RAIO-0718-61146

July 31, 2018

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 Response to NRC Request for Additional Information No. 9224 (eRAI No. 9224) on the NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 2

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9224 (eRAI No. 9224)," dated December 04, 2017
2. NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 2, dated April 2016

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Question from NRC eRAI No. 9224:

- 01.05-32

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 9224 (eRAI No. 9224). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Jennie Wike
Manager, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9224, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9224, nonproprietary

Enclosure 3: Affidavit of Thomas A. Bergman, AF-0718-61147

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9224, proprietary

RAIO-0718-61146

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9224, nonproprietary

Response to Request for Additional Information

eRAI No.: 9224
Date of RAI Issue: 12/04/2017

NRC Question No.: 01.05-32

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 done, 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 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) 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

Request for additional information

In the NRC staff's review of the topical report (TR), "Accident Source Term Methodology," TR-0915- 17565-P, Rev. 2, the staff requires the following information to complete its review. Also, the staff requests that the requested information be included in the TR, as appropriate.

1. The proposed methodology for determining the design basis source term in Section 4.2.1 includes using the median release fraction from the MELCOR calculations for the more likely severe accident scenarios. The NuScale Final Safety Analysis Report (FSAR) implements the methodology by taking the median over four MELCOR calculations. What is the basis for using the median release fraction as opposed to another statistical metric such as a mean or a 75th percentile? Given the limited number of scenarios (four) as implemented in the FSAR, what is the basis for using a median release fraction to evaluate the offsite radiological consequences as opposed to evaluating the offsite radiological consequences for each of the four scenarios?

2. The second paragraph of Section 4.2.4 describes the methodology {{

}}2(a),(c)

{{ }}2(a),(c) However, the methodology is not clear to the staff. The applicant is requested to clarify {{

}}2(a),(c)

3. Enclosure 1 to NuScale letter to NRC dated February 6, 2017, provided clarification of the basis of its use of an inert species ratio (i.e., ratio of non-radioactive aerosol to radioactive aerosol) of 2 to 4. As part of the clarification, NuScale stated that past studies showed an inert species ratio of 2 to 4 and that the NuScale fuel design is based on a conventional PWR fuel design with similar geometries, materials, and fuel composition with a similar ratio of control material to fission products. The staff's subsequent review of additional NuScale documents as part of the design certification review suggests that the inert species ratio could be different from that seen in the past studies because NuScale's core is different from previous PWR designs in two ways – it does not use 1% tin in its cladding and it mainly uses B4C control rods. The applicant is requested to provide additional information supporting the use of a ratio of 2 to 4 given these two differences.

4. With respect to the example implementation of the methodology on aerosol removal, the two curves (aerosol concentration and aerosol removal rate constant) in Figure 5-13 of the TR appear to be inconsistent with each other. For the period beginning at the end of the release, the decrease in the aerosol concentration has a lower removal rate than shown in the removal rate constant curve. Please clarify this apparent inconsistency and how the curves were calculated.

NuScale Response:

1. NuScale now utilizes the term "core damage maximum hypothetical accident (MHA)" instead of the term "design basis source term".

It should be noted that the number of severe accident scenarios implemented in determining the core damage MHA had been updated from four to five, as detailed in the previously submitted May 24, 2017 response to RAI 8774, Question 15.00.03-1 (ML17144A451). The example core damage MHA analysis shown in TR-0915-17565-P, Rev. 3 will be updated to use a set of five severe accident scenarios.

As noted in Section 4.2.3 of TR-0915-17565-P, Rev. 2, the use of the median is similar to the approach used in Sandia National Laboratory report SAND2011-0128 (Reference 7.2.10 of TR-0915-17565-P, Rev. 2). SAND2011-0128 gives the following justification for use of the median release fraction:

"The median is taken to be the representative value of the source term distribution… A percentile other than the median as the representative magnitude would require justification from some other source. By adopting the median, half the accidents have larger release fractions and longer release times than the representative accident and half have smaller release fractions and shorter release times."

While SAND2011-0128 notes as a limitation that "results are not applicable to small modular reactors that could have accident processes that differ substantially from those of the large power plants considered here", this limitation applies to results rather than evaluation methods. It is NuScale's view that the SAND2011-0128 release fraction selection method represents an applicable precedent of a solution approach to the problem of selecting representative release fractions from multiple postulated severe accident scenarios.

It is further noted that the implemented median release fractions from the MELCOR severe accident scenarios in the example core damage MHA analysis are conservatively higher than the mean release fractions from the MELCOR severe accident scenarios for all chemical groups, as can be observed in Table 1 of this RAI response (which will replace Table 5-9 of TR-0915-17565-P, Rev. 2 in the forthcoming TR-0915-17565-P, Rev. 3, with the exceptions that the "mean" column of Table 1 of this RAI response will not be included and the RG 1.183 and SAND 2011-0128 columns will be retained in the equivalent table in TR-0915-17565-P, Rev. 3).

The suggestion of evaluating the offsite radiological consequences for each of the five scenarios individually would imply a search for a single worst case event. However, RG 1.183 regulatory position 2.3 states "The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events." NuScale sought to represent a spectrum of severe accidents by using a median release fraction, the minimum onset time for fission product release from the gap, and the minimum duration of the release determined from the spectrum of source term design basis accidents (STDBAs) to evaluate the offsite radiological consequences.

2. {{

}}2(a),(c)

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}}[2(a),(c)]

3. An inert species ratio range of 2-4 was investigated in Section 4.4.2 of TR-0915-17565-P, Rev. 2 and showed that a smaller value of inert ratio produces more conservative dose results (as expected). Only an inert species ratio of 2 is utilized in NuScale containment aerosol removal analysis associated with the core damage MHA. An inert species ratio value of 2 is derived from core degradation experimental and theoretical data (Reference 7.2.43 of TR-0915-17565-P, Rev. 2). A survey of core damage experiments described in Reference 7.2.43 of TR-0915-17565-P, Rev. 2 yields an inert-to-radioactive species mass composition ratio range of 1-3. Additionally, theoretical mass compositions based on a chemical equilibrium estimate of aerosolized core materials described in Reference 7.2.43 of TR-0915-17565-P, Rev. 2 yields an inert-to-radioactive species ratio greater than 3. An inert aerosol species ratio of 2 therefore constitutes an average representative modeling surrogate for all inert materials forming aerosols based on experimental data, and a conservative modeling surrogate for all inert materials forming aerosols based on chemical equilibrium estimates of aerosolized core materials.

Although it is observed that the NuScale fuel assembly design includes M5 material, which does not include tin and for which little core degradation experimental data exists, it can be inferred that the contribution of tin-based inert aerosols to total aerosol composition is largely offset by the contribution of boron-based aerosols in the expected post-accident NuScale module steam environment. Further,tin inclusions remain in the NuScale fuel assembly structural components and therefore the generation of tin-based aerosols from a postulated NuScale core degradation is not entirely precluded.

Although Section 6 of "NuScale-HTP2TM Fuel and Control Rod Assembly Designs", TR-0813-51127-P, Rev. 1 shows that approximately {{ }}$^{2(a),(c)}$ of the total absorber material volume in the NuScale control rod assembly is composed of B_4C absorber material, for which little core degradation experimental data exists, it is observed that the lower absorber material in the control rods is the more typical silver-indium-cadmium (Ag-In-Cd) material, and the control rod cladding is 304 stainless steel tubing with stainless steel end plugs welded to each end. Therefore, some amount of typical material effects associated with available experimental aerosol generation data that would tend to raise the inert species ratio would still be expected to occur. Further, the observation that B_4C absorber material could oxidize to form aerosols is offered by multiple sources (TR-0915-17565-P, Rev. 2 References 7.2.10 and 7.2.43 and Reference 1 of this RAI response). The theoretical contribution of boron control material to overall aerosol composition of boiling water reactors (BWRs) is estimated to be as high as 75.1% per page 387 of Reference 2 of this RAI response. It is noted that known experimental information for the contribution of boron control material to aerosol composition is limited to core degradation experiment FPT3 (Reference 1 of this RAI response). FPT3 involved a steam-poor environment resulting in the limited oxidation of boron control material and, in turn, an overweighting of the contribution to vaporized core fraction of structural elements such as tin compared to the expected steam environment of a NuScale STDBA.

It is additionally noted, as already discussed in Enclosure 1 to NuScale letter to NRC dated February 6, 2017 LO-0117-52870 (ML17037D391), that the NuScale design inherently contains a higher ratio of structural materials to radioactive materials than conventional pressurized water reactors and BWRs. Although the contribution of structural materials to total aerosol may be less significant than the contribution of control materials, the higher NuScale structural material proportion nonetheless constitutes an aspect of the NuScale inert species ratio which would be greater than typical.

Finally, it is noted that the primary benefit of inert species abundance from an aerosol deposition standpoint is that it increases the suspended particle concentration without increasing the radioactive material. This has the benefit of increasing the coagulation rate and therefore the sedimentation rate (due to particle size). This has less of an impact on diffusiophoresis and thermophoresis, which will be credited in TR-0915-17565-P, Rev. 3, which are more dependent on system conditions than on particle concentration. These phenomena have a greater impact on the NuScale design as the relatively cold pool that touches the containment can drive these processes more strongly than the atmospheric boundary of a traditional containment. Therefore, while a lower inert species ratio is more conservative, it is expected to be of less significance in aerosol removal for the NuScale design than for a traditional light water reactor.

Therefore, based on the preceding observations, NuScale's utilization of an inert species ratio value of 2 is a reasonable assumption.

4. Figure 5-13 of TR-0915-17565-P, Rev. 2 was in error because of an internal post-processing error in the STARNAUA software. The vendor has been notified and the error has been addressed. NuScale will update Figure 5-13 and other associated STARNAUA example results in TR-0915-17565-P, Rev. 3. Figure 2 of this RAI response shows Figure 5-13's replacement that will be incorporated into TR-0915-17565-P, Rev. 3.

Description		STDBA No. 1	STDBA No. 2	STDBA No. 3	STDBA No. 4	STDBA No. 5	Median	Mean
onset of gap release (hr)		17.6	3.8	8.1	6.2	21.3	8.1	11.4
duration of gap plus early in-vessel release (hr)		12.0	1.0	9.0	1.3	14.0	9.0	7.5
fraction of initial core inventory released into containment	noble gases	0.39	0.19	0.41	0.19	0.48	0.39	0.33
	halogens	0.21	3.5E-02	0.16	1.9E-02	0.14	0.14	0.11
	alkali metals	0.25	5.9E-02	0.22	3.1E-02	0.20	0.20	0.15
	alkaline earths	5.9E-03	2.8E-03	6.7E-03	2.4E-03	5.3E-03	5.3E-03	4.6E-03
	tellurium group	0.22	3.8E-02	0.16	2.3E-02	0.15	0.15	0.12
	molybdenum	6.4E-02	1.3E-02	5.3E-02	5.8E-03	4.9E-02	4.9E-02	3.7E-02
	noble metals	1.2E-03	1.2E-04	1.5E-03	4.9E-05	7.9E-04	7.9E-04	7.3E-04
	lanthanides	3.3E-08	2.6E-09	3.1E-08	1.1E-09	2.1E-08	2.1E-08	1.8E-08
	cerium group	3.3E-08	2.6E-09	3.1E-08	1.1E-09	2.1E-08	2.1E-08	1.8E-08

Table 1: Comparison of release timing and magnitudes of example STDBAs

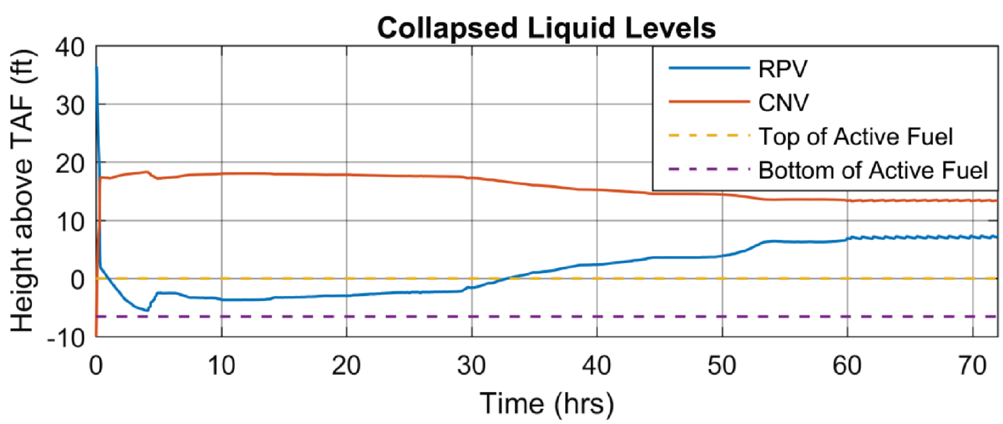

Figure 1: Example STDBA No. 2 reactor pressure vessel (RPV) and CNV collapsed liquid levels

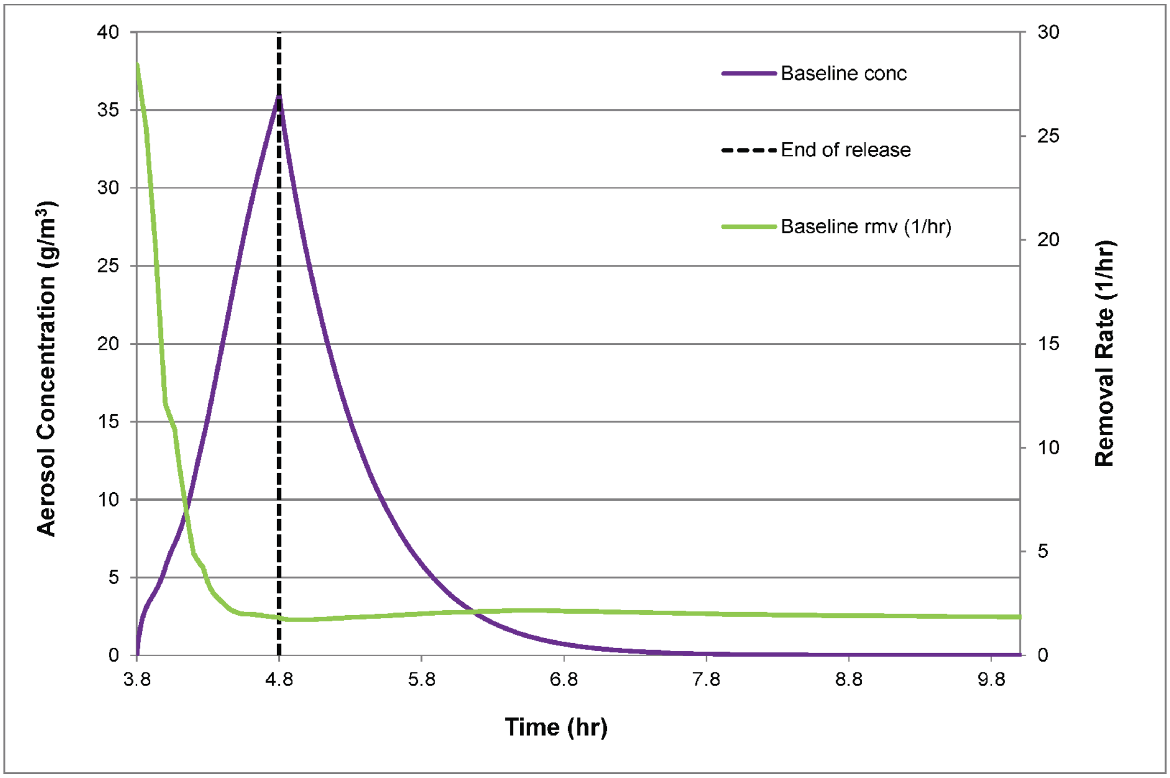

Figure 2: Baseline case aerosol concentration and removal

Reference 1: T. Haste, et al., "Phébus FPT3: Overview of Main Results concerning Core Degradation and Fission Product Behaviour", Paper 11147, Proceedings of ICAPP 2011, Nice, France, May 2011.

Reference 2: Robert P. Wichner & Roger D. Spence (1985) “A Chemical Equilibrium Estimate of the Aerosols Produced in an Overheated Light Water Reactor Core”, Nuclear Technology, 70:3, 376-393, DOI: 10.13182/NT85-A15964.

Additional Information:
TR-0915-17565-P will be revised as described in the response above.

Enclosure 3:

Affidavit of Thomas A. Bergman, AF-0718-61147

NuScale Power, LLC

AFFIDAVIT of Thomas A. Bergman

I, Thomas A. Bergman, state as follows:

1. I am the Vice President, 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 Additional Information response reveals distinguishing aspects about the methodology by which NuScale develops its accident source term.

   NuScale has performed significant research and evaluation to develop a basis for this methodology 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 enclosed response to NRC Request for Additional Information No, 9224, eRAI No. 9224. The enclosure 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 be 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 July 30, 2018.

Thomas A. Bergman