NuScale Power

RAIO-0418-59445

April 06, 2018

Docket No. 52-048

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. 234 (eRAI No. 9118) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 234 (eRAI No. 9118)," dated September 22, 2017
2. NuScale Technical Report Pressure and Temperature Limits Methodology, dated April 2018, TR-1015-18177

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 Questions from NRC eRAI No. 9118:

- 05.03.02-1
- 05.03.02-2
- 05.03.02-3
- 05.03.02-4
- 05.03.02-5

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 234 (eRAI No. 9118). 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 Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

Jennie Wike
Manager, Licensing
NuScale Power, LLC

Distribution: Samuel Lee, NRC, OWFN-8G9A
Prosanta Chowdhury NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9118, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9118, nonproprietary

Enclosure 3: Affidavit of Thomas A. Bergman, AF-0418-59446

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9118, proprietary

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9118, nonproprietary

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9118
Date of RAI Issue: 09/22/2017

NRC Question No.: 05.03.02-1

DCD Tier 2 Section 5.3.2.1 states that information on the specific methodology used to determine operational pressure limits on pressure and temperature for the reactor coolant pressure boundary are described in NuScale Technical Report TR-1015-18177, "Pressure and Temperature Limits Methodology." NuScale TR-1015-18177, Rev. 0, identifies that TCOLD for the NuScale design falls outside of the applicability of Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials." The TR utilizes the methodologies of ASTM E900-15, "Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," and 10 CFR 50.61a, "Alternative Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," to calculate necessary embrittlement results.

The applicability of 10 CFR 50.61a is restricted as noted therein (10 CFR 50.61a.(b)),

> The requirements of this section apply to each holder of an operating license for a pressurized water nuclear power reactor whose construction permit was issued before February 3, 2010 and whose reactor vessel was designed and fabricated to the ASME Boiler and Pressure Vessel Code, 1998 Edition or earlier.

This is due to the fact that the analytical basis (e.g., thermohydraulic transient types and frequencies, etc.) underlying 10 CFR 50.61a would not apply to a design like the NuScale SMR. The NRC has only approved of the use of the embrittlement models in 10 CFR 50.61a within the context of the overall analysis methodology contained in that regulation. To use the embrittlement models from 10 CFR 50.61a in the context of 10 CFR Part 50, Appendix G and/or 10 CFR 50.61 would require further justification regarding how acceptable margins will be maintained given other aspects of those methodologies. Similarly the use of E900-15, "Standard Guide for Predicting Radiation-Induced Transition Temperature Shift in Reactor Vessel Materials," would require substantial analysis and justification to be used in concert with 10 CFR 50.61. Consequently neither of these methodologies are acceptable as presented to the staff for this application.

The staff requests that the applicant revise TR-1015-18177 with a staff approved methodology to account for embrittlement at the presented TCOLD. The staff has approved of methodologies which account for TCOLD temperatures outside of the range explicitly addressed in RG 1.99, Rev. 2. An example of an approved methodology can be found in a staff presentation accessible using ADAMS Accession No. ML110070570, and at:

https://www.nrc.gov/docs/ML1100/ML110070570.pdf

In the methodology noted above a degree-per-degree should added to the results from RG 1.99, Rev. 2, starting from 525°F.

NuScale Response:

The NuScale reactor vessel embrittlement methodology has been revised to account for irradiation temperature using the degree-per-degree adjustment as requested by the RAI. The NuScale design assumes an irradiation temperature of 497 degrees F. Regulatory Guide 1.99, Rev 2, recommends increasing the embrittlent effect if irradiation temperatures are less than 525 degrees F. Accordingly, the revised NuScale methodology adds 28 degrees (=525-497) to the adjusted reference temperature calculated by the methods of Regulatory Guide 1.99, Rev 2.

NuScale technical report TR-1015-18177 has been revised to reflect this change.

Impact on DCA:

Technical Report TR-1015-18177, Pressure and Temperature Limits Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

where

CF = chemistry factor based on Cu and Ni contents, and is given by Tables 1 and 2 of RG 1.99 Rev. 2 or 10 CFR 50.61.

FF = fluence factor calculated per the following equation:

$$FF = f^{(0.28-0.10*logf)} \quad \text{Eq. 4-5}$$

"f" = fluence in units of 1E+19 n/cm^2

The fluence for PTS screening is the clad-to-base-metal interface fluence (0-T fluence) per 10 CFR 50.61. The ¼-T and ¾-T fluences for the beltline region ART calculations are determined per Eq. 4-2.

4.2.3.2 Adjustment for Irradiation Temperature

The RPV irradiation temperature is 497 degrees Fahrenheit for NuScale. Because 497 degrees Fahrenheit is below the applicable temperature range of RG 1.99 Rev. 2, adjustment is required by the following statement in RG 1.99 Rev 2:

> *"The [RG 1.99 Rev. 2] procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement. The correction factor used should be justified by reference to actual data."*

Currently, there is no actual NuScale RPV data to justify a correction factor for ΔRT_{NDT}. The Nuclear Regulatory Commission's RAI No. 234 (Reference 11.11) requested that ΔRT_{NDT} be increased by one degree Fahrenheit for each degree Fahrenheit below 525 degrees Fahrenheit (1°F/1°F). Because NuScale T_{cold} is 28 degrees Fahrenheit below 525 degrees Fahrenheit, the adjustment can be expressed by the following equation for the NuScale RPV:

$$\Delta RT_{NDT} = \Delta RT_{NDT,RG1.99Rev\ 2} + 28°F \quad \text{Eq. 4-6}$$

ΔRT_{NDT} at depths of 0-T, ¼-T, and ¾-T are shown in the following tables.

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eRAI No.: 9118
Date of RAI Issue: 09/22/2017

NRC Question No.: 05.03.02-2

The applicant has proposed relocating certain information related to pressure and temperature limits for the reactor coolant pressure boundary to a PTLR which should meet the acceptance criteria of Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits." As stated in DCD Tier 2 Section 5.3.2.1, the applicant has described the specific methodology used to determine operational pressure limits on pressure and temperature for the reactor coolant pressure boundary in NuScale Technical Report TR-1015-18177, "Pressure and Temperature Limits Methodology." The first criteria in GL 96-03 is to describe how the neutron fluence methodology is calculated. The staff reviews neutron fluence methodologies against the criteria of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The applicant stated in a meeting on May 15, 2017 that their neutron fluence methodology, as described in TR-0116-20781, "Fluence Calculation Methodology and Results," is compliant with RG 1.190. However, the staff cannot locate statements confirming this statement in the DCD. Therefore, the staff requires further information to adequately confirm that the neutron fluence values used in the PTLR are acceptable and consistent with the guidance contained in RG 1.190.

During a meeting with the applicant on May 15, 2017, the staff was informed that the neutron fluence methodology used for the neutron fluence information presented in TR-1015-18177 is consistent with the methodology documented in TR-0116-20781. Please confirm that all neutron fluence related information presented in the NuScale TR-1015-18177 is generated consistent with the methodology presented in TR-0116-20781. If this is not the case, justify that the approach used to calculate the neutron fluence values used in TR-1015-18177 are acceptable per Regulatory Guide 1.190 or other acceptable basis.

NuScale Response:

TR-0116-20781, Revision 0, Fluence Calculation Methodology and Results, describes the NuScale neutron fluence methodology which is based on the guidance provided in RG 1.190 Revision 0. FSAR Section 5.3.1 and TR-1015-18177 have been revised to indicate that the neutron fluences were calculated consistent with the guidance of Regulatory Guide 1.190, with exceptions as described in TR-0116-20781, Revision 0.

Impact on DCA:

FSAR Section 5.3 and related Technical Report TR-1015-18177, Pressure and Temperature Limits Methodology, have been revised as described in the response above and as shown in the markup provided with this response.

time, the archive materials are maintained as full-thickness sections instead of being machined into specimens. Therefore, the archive materials for Reactor Pressure Vessel Surveillance Program are taken from the actual production forgings, and from weldments made from the same weld wire heat and flux lot combination used in the production weld.

Table 5.3-4 lists the specimen matrix for the Material Surveillance program requirements. As shown in the table, the number of specimens meets the ASTM E185-82 (Reference 5.3-6) minimum requirements.

The NuScale reactor vessel is designed for 60 years. Therefore, for the first 40 years of the 60-year design life, the capsule withdrawal schedule complies with Table 1 of ASTM E185-82, which is based on 32 effective full-power years (EFPY). Three capsules are sufficient to cover the initial 40-year operation per E185-82. ~~For the remaining 20-year design lifetime, a fourth capsule is included. This fourth capsule is consistent with the license renewal requirements of NUREG-1801, Revision 2, for the 20-year license renewal period after the initial 40-year license.~~ The capsule withdrawal schedule is provided in Table 5.3-5.

The capsules are inside capsule holders that are attached to the outside of the core barrel at mid-height of the core. The capsules are positioned to achieve a lead factor between 1.5 and 4.5. The four capsules are positioned approximately 90 degrees apart around the circumference of the core support assembly. Figure 5.3-2 shows the core barrel horizontal cross-section and the location of the four capsule holders and capsule elevation on the core barrel.

RAI 05.03.02-2

The neutron flux and fluence calculations are ~~contained~~consistent with the guidance of RG 1.190, and are described in NuScale Technical Report TR-0116-20781, "Fluence Calculation Methodology and Results" (Reference 5.3-7).

The transition temperature upper shelf energy changes are calculated in accordance with RG 1.99, and are shown in Table 5.3-8, Table 5.3-9, and Table 5.3-10. Section 5.3.2 provides further information.

COL Item 5.3-3: A COL applicant that references the NuScale Power Plant design certification will describe their reactor vessel material surveillance program consistent with NUREG 0800, Section 5.3.1.

5.3.1.7 Reactor Vessel Fasteners

The RPV closure studs, nuts, and washers use SB-637 Alloy 718, instead of low alloy steels such a SA-540 Grade B23 or B24. The selection of Alloy 718 over traditional low alloy steels is to prevent general corrosion when the bolting is submerged during the plant startup and shutdown process. Because of its resistance to general corrosion, the concerns addressed by RG 1.65, Revision 1, position 2(b) do not apply to Alloy 718. Alloy 718 is an austenitic, precipitation-hardened, nickel-based alloy permitted for bolting materials by ASME BPVC Section III (Reference 5.3-1), Subsection NB-2128.

The capsules are inside capsule holders that are attached to the outside of the core barrel at mid-height of the core. The capsules are positioned to achieve a lead factor of approximately 2.5. The four capsules are positioned approximately 90 degrees apart around the circumference of the core support assembly. Figure 5.3-2 shows the core barrel horizontal cross-section and the location of the four capsule holders and capsule elevation on the core barrel.

RAI 05.03.02-2

The neutron flux and fluence ~~calculations~~calculation methods are consistent with the guidance of RG 1.190 ~~and are~~with exceptions as described in NuScale Technical Report TR-0116-20781, "Fluence Calculation Methodology and Results" (Reference 5.3-7).

The transition temperature upper shelf energy changes are calculated in accordance with RG 1.99, and are shown in Table 5.3-8, Table 5.3-9, and Table 5.3-10. Section 5.3.2 provides further information.

COL Item 5.3-3: A COL applicant that references the NuScale Power Plant design certification will describe the reactor vessel material surveillance program consistent with NUREG 0800, Section 5.3.1.

5.3.1.7 Reactor Vessel Fasteners

The RPV closure studs, nuts, and washers use SB-637 Alloy 718, instead of low alloy steels such a SA-540 Grade B23 or B24. The selection of Alloy 718 over traditional low alloy steels is to prevent general corrosion when the bolting is submerged during the plant startup and shutdown process. Because of its resistance to general corrosion, the concerns addressed by RG 1.65, Revision 1, position 2(b) do not apply to Alloy 718. Alloy 718 is an austenitic, precipitation-hardened, nickel-based alloy permitted for bolting materials by ASME BPVC Section III (Reference 5.3-1), Subsection NB-2128.

Furthermore, because Alloy 718 is not a ferritic material, the fracture toughness requirements of NB-2333 are not required. Further information is provided in Section 3.13.

RAI 05.03.01-3

Threaded inserts are used for RPV threaded fasteners except for the main RPV flange studs and steam generator inlet flow restrictor hardware. See Table 5.2-4 for threaded insert materials and applicable specifications.

5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses

Analyses

The information provided in this section describes the bases for setting operational limits on pressure and temperature for the RCPB and ensures the requirements of 10 CFR 50, Appendices G and H, and 10 CFR 50.61 are complied with throughout the 60-year life of the plant.

4.0 Components of the Pressure-Temperature Calculations

4.1 Neutron Fluence

This section provides a summary of the NuScale neutron fluence methodology, which is consistent with the guidance provided in RG 1.190 (Reference 11.5) with exceptions as described in NuScale Power technical report TR-0116-20781-P, “Fluence Calculation Methodology and Results” (Reference 11.10).

A Monte Carlo method using Monte Carlo N-Particle Transport Code (MCNP) 6.1 (Reference 11.9) was chosen to develop the fluence profile. MCNP6.1 is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. The ENDF-B/VII.I pointwise (continuous energy) cross-section data from ENDF-B/VII.I are available with the MCNP6.1 package used in this analysis.

The neutron flux in the lower RPV, in units of n/cm^2-sec, is calculated by using MCNP cylindrical mesh tallies of the neutron flux with a defined mesh structure superimposed over the region of interest. Each data block provides the mesh central coordinates in three dimensions as well as tally results and its relative error.

The fast neutron (E > 1 MeV) fluence (n/cm^2) in the RPV is calculated using flux tallies with a 1 MeV energy cutoff. Neutron fluence is evaluated at several locations, including the surveillance capsules, by using MCNP mesh tallies.

The total fission neutron source intensity *S* in the NuScale module at a given power is determined by the following equation.

$$S = \frac{\upsilon * P * 10^6 \left[\frac{W}{MW}\right]}{1.602 \times 10^{-13} \left[\frac{J}{MeV}\right] * K_{eff} * Q_{ave}} \quad \text{Eq. 4-1}$$

where

ν: Average number of neutrons produced per fission,

P: Fission power defined (MW),

K_{eff}: Effective multiplication factor (= 1.000), and

Q_{ave}: The average recoverable energy per fission for all materials (MeV).

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9118
Date of RAI Issue: 09/22/2017

NRC Question No.: 05.03.02-3

In order to reach a safety conclusion regarding the PTLR methodology cited in DCD, Tier 2, Section 5.3.2 (i.e., NuScale TR-1015- 18177, "Pressure and Temperature Limits Methodology"), the staff requests that the applicant submit EC-A011-3215, "Pressure- Temperature Limits Calculation for RPV at 57-EFPY Fluence." Based on the results of a recent staff audit, this report provides a more detailed and thorough basis for the results reported in the PTLR and includes information necessary to complete the review of the PTLR.

NuScale Response:

Internal documents are not typically included in RAI responses. NuScale technical report TR-1015-18177, *Pressure and Temperature Limits Methodology*, has been revised (to Revision 1) to include more information from EC-A011-3215, *Pressure-Temperature Limits Calculation for RPV at 57-EFPY Fluence.*

Impact on DCA:

Technical Report TR-1015-18177, Pressure and Temperature Limits Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

NUSCALE POWER™

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9118
Date of RAI Issue: 09/22/2017

NRC Question No.: 05.03.02-4

In order to reach a safety conclusion regarding the PTLR methodology cited in DCD, Tier 2, Section 5.3.2 (i.e., NuScale TR-1015- 18177, “Pressure and Temperature Limits Methodology”), the staff requires the following information:

For all 10 values of KIm in Table 5-1 of the TR-1015-18177 (also in Table 4-3 of the report EC-A011-3215, “Pressure-Temperature Limits Calculation for RPV at 57-EFPY Fluence”), provide the KIm values extracted from the finite element modeling for the first 6 pathlines centered around the crack tip. This information is needed to verify the adequacy of the final values reported in Table 5-1 of the TR (i.e., that the pathline chosen for the final KIm value is adequate). Specifically, the NRC requests that the following table be filled out.

K_{Im} (psi.in0.5)						
Crack	**Pathline P1**	**Pathline P2**	**Pathline P3**	**Pathline P4**	**Pathline P5**	**Pathline P6**
Crack 1						
Crack 2						
Crack 3						
Crack 4						
Crack 5						
Crack 6						
Crack 7						
Crack 8						
Crack 9						
Crack 10						

NuScale Response:

The K_{Im} data for the first 6 contours are provided in Table 1 below. As stated in TR-1015-18177, Revision 1, Section 4.4.3, the maximum value from Contours 2 through 5 is the maximum SIF

(K_{Im}). Contour 1 is not used since it is not accurate due to numerical inaccuracies in the stresses and strains at the crack tip. Contour 6 is not used because it is too far from the crack tip. In Table 1 below, the maximum value from Contours 2 through 5 in each crack is provided in the last column.

Table 1: The K_{Im} for the first 6 contours (unit: psi.in$^{0.5}$)

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}}2(a)(c)

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9118
Date of RAI Issue: 09/22/2017

NRC Question No.: 05.03.02-5

NuScale TR-1015-18177, "Pressure and Temperature Limits Methodology," states that the stress intensity factor due to transient thermal loads, KIT, is calculated using Equation 4-14 of the TR. Equation 4-14 is only accurate if the stress field can be fitted by a 3rd order polynomial with sufficient goodness-of-fit. Further basis for these results is given in Appendix F of EC-A011-3215, "Pressure- Temperature Limits Calculation for RPV at 57-EFPY Fluence." In order to reach a safety conclusion regarding the PTLR methodology cited in DCD, Tier 2, Section 5.3.2 (i.e., NuScale TR-1015-18177, "Pressure and Temperature Limits Methodology") the staff requires the following to verify that the thermal stresses are adequately modeled by a 3rd order polynomial:

- For each combination of crack and transient presented in Appendix F of EC-A011-3215 (12 cases in total), provide the raw finite element analysis (FEA) stress profile data at the time when the maximum KIT is reached.
- For the purpose of comparing the raw data with the data fits, also provide in your response the C0, C1, C2, and C3 coefficients listed in table F.12 of EC-A011-3215 at the time when the maximum KIT is reached, and the R2 correlation coefficient for the 3rd order polynomial fit to the raw FEA data at this time step.

NuScale Response:

Subquestion 1:

The maximum K_{IT} values from all 12 cases in Appendix F are summarized in Table 1 below, which also includes the corresponding step number and time.

Table 1: Maximum K_{IT} values in Appendix F

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}}2(a)(c)

On each pathline, 81 data points are extracted from inner diameter (ID) to outer diameter(OD) along the pathline. Data points 1 through 21 are from ID surface to ¼ T, while data points 61 through 81 are from ¾ T to OD surface. The stress data for the step and pathline numbers identified in Table 1 are summarized in Tables 2 and 3 below. Table 2 lists the data for Sections F.1 through F.6, where cracks are at ID. Therefore, the data points listed are from 1 to 21. Axial stresses are listed for circumferential cracks while hoop stresses are listed for axial cracks. Table 3 lists the data for Sections F.7 through F.12, where cracks are at OD. Thus, the data points are from 61 to 81.

Table 2: Thermal stress at maximum K_{IT} step (cracks at Sections F.1 through F.6)

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}}2(a)(c)

Table 3: Thermal stress at maximum K_{IT} step (cracks at Sections F.7 through F.12)

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}}2(a)(c)

Subquestion 2:

As listed in Table 1, the maximum K_{IT} is reached in Section F.12 after 26200 seconds. The C_0, C_1, C_2, and C_3 coefficients at that time are summarized in Table 4 (from Section F.12). The curve fit using Excel trendline (third order polynomial) and the curve calculated using Table 4 coefficients are compared with raw FEA data in Figure 1. The equation of the Excel trendline is also illustrated in Figure 1 (shown in green color). The R^2 correlation using the coefficients is listed in Table 4, which is calculated using Excel function RSQ. A near 1.0 value of R^2 correlation coefficient and an almost perfect curve fitting in Figure 1 demonstrate that the coefficients used are adequate.

Table 4: Section F.12 step 19 coefficients

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}}2(a)(c)

Figure 1: Section F.12 time 26200 second ¾ T hoop stress

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Enclosure 3:

Affidavit of Thomas A. Bergman, AF-0418-59446

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 method by which NuScale develops its pressure and temperature limits for the reactor vessel.

    NuScale has performed significant research and evaluation to develop a basis for this 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 enclosed response to NRC Request for Additional Information RAI No. 234, eRAI 9118. 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 4/6/2018.

Thomas A. Bergman