.

NRC RAI Question Number: 01.05-13

NRC RAI Question

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.

NuScale RAI Question Response

The proposed methodology of the report was developed based upon use of the following software. The versions currently used for NuScale analyses are also listed.

- SCALE 6.1.3
- NARCON 1.0.1 (NuScale version of ARCON96 with input/output edit differences)
- NMAEROS 1.0.0 (NuScale version of MAEROS (October, 1982) with input/output edits)
- RADTRAD Version 3.10
- MELCOR 2.1
- NRELAP5 Version 1.3
- STARNAUA Version 1.04
- $pH_T$ Version 1.0.1
- MCNP6 Version 2.0

The software codes listed above are subject to occasional adjustments and corrections (e.g., input/output functions or numerical techniques that improve execution time) without affecting the basic solution technique or results. Software is also subject to error correction processes that can update version numbers. Accordingly, the proposed methodology of the report used the above codes, but does not require use of the specified software versions.

Impact of NRC RAI Question Response on TR-0915-17565:

Clarification is provided for the use of MCNP6 and NuScale-specific versions of ARCON96 in Section 3.1 and MAEROS in Section 4.3.10.

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 3-3 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 3-2 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 3-3, this corresponds to an organic fraction of 0.15 percent and an organic fraction of 99.75 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.

NRC RAI Question Number: 01.05-15

NRC RAI Question

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?

NuScale RAI Question Response

a) NuScale is not requesting approval of the method for determining the primary coolant radionuclide inventory as part of this report. Chapter 11 of the FSAR presents how the normal operation primary coolant radionuclide inventory for the NuScale design is established.

b) Chapter 11 of the FSAR presents how the normal operation primary coolant radionuclide inventory for the NuScale design is established. As described in Section 3.3.2 of the report, the design limits for maximum dose equivalent I-131 and Xe-133 are used to scale the normal operation concentrations to design limit concentrations. See RAI Number 01.05-14(b) for a discussion of design basis limits and Technical Specifications.

Impact of NRC RAI Question Response on TR-0915-17565:

This RAI response does not require a revision to the report.

NRC RAI Question Number: 01.05-16

NRC RAI Question

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.

NuScale RAI Question Response

The pre-accident coolant radiation levels are based on a 0.028 percent failed fuel fraction, of which only a fraction of the radionuclides in the failed fuel are released into the coolant, based on the isotope specific escape coefficient. Primary coolant cleaning systems further reduce this pre-accident coolant radionuclide content. The events in which damaged fuel rods are the source of radiation are the fuel handling accident (FHA) and the DBST. For the DBST, Table 5-9 of the report shows 22 percent of the core inventory of halogens are released into containment for the median case. A fraction of 0.028 percent is negligible compared to 22 percent, so pre-accident coolant radiation levels being neglected is justified for the DBST.

The FHA occurs in the reactor pool, therefore pre-accident primary and secondary coolant radiation levels are inapplicable. As noted in Section 3.3.4.7 of the report, the contribution from reactor pool boiling to the radiological consequences is considered.

The methodology for the calculation of the radiological consequences of the rod ejection accident (REA) is presented in Section 3.2.1 of the report in the event fuel failures were to occur. The safety analysis for the REA determined that no fuel failures occur for the NuScale design, thus no radiological analysis was presented in Chapter 15 of the FSAR for the REA. An example analysis for one fuel assembly with all fuel pins damaged is presented in Table 5-6 of the report resulting in doses below the acceptance criteria. One fuel assembly of damaged fuel in the NuScale 37 fuel assembly core is an approximately 2.7 percent failed fuel fraction, which makes any contribution from the expected maximum 0.028 percent failed fuel fraction during normal operation negligible. Therefore, in the event of calculated fuel damage from the REA, pre-accident coolant radiation levels are neglected in the calculation of accident radiological consequences.

Impact of NRC RAI Question Response on TR-0915-17565:

This RAI response does not require a revision to the report.

NRC RAI Question Number: 01.05-17

NRC RAI Question

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.

NuScale RAI Question Response

The shine dose for the bounding event is calculated and applied to each of the DBAs rather than specifically calculating the shine dose for each event.

Impact of NRC RAI Question Response on TR-0915-17565:

Section 3.3.4.6 of the report will be revised to clarify that the shine dose from the bounding event is applied to all events rather than specifically calculating the shine dose for each event.

<u>NRC RAI Question Number:</u> 01.05-18

<u>NRC RAI Question</u>

On page 32, Section 3.3.8, the topical report proposed use of the exponential function from the Burley paper (U.S. Nuclear Regulatory 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?

<u>NuScale RAI Question Response</u>

a) There is an upper limit on the total iodine decontamination factor of 400 per page 13 and 26 of Reference 7.2.12 (i.e., “the Burley paper”) that is fundamentally based on the organic iodine fraction of 0.25 percent.

b) The NuScale reactor pool pH is 4.75 at nominal temperatures. Over the full range of possible pool temperatures the temperature effect on pH is negligible. Page 23 of Reference 7.2.12 (i.e., “the Burley paper”) discusses the effect of pH on partition factors and the value assumed in the analysis, and states “…a pH of about 5 are generally appropriate”. As the pH in the NuScale reactor pool is similar to the pH in a conventional PWR spent fuel pool, the effect of pH on the partition factors documented in conventional PWRs is directly applicable.

c) The assumption of 1200 psig or less provided in Reference 7.2.12 (i.e.,”the Burley paper”) is relevant to the bubble rise velocity term in Equation 3-18 of the report. However, this equation is still appropriate to use for release pressures greater than 1200 psig. The assumption of 1200 psig was implemented in RG 1.25 (circa 1972), along with other assumptions related to power density, fuel temperatures, and fuel exposures (i.e., 25 GWD/MTU). The NRC letter entitled ‘Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Task Interface Agreement 99-03 Regarding Potential Nonconservative Assumptions for Fuel-Handling Accident,’ (ML993340503) (circa 1999) documents a Safety Evaluation (SE) in regard to the assumptions identified in RG 1.25. The SE states “*that there is reasonable assurance that adequate protection of the public from the effects of design basis FHAs involving fuel with peak rod average burnups as high as 62 GWD/MTU will continue.*” Subsequent to this NRC SE, RG 1.183 (circa 2000)

was issued in which these assumptions from Reference 7.2.12 and RG 1.25 were not included. In regard to the maximum fuel rod pressurization of 1200 psig specifically, the appendix of the SE notes that a higher pressure may result in smaller gas bubbles, leading to a reduced terminal velocity, longer rise time, and thus higher decontamination. Therefore, the restriction of 1200 psig identified in Reference 7.2.12 is not applicable to high exposure fuel (i.e., 62 GWD/MTU) in conventional PWR designs or in the NuScale design.

d) A ratio corresponding to an organic fraction of 0.0015 is derived from RG 1.183 and a ratio corresponding to an organic fraction of 0.0025 is derived from Reference 7.2.12. As the NuScale fuel design is based on conventional 17x17 PWR fuel design with similar geometries, materials, and fuel exposures, the ratio of organic to inorganic iodine is not expected to be different for NuScale compared to a conventional PWR.

Impact of NRC RAI Question Response on TR-0915-17565:

In response to NRC RAI Question 01.05-18(a), Section 3.3.8 of the report will be revised to impose an upper limit of 400 on the total iodine decontamination factor.

NRC RAI Question Number: 01.05-19

NRC RAI Question

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?

NuScale RAI Question Response

RG 1.183, Appendix A states that "The assumptions in this appendix are acceptable to the NRC staff for evaluating the radiological consequences of loss-of-coolant accidents (LOCAs) at light water reactors (LWRs)."

The NuScale design is an LWR that uses a reduced height 17x17 PWR fuel design that has similar geometries, materials, and fuel exposures as conventional PWR fuel designs. Therefore, the chemical form of iodine assumed for conventional LWRs dictated by RG 1.183, Appendix A, is applicable to the NuScale design.

Impact of NRC RAI Question Response on TR-0915-17565:

This RAI response does not require a revision to the report.

NRC RAI Question Number: 01.05-20

NRC RAI Question

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.

NuScale RAI Question Response

The methodology conservatively does not take credit for elemental iodine removal. Rather, only aerosol removal is credited.

Impact of NRC RAI Question Response on TR-0915-17565:

Section 1.2, Section 4.3.5, and Section 6 of the report will remove the word “elemental” from the phrase “the assumption that no ~~elemental~~ iodine decontamination factor limit should be applied to natural aerosol removal phenomenon in the NuScale containment.” Additionally, Section 4.3.5 of the report will be revised to clarify “Aerosol and Elemental Iodine Removal” and to state that NuScale does not take credit for elemental iodine removal.

NRC RAI Question Number: 01.05-21

NRC RAI Question

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.

NuScale RAI Question Response

Following the DBST event for the NuScale design, the minimum calculated post-accident $pH_T$ is 6.6. A $pH_T$ of 6.5 is assumed for the purpose of this RAI response. The nominal iodine concentration in the water inside the containment following the DBST event is calculated to be 1.53E-4 gram-atoms/liter for the NuScale design. Using the equations for iodine re-evolution from Section 3.2 of NUREG/CR-5950 (Reference 7.2.52), which is the basis for Figure 4-24 of the report; an iodine re-evolution of approximately 0.05 percent for a $pH_T$ of 6.5 is calculated.

A sensitivity study was performed that accounted for the contribution of this approximately 0.05 percent re-evolution of iodine in containment to the dose, as shown in the table below.

Table 1 – Sensitivity of Dose to Iodine Re-evolution

| Case | Control Room | Technical Support Center | Exclusion Area Boundary | Low Population Zone |
|---|---|---|---|---|
| | Acceptance Criteria | | | |
| | <5 rem | <5 rem | <25 rem | <25 rem |
| | Dose (rem TEDE) | | | |
| Baseline DBST dose | 2.29 | 1.30 | 0.50 | 1.44 |
| Contribution to Dose from Iodine Re-evolution | 0.015 | 0.009 | 0.001 | 0.010 |
| **Percent of Baseline DBST dose** | 0.67% | 0.66% | 0.15% | 0.67% |

The dose caused by iodine re-evolution in containment is shown to be less than 0.7 percent of the baseline DBST dose that did not include iodine re-evolution, demonstrating that the contribution from iodine re-evolution is negligible. This sensitivity study did not credit any decontamination of the re-evolved iodine, which is conservative.

Impact of NRC RAI Question Response on TR-0915-17565:

This RAI response does not require a revision to the report.

NRC RAI Number: 11 NRC RAI Date: December 23, 2016

NRC Review of: Accident Source Term Methodology, TR-0915-17565-P, Revision 1

NRC RAI Question Number: 01.05-31

NRC RAI Question

In the NRC staff's review of the topical report (TR), "Accident Source Term Methodology," TR-0915-17565-P, Rev. 1, the staff requires the following information to complete its review. Also, the staff requests that the requested information be included in the subject topical report, as appropriate.

1. Pages 5 and 64: Considering the NuScale design's unique geometry and dimensions, it is not clear that temperatures in the containment would remain low enough to prevent re-vaporization during a core-melt accident. Please describe the basis for assuming no re-vaporization of aerosols in containment.

2. Page 32: The TR states "For both Category 1 and Category 2 event radiological analyses, the containment is assumed to leak at half of the design basis limit leak rate after 24 hours and then at half of the design basis limit leak rate thereafter." The TR does not provide NuScale design-specific justification for this assumption. Please provide the basis for the assumption that the leak rate reduces after 24 hrs., including any necessary analysis to support the NuScale assumption.

3. Page 60: Table 4-5 gives an inert species ratio range from 2 to 4.The TR states that the range is based on experiments performed for light water reactor (LWRs).Please clarify how these experiments are applicable to the NuScale design which is a unique design compared with large LWRs.

4. Page 78: Table 4-9 shows an inert species ratio range from 6.8 to 68. Please clarify the basis for the inert species ratio range in Table 4-9 including why it is different from the inert species ratio from 2 to 4 in Table 4-5.

5. Page 78: The TR includes sensitivity calculations using a range of values shown in Table 4-9 for each of the following parameters: CNV volume, sedimentation area, diffusion area, condensation area, CNV radius and CNV height. Because the values of these parameters are fixed by the geometry of the NuScale design, please clarify why the values of these parameters were varied in the sensitivity calculations.

6. Page 89: Please elaborate and provide clarification on the purpose of the TR's example sensitivity analysis.

NuScale RAI Question Response

1. As stated in Section 4.3.6 of the report, Reference 7.2.43 provides vapor pressures for key fission products (notably those containing iodine) as a function of temperature. The

vaporization pressure at atmospheric pressure (lower pressure is conservative for this evaluation) is in the range of 1500-2200 degrees F (1100-1500K). In a plot of containment temperatures for an example severe accident (Figure 5-10), the containment air, wall, and pool temperatures are at a maximum of approximately 550 degrees F and reach temperatures of roughly 200 degrees F at 24 hours. As 550 degrees F is much less than 1500-2200 degrees F, re-vaporization is a negligible transport mechanism in the containment. Although the aerosol transport modeling methodology does not include re-suspension and re-vaporization effects explicitly, re-suspension and re-vaporization effects are inherently included in the benchmarked experiments.

2. Chapter 6 of the FSAR provides the justification for this assumption. The description of the design basis for containment is provided in Section 6.2 of the NuScale FSAR, which states that for "all postulated events, containment pressure is shown to be reduced to less than 50 percent of the peak calculated pressure in less than 24 hours after the postulated accident.…Specifically, for the limiting peak pressure case, the CNV pressure is reduced to less than 50 percent of its peak value in less than two hours." NuScale followed the guidance of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" for modeling CNV leakage. Specifically, RG 1.183 Appendix A, Paragraph 3.7 indicates that "the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate" for evaluating consequences of a LOCA. This assumption is supported by the results of the CNV analysis.

3. Reference 7.2.43 of the report provides an estimated percentage of fission products, control, and structural material inventory of core materials for Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) designs based on chemical equilibrium calculations. This information yields an estimate for the inert species ratio of 4 (reciprocal of 25 percent). As indicated in Table 4-5, a range of two to four has been measured. The NuScale fuel design is based on a conventional 17x17 PWR fuel design, with similar geometries, materials, and fuel composition. While the fuel assembly is half the height of a typical PWR fuel assembly, the ratio of control materials and fission products is similar. The NuScale design contains a higher ratio of structural materials, however only a small amount of structural materials (approximately 2-3 percent for conventional LWRs) would be contained in the aerosols. Therefore, the range of the inert species ratio for conventional PWRs and BWRs is directly applicable to the NuScale design.

Table 2 – Estimated Percentage of Structural, Control, and Fission Product Material (from Reference 7.2.43)

| Material | Estimated Percentage | |
|---|---|---|
| | PWR | BWR |
| Structural | 3 | 2 |
| Control | 73 | 75 |
| Fission Products | 24 | 23 |

4. The inert species ratio range in Table 4-9 of the report was found to contain a typographical error. The inert species ratio in Table 4-9 should be consistent with Table 4-5 (i.e., 2 to 4). Other typographical errors were also identified in Table 4-9 that require correction.

5. As described in the response to RAI Question Number 1.05-31(6), sensitivity analyses were used to determine the appropriate biasing direction for an input and to check for the possibility of a non-linear system with corresponding feedback effects. Sampling of these "fixed" parameters allowed for a high confidence that a conservative input, including the effect of tolerance, is utilized and that the methodology is robust (i.e., there are no non-linear or "cliff edge" effects that result in an unexpected trend not accounted for in the development of the methodology).

6. General sensitivity analysis methodologies are used throughout Section 4.4.1 of the report. The sensitivity analyses are used in the accident radiological methodology to determine the appropriate biasing direction for an input, in order to obtain a conservative solution and to determine relative importance of the inputs. Another important use of this sensitivity analysis is to check for the possibility of a non-linear system with corresponding feedback effects. It is important to check for a non-linear system as parameter importance and bias directions are not consistent in such systems. Thus, in the case of a non-linear system, conditional bias directions have to be established in order to ensure a robust deterministic methodology. A review of the input parameters ranked in importance by linear, rank, and quadratic regression indicates clear consistency for each case individually, but also between different cases. This is a strong indication that the analysis of the NuScale design, as performed with a deterministic methodology, results in the modeled system behaving largely as a linear system (i.e., consistent importance and bias directions). A statistically based nonparametric input sampling process is implemented through automated software tools. Input parameters are randomly varied across their pertinent range of values within the input deck and run as a single sample whose output file gives a value.

Impact of NRC RAI Question Response on TR-0915-17565:

In response to RAI Question Number 1.05-31(4), several typographical errors in Table 4-9 will be corrected. As a result, the inert species ratio is consistent with Table 4-5.

In response to RAI Question Number 1.05-31(6), Section 4.4.1 will be revised to include additional clarification on how the example sensitivity analyses were used in the development of the report methodology.

## Enclosure 2:

Redline/strikeout pages of NuScale Topical Report TR-0915-17565, “Accident Source Term Methodology,” Revision 1, proprietary version

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## Enclosure 3:

Redline/strikeout pages of NuScale Topical Report TR-0915-17565, “Accident Source Term Methodology,” Revision 1, nonproprietary version

## Abstract

This NuScale topical report describes the methodology used for establishing the source terms and radiological consequences for a spectrum of design basis accidents. In instances where significant differences between the NuScale small modular reactor design and a large light water reactor cause the methodology to depart from existing regulatory guidance, these departures are justified in detail.

A methodology for establishing the NuScale design basis source term (DBST) release timing and magnitude, which meets the intent of 10 CFR 52.47 (a)(2)(iv), is presented in this report. DBST associated aerosol transport and iodine re-evolution assessment methodologies are also presented. Approval is sought for application of STARNAUA aerosol modeling software for NuScale's range of post-accident containment conditions and the assumption that no ~~elemental~~ iodine decontamination factor limit should be applied to natural aerosol removal phenomenon in the NuScale containment. Approval is also sought for the use of ARCON96 for establishing offsite atmospheric dispersion factors. This topical report is not intended to provide final DBST isotopic inventory values, dose values, atmospheric dispersion factors, or final values of any other associated accident source term evaluation; rather, example values for the various evaluations are provided for illustrative purposes in order to aid the reader's understanding of the context of the application of these methodologies.

Calculations associated with the DBST take credit for the natural aerosol removal mechanisms inherent in the NuScale containment design. The STARNAUA containment aerosol transport and removal code was benchmarked against experimental data and was shown to be appropriate for modeling aerosol removal in the DBST analysis associated with the post-accident NuScale containment conditions. Consistent with RG 1.183, NuScale utilizes the assumption that no ~~elemental~~ iodine decontamination factor limit should be applied to natural aerosol removal phenomenon for modeling removal in containment. Through sensitivity analysis on the modeling parameters utilized as input to the STARNAUA code, it was shown that the wide range of valid aerosol modeling parameters utilized were of equal or less importance for the DBST radiological consequence results compared to other key modeling parameters. This insight should reduce the relative importance of the particular aerosol modeling inputs selected for the DBST analysis in the NuScale design certification application.

Example calculations are provided in this report to demonstrate applicability of the methodology and to aid the reader's understanding of the application of these methodologies. The NuScale design certification application referencing this topical report is expected to present design-specific calculations utilizing the methodologies presented herein.

# 1.0 Introduction

## 1.1 Purpose

The purpose of this report is to define and justify the methodology for assessing the source terms and radiological consequences of design basis accidents (DBA). NuScale plans to use this methodology in the NuScale design certification application, such as in Chapter 15 of the forthcoming NuScale Design Control Document (DCD). NuScale requests NRC approval that the assumptions, codes, and methodologies presented in this report are technically acceptable and consistent with current regulations.

## 1.2 Scope

This report describes assumptions, codes, and methodologies utilized to calculate the radiological consequences of design basis accidents. NuScale seeks approval for the methodology for establishing the NuScale DBST release timing and magnitude that meets 10 CFR 52.47 (a)(2)(iv). NuScale also seeks approval of the DBST associated aerosol transport and iodine re-evolution assessment methodologies. Approval is requested for application of STARNAUA aerosol modeling software to NuScale's range of post-accident containment conditions and the assumption that no ~~elemental~~ iodine decontamination factor limit should be applied to natural aerosol removal phenomenon in the NuScale containment. Approval is also requested for the use of ARCON96 for establishing offsite atmospheric dispersion factors instead of PAVAN.

This topical report is not intended to provide final DBST isotopic inventory values, final dose values, final atmospheric dispersion factors, or final values of any other associated accident evaluation; rather, example values for the various evaluations are provided for illustrative purposes. Radiological consequence dose results and comparisons to 10 CFR 52.47 and General Design Criterion (GDC) 19 limits are provided for illustration to aid the reader's understanding of the context of the application of these methodologies.

A summary of specific positions that NuScale is seeking approval for in this topical report are as follows:

1. Use of ARCON96 methodology for the calculation of offsite atmospheric dispersion factors.
2. {{ }}2(a),(c)
3. Minimum time to core damage from the spectrum of STDBAs is taken as the DBST time to core damage.
4. Minimum release duration from the spectrum of STDBAs is taken as the DBST release duration.
5. Representative (median) release fractions from fuel into containment from the spectrum of STDBAs are taken as the DBST release fractions.
6. Use of SAND2011-0128 radionuclide groups for the DBST.

7. STARNAUA is appropriate for modeling natural removal of containment aerosols for the NuScale design.
8. No maximum limit on ~~elemental~~ iodine decontamination factor for natural removal of containment aerosols.
9. {{ }}2(a),(c)
10. Utilizing the iodine spiking assumptions of RG 1.183 is appropriate.
11. Generalized process for determining the analytical effective decontamination factor based on a minimum depth of water above the damaged fuel in a fuel handling accident.
12. With respect to accident analysis, it is appropriate to neglect the small secondary side volume that could contain activity from primary to secondary leakage for the NuScale design.
13. For $pH_T$ values of 6.0 or greater, the amount of iodine re-evolution that could occur between $pH_T$ values of 6.0 and 7.0 is negligible and not included in the dose calculation.
14. Containment shine of the radiation in the containment airspace through the containment vessel, reactor pool water, and then through the reactor building walls or ceiling to the environment is negligible for the NuScale design.

## 1.3 Abbreviations

Table 1-1. Abbreviations

| Term | Definition |
|---|---|
| ALWR | advanced light water reactor |
| AST | alternative source term |
| Bq | Becquerel (unit of radioactivity) |
| Ci | curie (unit of radioactive decay) |
| μCi | micro-Curie (1.0E-06 Ci) (unit of radioactive decay) |
| cfm | cubic feet per minute (unit of flow) |
| COL | combined license |
| CR | control room |
| CVCS | chemical and volume control system |
| DBA | design basis accident |

## 3.1 Software

### 3.1.1 SCALE 6.1/TRITON/ORIGEN-S

SCALE 6.1 modular code package, developed by Oak Ridge National Laboratory, is used for development of reactor core and primary coolant fission product source terms. Specifically, the TRITON and ORIGEN-ARP analysis sequences of the SCALE 6.1 modular code package, and ORIGEN-S, run as a standalone module, are used to generate radiation source terms for the NuScale fuel assemblies and primary coolant (Reference 7.2.25). The aforementioned software has been used in the evaluation of operating large LWRs. The operating environment, nuclear fuel and structural materials in the NuScale design are expected to be similar to, or bounded by, that in large pressurized water reactors (PWR).

#### 3.1.1.1 TRITON

As described in the SCALE manual (Reference 7.2.25), the TRITON sequence of the SCALE code package is a multipurpose control module for nuclide transport and depletion, including sensitivity and uncertainty analysis. TRITON can be used to generate problem- and exposure-dependent cross sections as well as perform multi-group transport calculations in one-dimensional, two-dimensional, or three-dimensional geometries. The ability of TRITON to model complex fuel assembly designs improves transport modeling accuracy in problems that have a spatial dependence on the neutron flux. In this case, TRITON is used to generate burnup-dependent cross sections for NuScale fuel assemblies for subsequent use in the ORIGEN-ARP depletion module.

#### 3.1.1.2 ORIGEN (ORIGEN-ARP and ORIGEN-S)

Reference 7.2.25 describes ORIGEN-ARP as a SCALE depletion analysis sequence used to perform point-depletion and decay calculations with the ORIGEN-S module using problem- and burnup-dependent cross sections. ORIGEN-S nuclear data libraries containing these cross sections are prepared by the ARP module using interpolation in enrichment and burnup between pre-generated nuclear data libraries containing cross section data that span the desired range of fuel properties and operating conditions. The ORIGEN-ARP sequence produces calculations with accuracy comparable to that of the TRITON sequence with a savings in problem setup and computational time as compared to repeated use of TRITON. Many variations in fuel assembly irradiation history can be modeled. For depletion calculations involving NuScale fuel assemblies, the ORIGEN-S nuclear data libraries are generated by the TRITON sequence, as described in the previous Section 3.1.1.1.

### 3.1.2 ARCON96

The calculation of both onsite and offsite atmospheric dispersion factors for design basis accidents is performed with NARCON~~96 (Reference 7.2.24)~~. NARCON is the NuScale version of ARCON96 (Reference 7.2.24). NARCON is equivalent to ARCON96 with input/output edit differences only. The program implements the guidance provided in RG 1.194 (Reference 7.2.8). The code implements a building wake dispersion algorithm; an

included to permit modeling of plant controls, turbines, condensers, and secondary feedwater systems. NRELAP5 ~~is~~ was developed at NuScale, with RELAP5-3D© v.4.1.3 as the initial baseline. RELAP5-3D© v.4.1.3 was procured from the Idaho National Laboratory through a commercial grade dedication process. Upon dedication, the RELAP5-3D© v.4.1.3 code was renamed ~~to~~ NRELAP5 and further developed by NuScale.

### 3.1.6 STARNAUA

Aerosol transport and removal calculations are provided by the program STARNAUA. STARNAUA is an aerosol transport and removal software program that was developed by Polestar Applied Technology, Inc., a company later purchased by WorleyParsons. STARNAUA is an enhanced version of NAUAHYGROS and was developed by Polestar for performing aerosol removal calculations in support of work to develop and apply a realistic source term for advanced and operating LWRs.

It models natural removal of containment aerosols by gravitational settling and diffusiophoresis, and considers the effect of hygroscopicity (growth of hygroscopic aerosols due to steam condensation on the aerosol particles) on aerosol removal. In developing STARNAUA, Polestar enhanced NAUAHYGROS by adding a model for thermophoresis, a model for spray removal, and the capability to directly input steam condensation rate or condensation heat transfer rate, and total heat transfer rate such as would be provided from an external containment thermal hydraulics code calculation.

This software was developed for the purpose of performing aerosol removal calculations to apply in realistic source terms for advanced and operating LWRs. This realistic source term methodology is consistent with existing industry practice used for large passive LWR design certification.

### 3.1.7 $pH_T$

The Fortran program developed by NuScale to calculate post-accident aqueous molar concentration of hydrogen ions ($pH_T$) is called “$pH_T$”. This program calculates $pH_T$ utilizing the methodology described in Section 4.5. This program takes inputs for initial boron and lithium concentrations, the total core inventory of iodine and cesium, the integrated photon dose to the containment and total dose to the coolant, the initial mass of coolant, the mass of coolant, and the temperature of the coolant. The program then calculates the coolant $pH_T$ as a function of time.

### 3.1.8 MCNP6

MCNP6 is utilized for evaluating potential shine radiological exposures, or doses, to operators within the control room following a radiological release event. Direct shine, sky-shine, and shine from all possible filters are evaluated. MCNP is a general-purpose tool used for neutron, photon, electron, or coupled neutron, photon, and electron transport (Reference 7.2.27). MCNP treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-

4. Primary coolant leaks into both steam generators at the maximum leak rate allowed by design basis limits. The leakage continues until the reactor is shut down and depressurized and the primary and secondary systems are at an equal pressure.
5. Activity is released to the environment through the condenser until isolation is achieved.
6. Leakage through the secondary isolation valves (main steam and feedwater) occurs in the reactor building until the reactor is shut down and depressurized. No credit is taken for any source term reduction within the reactor building.

The following is a summary of the assumptions used from Appendix H of RG 1.183:

- containment iodine chemical form of 95% cesium iodide, 4.85% elemental iodine, and 0.15% organic iodide
- primary system iodine chemical form of 97% elemental iodine and 3% organic iodide
- no reduction or mitigation of noble gas radionuclides released from the primary system
- density for leak rate conversion: 62.4 pound mass (lbm)/ft$^3$

### 3.2.2 Fuel Handling Accident

The methodology for determining FHA radiological consequences is based on the guidance provided in Appendix B of RG 1.183 and Section 15.7.4 of the SRP. The explicit guidance enumerated in Appendix B of RG 1.183 is followed with one exception, which is that the iodine decontamination factor will be calculated with a generalized methodology instead of utilizing the prescribed RG 1.183 values for a depth of water above the damaged fuel of 23 feet or greater. The methodology assumes failure of all the fuel rods in one irradiated fuel assembly occurs.

As presented in Section 3.3.8 of this report, the NuScale reactor pool has a minimum depth above the damaged fuel greater than the minimum 23 foot depth specified as the basis for the iodine decontamination factor in Reference 7.2.11. Therefore, a generalized methodology for calculating increased decontamination factor was used, and is based on the methodology and assumptions of Reference 7.2.12. This methodology is presented in more detail in Section 3.3.8.

The following is a summary of the assumptions used from Appendix B of RG 1.183:

- radionuclides considered include xenon, krypton, halogen, cesium, and rubidium
- ~~iodine chemical form of 57 percent elemental iodine and 43 percent organic iodide~~
- no reduction or mitigation of noble gas radionuclides released from the fuel
- release to the environment over a two hour period

The standard activity release period of two hours is used for the dose assessment. This period is the standard assumption provided in Section 4.1 of Appendix B of RG 1.183,

### 3.3.4.3 Reactor Building Decontamination

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}}2(a),(c)

### 3.3.4.4 Pre-Accident Coolant Radiation Levels

For events in which the source of radiation is from damaged fuel, it is assumed that the primary and secondary coolant radiation levels are zero prior to the accident level with the exception of pre-incident iodine spiking scenarios required for evaluation. This assumption is in accordance with RG 1.183 (Reference 7.2.2) that prescribes the source term assumptions and isotopes to be used for each event. In particular, the fuel releases are assumed to occur and mix instantaneously within the reactor coolant system. For events in which the source of radiation is from primary coolant, the primary coolant radiation levels prescribed by RG 1.183 are utilized.

### 3.3.4.5 Control Room Exhaust

The control room exhaust is equal to the total inlet flow in both the normal and emergency operating modes. For the pressure to remain in equilibrium, an equal amount of air must be exhausted. This assumption takes credit for air escaping through doors when open or other potential leakage pathways, such as penetrations in the control room envelope.

### 3.3.4.6 Radiation Shine Radiological Consequences

Per Section 4.2.1 of RG 1.183 (Reference 7.2.2), the following contributions of radiation shine to the control room dose are included in the methodology:

- radioactive material in systems and components inside or external to the control room envelope. This is typically applicable to radionuclides collected in filters
- sky shine from an external radioactive plume released from the facility
- direct shine from airborne fission product gases within the reactor building and contamination of structural surfaces within the reactor building

The calculated shine dose from the bounding event is applied to all events rather than specifically calculating the shine dose for each event. For the event that results in the largest activity released into the reactor building and control room, the radionuclide activities calculated by RADTRAD at key time intervals throughout the event are utilized as input into ORIGEN-S and MCNP models.

The photon source terms are defined according to a user-supplied energy group structure, which consists of a series of energy bins over which average source strengths

$F_{inorg}$ = Fraction of inorganic iodine
$F_{org}$ = Fraction of organic iodine
$H$ = Height of bubble rise (i.e., bubble rise height)
$k_{eff}$ = Effective flow characteristics of bubble
$v_b$ = Rise velocity of a bubble from pressurized source

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Eq 3-19

}}2(a),(c)

The exponential term in the non-hygroscopic condensational growth equation represents the lowering of the saturation pressure at the surface of the droplet due to the curvature of the droplet, the so-called Kelvin effect. If the particle is hygroscopic or soluble, then there is an additional decrease in the saturation pressure due to the solute effect. This leads to an additional water activity term $A$ in the numerator of the equation. For dilute solutions with many hygroscopic species, the water activity can be determined by the following Reference 7.2.45 equation.

$$A = \frac{1}{1 + \sum_i Q_i M_w m_i} \quad \text{Eq 4-13}$$

Where $Q_i$ is the van't Hoff factor and $m_i$ is the molality of each hygroscopic material dissolved in the droplet.

### 4.3.5 Aerosol and Elemental Iodine Removal

Deposition of ~~elemental iodine~~aerosol onto inner-containment surfaces results in ~~its~~ removal~~ as a source term for release to the environment~~. A key assumption of the NuScale aerosol transport methodology is ~~this report's aerosol transport methodology is that there is ~~no maximum iodine decontamination factor limit ~~on elemental iodine decontamination factor~~should be applied to natural aerosol removal phenomenon in the NuScale containment. The basis for this assumption is that this removal is facilitated by natural processes, as opposed to an active spray system. Additionally, as described in the following sections, the NuScale removal rate calculation methodology ~~for this report~~ is based on calculated time-dependent airborne aerosol mass. This ~~methodology ~~is in agreement with Section 3.3 of Appendix A of RG 1.183 in which it is stated that "…reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass." ~~The ~~NuScale ~~methodology~~ conservatively does not take credit for elemental iodine removal. Rather, only aerosol removal is credited.

### 4.3.6 Aerosol Resuspension and Revaporization

Resuspension of aerosol particles from deposition surfaces is generally considered a transport mechanism of importance only for pipes. Additionally, resuspension is based on turbulent flow conditions (Reference 7.2.43), {{

}}2(a),(c)

Revaporization is considered a negligible transport mechanism in the containment. It is noted in Reference 7.2.43 that revaporization is unlikely for fission products of interest, considering maximum post-LOCA temperatures compared against vaporization temperature/pressure curves for fission products of interest.

Although the aerosol transport modeling methodology does not include resuspension and revaporization effects explicitly, resuspension and revaporization effects are inherently included in the benchmarked experiments. {{ }}2(a),(c)

### 4.3.10 Benchmarking to MAEROS

NMAEROS, a component of MELCOR, is the NuScale version of MAEROS (October, 1982). NMAEROS is equivalent to MAEROS (October, 1982) with input/output edit differences only. MAEROS ~~may be~~is used to simulate the evolution of an aerosol based on the particle mass. Physical phenomena included in the calculations are Brownian coagulation, gravitational coagulation, diffusion, thermophoresis, and deposition. Models for diffusiophoresis and hygroscopicity are not included in the MAEROS code.

As described in Reference 7.2.65, the general numerical approach of MAEROS uses sections (bins) defined by particle mass instead of particle dimensions. For N sections, MAEROS calculates a set of 2N(N+2) sectional coefficients. Each coefficient is a rate constant for a different transport mechanism. These coefficients, for a given containment, often depend only on gas temperature and pressure. Given this condition, the user can specify upper and lower temperature and pressure limits. These limits give four sectional coefficients corresponding to the four combinations of temperature and pressure limits. MAEROS is able to linearly interpolate temperature and pressure values that differ from the limits and perform calculations with aerosol components, where a component is a physical constituent of the aerosol.

As described in Reference 7.2.43, MAEROS was originally developed in part to eliminate numerical diffusion that occurred in early aerosol codes through the use of a "moving boundary" numerical scheme for bin sizing. The aerosol sections, or bins, are defined by particle mass in MAEROS, as opposed to particle size as in STARNAUA. As a result, the numerical solution of the code, including time step sizes, is fundamentally different than that of STARNAUA. However, both codes utilize similar mechanics for coagulation and diffusion. As such, MAEROS is appropriate for an independent code-to-code benchmark of STARNAUA with respect to the numerical solution of aerosol coagulation and diffusion.

For the benchmark, a hypothetical test roughly inspired by the LACE experiment was crafted. In order to compare the models, the hygroscopic and diffusiophoresis models in STARNAUA were turned off. The only difference between Case 1 and 2 is the aerosol source rate. Case 1 is prescribed a CsOH source rate of 6.0 grams/sec while Case 2 provides a source rate of 0.6 grams/sec. The test geometry that the geometric inputs are derived from is shown in the following table.

## 4.4 DBST Sensitivity Analysis

### 4.4.1 General Sensitivity Analysis Methodology

General sensitivity analysis methodologies are used throughout this section and are provided herein for reference. The sensitivity analyses are used in the accident radiological methodology to determine the appropriate biasing direction for an input, in order to obtain a conservative solution and to determine relative importance of the inputs. Another important use of this sensitivity analysis is to check for the possibility of a non-linear system with corresponding feedback effects. It is important to check for a non-linear system as parameter importance and bias directions are not consistent in such systems. Thus, in the case of a non-linear system, conditional bias directions have to be established in order to ensure a robust deterministic methodology. A review of the input parameters ranked in importance by linear, rank, and quadratic regression indicates clear consistency for each case individually, but also between different cases. This is a strong indication that the analysis of the NuScale design, as performed with a deterministic methodology, results in the modeled system behaving largely as a linear system (i.e., consistent importance and bias directions). A statistically based nonparametric input sampling process is implemented through automated software tools. Input parameters are randomly varied across their pertinent range of values within the input deck and run as a single sample whose output file gives a value.~~Section 4.4 and are briefly mentioned here for convenience.~~

A statistically based nonparametric input sampling process is implemented through automated software tools. Input parameters are randomly varied across their pertinent range of values within the input deck and run as a single sample whose output file gives a value. Scatter plots are created to visually illustrate the sensitivity of the system to each individual input and sensitivity metrics are calculated to assist in identifying trends seen in the scatter plots.

The partial rank correlation coefficient (PRCC) value is also calculated. A positive PRCC value means that the effect of the input on the output is the same, i.e., an increase of the value of the input leads to an increase of the value of the output. A negative PRCC value means the effect is the opposite, i.e., an increase in the input leads to a decrease in the output.

The primary indicator of importance is the incremental $R^2$ from the quadratic regression model. An input is not sufficiently important if it has an incremental $R^2$ less than 0.02. A high incremental $R^2$ (close to 1.0) indicates that an input is highly influential on the evaluated system output.

### 4.4.2 Application to DBST

General sensitivity analysis methodologies are applied to the DBST calculation, including the aerosol modeling component, in order to provide a quantitative evaluation of the impact of aerosol modeling on the key output of the DBST calculation, which is the radiological consequence.

Table 4-9. Summary of sampled input assumed for sensitivity analysis

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| Parameter | Units | Min | Max | Description |
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}}2(a),(c) ~~ECI~~

Observations of the quadratic regression, PRCC, and adjusted $R^2$ values for the example sensitivity analysis are described in the following paragraphs. In all cases the adjusted $R^2$ indicated fair to good performance with quadratic values ranging approximately from 0.7 to 0.9, depending on the number and type of inputs for a case. Considering the fair to good performance of the adjusted $R^2$, non-linear effects with respect to sensitivity analysis are ruled out, and the resulting sensitivity analysis is taken to be reasonable. A minimum of 1,000 samples were utilized in the analysis and taken to be appropriate based on acceptable adjusted $R^2$ values.

The quadratic regression criteria consistently aligned across all cases and with the PRCC rankings. {{

}}2(a),(c)

As noted previously, one use of this analysis is to check for the possibility of a non-linear system with corresponding feedback effects. It is important to check for a non-linear system as parameter importance and bias directions are not consistent in such systems. Thus, in the case of a non-linear system, conditional bias directions would have to be established in order to ensure a robust deterministic methodology.

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}}2(a),(c)

## 6.0 Summary and Conclusions

A methodology for developing accident source terms and performing the corresponding radiological consequence analyses was presented in this report. The methodology was shown to be conservative by providing example results from sensitivity analyses. Key unique features of the methodology presented in this report are the use of ARCON96 for offsite atmospheric dispersion factors, the development of a DBST to meet the intent of 10 CFR 52.47 (a)(2)(iv), the utilization of STARNAUA containment aerosol transport code in the range of NuScale's expected post-accident containment conditions, and evaluation of post-accident $pH_T$.

ARCON96 was found to be suitably conservative for NuScale's intended use of the code as a substitute for PAVAN for offsite atmospheric dispersion factor calculations. As presented in this report, the ARCON96 methodology has less of a tendency to over-predict concentrations, while still providing predictions that are sufficiently conservative.

The STARNAUA containment aerosol transport and removal code was benchmarked against experimental data and was shown to be appropriate for modeling aerosol removal in the DBST analysis associated with the post-accident containment conditions. Consistent with RG 1.183, the assumption that no ~~elemental~~ iodine decontamination factor limit should be applied to natural aerosol removal phenomenon was utilized for modeling removal in the containment vessel. Through example sensitivity analysis on the modeling parameters utilized as input to the STARNAUA code, {{

}}2(a),(c)

This insight provides confidence that the aerosol modeling methodology is robust and the inputs utilized for the DBST analysis are conservative in the NuScale design certification application.

Example calculations were provided in order to demonstrate applicability of the methodology.

## 6.1 Criteria for Establishing Applicability of Methodologies

The generalized methodologies presented in this topical report are based upon numerous modeling assumptions. For completeness, the following set of criteria for establishing the applicability of these methodologies is provided. The design certification application or combined operating license application that utilizes the methodology of this topical report must satisfy these criteria in order to establish applicability. Any deviations to these criteria must be explicitly defined and justified as part of the application that references this topical report.

## Enclosure 4:

Affidavit of Thomas A. Bergman, AF-0217-52946

**NuScale Power, LLC**

AFFIDAVIT of Thomas A. Bergman

I, Thomas A. Bergman , state as follows:

(1) I am the Vice President of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Information Responses reveal distinguishing aspects about the process and method by which NuScale develops its Accident Source Term Methodology.

NuScale has performed significant research and evaluation to develop a basis for this process and method and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the Enclosure 2 to NuScale letter titled "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." Enclosure 2 contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 6, 2017

Thomas A. Bergman