RAIO-0119-64377

January 31, 2019

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 Supplemental Response to NRC Request for Additional Information No. 9306 (eRAI No. 9306) on the NuScale Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 0

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9306 (eRAI No. 9306)," dated April 04, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 9306 (eRAI No.9306)," dated June 04, 2018
3. NuScale Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 0, dated December 2016

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

The Enclosures to this letter contain NuScale's supplemental response to the following RAI Questions from NRC eRAI No. 9306:

- 15.04.08-5
- 15.04.08-6
- 15.04.08-15
- 15.04.08-16

Enclosure 1 is the proprietary version of the NuScale Supplemental Response to NRC RAI No. 9306 (eRAI No. 9306). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The proprietary enclosures have been deemed to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810. 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,

Carrie Fosaaen
Supervisor, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9306, proprietary

Enclosure 2: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9306, nonproprietary

Enclosure 3: Affidavit of Thomas A. Bergman, AF-0119-64378

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9306, proprietary

Enclosure 2:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9306, nonproprietary

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

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 15.04.08-5

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. The applicant must use computer codes to demonstrate the compliance with appropriate limits and utilize models that capture the phenomena associated with the event being analyzed.

Section 3.2.1.3 of TR-0716-50350 states that SIMULATE-3K is used to determine the power response for the accident, which is subsequently used in NRELAP5 and VIPRE-01. The power response is dependent on the timing of the reactor trip and is critical in the analysis of the REA in limiting clad damage. For the most limiting cases a reactor trip is expected from high flux rate or high neutron flux signal. TR-0716-50350-P does not describe how SIMULATE-3K modeled the excore detectors.

Describe how the excore detectors are modeled in the SIMULATE-3K analysis

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NuScale Response:

<u>NuScale Supplement Response</u>

The original NuScale response as submitted in NuScale correspondence RAIO-0618-60285 and dated June 4, 2018, is augmented with the following information.

As described in Section 4.3, item D of the Rod Ejection Accident Methodology topical report (TR-0716-50350), the reactor trip has a negligible effect on the limiting cases because the limiting cases are those that experience prompt, or near prompt, criticality due to the reactivity insertion. This example behavior is generic to all power transient initial conditions screened by NRELAP5 as being possibly limiting (see Section 6.2 of the topical report for more detail), as may be observed in the following Table 1, Figure 1, and Figure 2. As an example, the time of minimum critical heat flux ratio (MCHFR) for cases ‘EOC 50’ and ‘EOC 70’ are effectively the same. The peak power for ‘EOC 50’ is slightly higher, but occurs slightly slower. There is no case in which a reactor trip mitigates the consequences of the transient. Table 1 shows that for all cases, peak power and MCHFR occurred well before the control rods would have started to move, 2 seconds after a trip signal, should a trip signal have occurred.

Table 1. Summary of Example Cases Screened by NRELAP5

Case Name	Cycle Exposure (GWd/MT)	Initial Power (% Rated)	Peak Power (% Rated)	Time Peak Power (sec)	Time of MCHFR (sec)	MCHFR
4GW 50	4	50	185.5	0.0823	{{	
4GW 70	4	70	240.2	0.0780		
BOC 50	0	50	133.0	0.0930		
BOC 70	0	70	177.5	0.0701		
EOC 45	12.1	45	642.4	0.0928		
EOC 50	12.1	50	648.5	0.0917		
EOC 55	12.1	55	660.5	0.0890		
EOC 60	12.1	60	649.2	0.0856		
EOC 70	12.1	70	614.5	0.0837		
EOC 80	12.1	80	261.7	0.0762		}}2(a),(c),ECI

Figure 1 and Figure 2 illustrate the transient progression of power and MCHFR, respectively, for the cases listed in Table 1. The limiting values for both of these parameters occur very early in the transient.

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}}2(a),(c),ECI

Figure 1. Comparison of Input Core Power Forcing Functions

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}}2(a),(c),ECI

Figure 2. Comparison of Minimum CHF Ratio

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.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 15.04.08-6

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. In performing the analysis of the REA the applicant must select inputs to assure a bounding calculation that would envelop plant operation and possible future cycle designs and reflect limits in Technical Specifications or COLR.

Sections 4.1.1.1, 4.1.1.2, and 5.2.1.1 of TR-0716-50350-P discuss the application of uncertainty factors applied to SIMULATE-3K for the rod ejection analysis. For intrinsically (code determined) parameters in Table 5-1 (DTC, B_{eff}, ejected CRA worth, MTC) it is unclear to the staff how the multipliers are applied to SIMULATE-3K.

Describe in detail how these uncertainty multipliers for intrinsically determined parameters are applied to SIMULATE-3K.

NuScale Response:

The original NuScale response as submitted in NuScale correspondence RAIO-0618-60285 and dated June 4, 2018, is augmented with the following information.

As described in Section 5.2.1.2, conservatism is applied to key nuclear parameters in SIMULATE-3K to produce a conservative transient response from the code. The conservatisms

are also referred to as nuclear reliability factors (NRFs). Conservatism is applied to the effective delayed neutron fraction (β_{eff}), fuel temperature coefficient (FTC), moderator temperature coefficient (MTC), and control rod assembly (CRA) worth via the 'KIN.MUL' card in SIMULATE-3K.

For β_{eff}, the conservatism is applied as a {{

}}2(a),(c) The delayed neutron data is supplied by the cross-section (neutron data) library created by CASMO5 and input into the code.

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}}2(a),(c)

For the FTC, the Doppler feedback can be estimated as the product of the FTC and the change in fuel temperature with respect to the steady-state condition:

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}}2(a),(c)

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}}2(a),(c) and is applied in SIMULATE-3K to account for conservatism in the Doppler feedback. Since the NRF for FTC is a relative value, the multiplier is directly applied and no iterations are necessary.

For MTC, the SIMULATE-3K methodology is similar to FTC, but the NuScale NRF is an absolute value, so it is not directly applied as the multiplier. The multiplier must be iterated upon to determine a relative value corresponding to an adjusted MTC accounting for the application of the NRF in a conservative manner.

For CRA worth, the {{ }}2(a),(c) the multiplier must be iterated upon to determine a value corresponding to an adjusted rod worth accounting for the application of the NRF in a conservative manner.

Impact on DCA:

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

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9306

Date of RAI Issue: 04/04/2018

NRC Question No.: 15.04.08-15

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. For an applicant to correctly predict fuel failures resulting from overheating of the fuel cladding in support of demonstrating compliance with GDC 28, the fuel melting analysis methodology must be shown to conservatively calculate the fuel centerline temperature.

Table 5-1, “Uncertainties for REA calculations,” of TR-0716-50350 provides the uncertainties applied to the rod ejection analysis. It is unclear to the staff if the uncertainties in Table 5-1 will be updated as described in Section 7.0 of the “Nuclear Analysis Codes and Methods Qualification” topical report (TR-0616-48793, Rev. 0). The staff also notes that the $F_{\Delta H}$ provided in Table 5-1 is less conservative than the FΔH given in Section 7.7.1, “Base Nuclear Reliability Factors,” of TR-0616-48793.

a. Please indicate if the uncertainties in Table 5-1 will be updated consistent with TR-0616-48793. If the uncertainties will not be updated as discussed in TR-0616-48793, either describe the method for updating them or provide a justification as to why an update is not necessary. If the uncertainties in Table 5-1 will be updated, modify TR-0716-50350 to indicate the method by which updates will be made.

b. Justify the use of a lower $F_{\Delta H}$ uncertainty for the rod ejection analysis relative to the steady-state $F_{\Delta H}$ uncertainty.

NuScale Response:

The original NuScale response as submitted in NuScale correspondence RAIO-0618-60285 and dated June 4, 2018, is augmented with the following information.

The rod ejection methodology is a cycle-specific approach to evaluate rod ejections for each core reload. As discussed in Section 5.4.2.1 of the topical report, the radial power distribution used in the minimum critical heat flux ratio (MCHFR) evaluation is a conservative artificial distribution contrived from the peaking results in the SIMULATE-3K analysis. In addition to the mentioned $F_{\Delta H}$ engineering uncertainty of {{ }}$^{2(a),(c)}$ applied to the peak rod, the uncertainty for the pin peaking nuclear reliability factor (NRF) of {{ }}$^{2(a),(c)}$ was incorporated. This additional pin peaking NRF is consistent with the steady-state uncertainty discussed in the Nuclear Analysis Codes and Methods Qualification Report (TR-0716-48793). Text was added to indicate the incorporation of the pin peaking NRF into the NuScale rod ejection accident methodology (Section 5.4.2.1 and Table 5-1 of TR-0716-50350) as indicated at the end of this response.

Impact on Topical Report:

Topical Report TR-0716-50350, Rod Ejection Accident Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

Table 5-1 Example ~~U~~uncertainties for rod ejection accident calculations

Parameter	Uncertainty	Analysis
Delayed neutron fraction	6 percent	SIMULATE-3K
Ejected CRA worth	12 percent	SIMULATE-3K
Doppler temperature coefficient	15 percent	SIMULATE-3K
MTC	2.5 pcm/°F	SIMULATE-3K
CRA position	6 steps	SIMULATE-3K
Initial power	2 percent	NRELAP5
F_Q	{{	Adiabatic Heatup
$F_{\Delta H}$ engineering uncertainty		VIPRE-01
$F_{\Delta H}$ pin peaking nuclear reliability factor	}}2(a),(c)	VIPRE-01

5.2.3 Results and Downstream Applicability

No explicit acceptance criteria are evaluated in the core response calculations. Instead, the boundary conditions are generated to be used by the system response, subchannel, and fuel response analyses. Applicable acceptance criteria are applied to these downstream analyses.

5.3 System Response

The generic non-LOCA methodology is discussed in more detail in the non-LOCA evaluation methodology topical report (Reference 8.2.10); for the system analysis using NRELAP5, REA utilitizes this methodology. However, in order to assess the NuScale criteria outlined in Section 2.3, some deviations or additions to the non-LOCA methodology are implemented. The event-specific analysis is discussed in this section.

5.3.1 Calculation Procedure

For the system response, calculations are performed for the purpose of determining the peak RCS pressure analysis and to provide inputs to the subchannel analysis for CHF determination. Because it is determined that pressurization, and not depressurization, is limiting for CHF, all NRELAP5 system calculations are performed assuming no depressurization effects.

Critical heat flux scoping cases are performed to determine the general trend and to select the cases to be evaluated in the VIPRE-01 subchannel analysis for final confirmation that no MCHFR fuel failures occur.

Competing scenario evaluations exist between the peak pressure and the MCHFR calculations. The two scenarios to consider within the system response are as follows:

- The SIMULATE-3K power response is used to maximize the impact on MCHFR. This tends to be a rapid, peaked power response due to using the maximum possible ejected CRA worth based on insertion to the PDIL.

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}}2(a),(c)

5.4.2 Analysis Assumptions and Parameter Treatment for Subchannel Response

5.4.2.1 Radial Power Distribution

The radial power distribution to be used for the subchannel REA evaluations is a case-specific conservative artificial distribution based on the highest peaked $F_{\Delta H}$ rod at the time of peak neutron power as predicted in the SIMULATE-3K analysis. This condition will occur after the ejected CRA is fully out of the core. In addition, the $F_{\Delta H}$ engineering uncertainty ~~is~~and the pin peaking nuclear reliability factor are applied to the highest peaked $F_{\Delta H}$ rod. The uncertaintie~~y~~s associated with $F_{\Delta H}$ are~~is~~ given in Table 5-1 and are combined using the root-sum-squared method similar to that discussed in Section 3.10.7 of Reference 8.2.11. The radial power distribution slope described in Section 3.10.6 of Reference 8.2.11 is used to determine the REA-specific normalized radial power distribution for use in VIPRE-01. In summary, the process for each case is to (i) determine the peak $F_{\Delta H}$ rod (ii) apply uncertainty to that rod only (iii) calculate a normalized power shape for both fully-detailed rods and lumped rods (iv) utilize artificial shape in VIPRE-01 simulation of the case.

The conservative nature of this modeling is described in Section 6.4.2.5. Additionally, as described in Section 6.4.2 of Reference 8.2.11, the radial power distribution more than a few rows removed from the hot subchannel has a negligible impact on the MCHFR results. Analysis of different power distributions of the NuScale core demonstrate that rod powers a few rod rows beyond the hot rod or channel have a negligible impact on the MCHFR.

5.4.2.2 Axial Power Distribution

The axial power distribution to be used will be a normalized representation of the SIMULATE-3K assembly-average axial power at time of maximum core neutron power for the assembly containing the highest peak $F_{\Delta H}$ rod.

5.4.2.3 Core Inlet Flow Distribution

The inlet flow distribution for subchannel analyses is described in Reference 8.2.11. For REA calculations, the limiting inlet flow fraction is applied to the assembly containing the rod with the highest $F_{\Delta H}$ as described above.

5.4.2.4 Fuel Conductivity and Gap Conductance

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

eRAI No.: 9306

Date of RAI Issue: 04/04/2018

NRC Question No.: 15.04.08-16

In accordance with 10 CFR 50 Appendix A GDC 28, "Reactivity Limits," the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 provides review guidance related to the spectrum of REAs.

Section 5.5.1 of TR-0716-50350-P states that the change in fuel centerline temperature determined by Equation 5-2 is added to the initial fuel centerline temperature as the bounding starting temperature. Likewise, the change in enthalpy, as calculated by Equation 5-4, is dependent on the maximum pre-transient fuel centerline temperature as described by Equation 5-3. Section 3.2.1.3 of TR-0716-50350-P, "SIMULATE-3K," states that within-pin fuel temperature distribution is governed by the one-dimensional radial heat conduction equation. Section 3.2.1.3 of TR-0716-50350-P goes on to state that material properties are temperature and burnup dependent, and gap conductance is dependent on exposure and fuel temperature. This method assumes the transient, within pellet radial temperature distribution remains constant (i.e., initial steady-state, within pellet radial shape is preserved). In a rod ejection transient, within pellet radial power distributions may not remain constant (e.g., radial power profile may become more edge peaked).

Demonstrate that the proposed method produces a conservative, maximum fuel pellet temperature. As part of this demonstration describe how SIMULATE-3K is used to determine the initial within pellet radial temperature distribution and provide comparisons, including the effects of burnup-dependent thermal conductivity degradation, to either experimental data or an NRC approved fuel performance code to show a reasonably conservative initial (steady-state) temperature distribution.

NuScale Response:

<u>NuScale Supplement Response</u>

The original NuScale response as submitted in NuScale correspondence RAIO-0618-60285 and dated June 4, 2018, is augmented with the following information.

For the rod ejection accident, the fuel is modeled in two different manners for the two different sets of fuel failure acceptance criteria, referred to as (1) critical heat flux (CHF) and (2) non-CHF related for the purposes of this response. The SIMULATE-3K (S3K) calculation is not directly relied upon to perform initial or maximum fuel pellet temperature calculations, rather it calculates the power pulse and power peaking for use in the downstream analysis.

- **CHF:** The critical heat flux ratio is calculated in VIPRE-01 using the power pulse and power peaking as input. As described in Section 4.4 of the Subchannel Analysis Methodology topical report (TR-0915-17564), for each fuel design the VIPRE-01 fuel conduction model is updated based on a fuel performance code benchmark in order to ensure the MCHFR calculation conservatively accounts for the entire range of possible time-in-life parameters, including exposure, uranium enrichment, gadolinium enrichment, gap conductance, and fuel density.

- **Non-CHF**: The adiabatic calculation described in Section 5.5 of the topical report, with conservative modeling and assumptions, utilizes the NRC-approved fuel performance code COPERNIC for the initial fuel temperature calculation. From this and other inputs, the various parameters are calculated and compared to the acceptance criteria. The initial fuel temperature is ensured to be bounding for a given fuel-design by conducting a fuel design-specific evaluation, similar to that performed for the subchannel analysis described in Section 4.4 of TR-0915-17564. Specifically, this methodology requires that the entire range of possible time-in-cycle parameters (i.e., exposure, uranium enrichment, gadolinium enrichment, gap conductance, and fuel density) are evaluated using the COPERNIC fuel performance code.

The S3K code is not directly relied upon to perform initial or maximum fuel pellet temperature calculations. S3K uses the fuel average temperature as the main feedback mechanism (92%) to calculate the Doppler feedback. S3K uses pre-calculated radial profiles that vary as a function of exposure and does not explicitly model the pellet rim. This use of S3K, in conjunction with the uncertainty treatment described in Section 5.2 of the topical report assures conservative fuel

performance modeling, and is appropriate for calculating the power pulse and power peaking for use in the downstream analysis for rod ejection accidents.

Impact on DCA:

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

Enclosure 3:

Affidavit of Thomas A. Bergman, AF-0119-64378

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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 rod ejection analysis for the NuScale Power Module.

   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 No. 9306, eRAI No. 9306. 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 January 31, 2019.

Thomas A. Bergman