RAIO-1117-57040

November 09, 2017

Docket: PROJ0769

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 9093 (eRAI No. 9093) on the NuScale Topical Report, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," TR-0516-49417, Revision 0

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9093 (eRAI No. 9093)," dated September 10, 2017
2. NuScale Topical Report, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," TR-0516-49417, Revision 0, dated July 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 Questions from NRC eRAI No. 9093:

- 01-36
- 01-37
- 01-38
- 01-39
- 01-40

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 9093 (eRAI No. 9093). 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 Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Jennie Wike
Manager, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9093, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9093, nonproprietary

Enclosure 3: Affidavit of Thomas A. Bergman, AF-1117-57041

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9093, proprietary

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9093, nonproprietary

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9093
Date of RAI Issue: 09/10/2017

NRC Question No.: 01-36

Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix A, General Design Criterion (GDC) 10 – Reactor Design, states that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). GDC 12-Suppression of reactor power oscillations, states that the reactor core and associated coolant, control, and protection system shall be designed to assure that power oscillation which can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed. The Design-Specific Review Standard (DSRS), 15.9.A, "Design-Specific Review Standard for NuScale SMR Design, Thermal Hydraulic Stability Review Responsibilities," indicates that the applicant's analyses should correctly and accurately identify all factors that could potentially cause instabilities and their consequences. The Standard Review Plan (SRP) 15.0.2 acceptance criteria with respect to accident scenario identification states that the process must include evaluation of physical phenomena to identify those that are important in determining the figure of merit for the scenario.

In section 4.4, "Phenomena Identification and Ranking Table," of the topical report, TR-0516-49417-P, natural circulation is ranked as a medium important phenomenon. However, natural circulation is a key phenomenon that affects the primary instability mode for the NuScale power module and the natural circulation flow rate is expected to be highly sensitive to pressure drop through the primary flow circuit.

In order to make an affirmative finding NRC staff requests NuScale to justify ranking pressure drop as being less than highly important in the PIRT or revise the PIRT to rank pressure drop as a highly important phenomenon.

NuScale Response:

Steady state primary flow dependence on pressure drop is direct and therefore important. The dependence of the stability of the flow on pressure drop (loss) coefficient is a different phenomenon where the important parameter is the relative response of the buoyancy and

friction forces to a relative perturbation in flow. While the reference flow is the steady state value which depends on the pressure loss coefficient, the same reference value applies to calculating both the buoyancy and friction components. In the TR, the lower ranking of the pressure drop is justified relative to the historical experience with BWR stability. For BWR stability, the location of the pressure drop component is highly important such that an increase in the pressure drop at the exit of a boiling channel, even when equally compensated by a decrease in the pressure drop at the inlet, is demonstrably destabilizing. By contrast, for the NuScale Module, the location (distribution) of pressure drop is not important due to the near-constant fluid density (Boussinesq approximation). Note that Boussinesq approximation is not explicitly applied in PIM where the code is capable of accounting for density changes and consequently the phase of flow perturbation changes along the primary flow path.

It should be emphasized that the medium ranking of any phenomenon, e.g. pressure drop, is a reflection of the qualitative understanding of its impact, however, its modeling is implemented with the same care afforded a highly ranked phenomenon. Therefore, a change in the tabulated ranking would not result in a model update to PIM.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0516-49417, Evaluation Methodology for Stability Analysis of the NuScale Power Module, as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9093
Date of RAI Issue: 09/10/2017

NRC Question No.: 01-37

Title 10, the code of federal regulations (CFR), Part 50, Appendix A, General Design Criterion (GDC) 12- Suppression of reactor power oscillations, requires that oscillations be either not possible or reliably detected and suppressed. The Design-Specific Review Standard (DSRS), 15.9.A, "Design-Specific Review Standard for NuScale SMR Design, Thermal Hydraulic Stability Review Responsibilities," indicates that the applicant's analyses should correctly and accurately identify all factors that could potentially cause instabilities and their consequences. The analyses should also demonstrate that design features that are implemented prevent unacceptable consequences to the fuel. The Standard Review Plan (SRP) 15.0.2 acceptance criteria with respect to accident scenario identification states that the process must include evaluation of physical phenomena to identify those that are important in determining the figure of merit for the scenario.

The disposition of flashing phenomenon in section 4.4, " Phenomena Identification and Ranking Table," of the topical report (TR), TR-0516-49417-P does not appear to be consistent with transient analyses in subsequent sections of the TR. The transient analyses documented in subsequent sections of the TR, essentially, demonstrate that the long term stability (LTS) solution is effective in tripping the reactor before riser flashing occurs. This approach to maintaining LTS relies on accurately determining the timing of flashing onset relative to the timing of MPS trip. While an LTS solution is proposed to preclude flashing, PIM must be capable of reliably predicting flashing in order to demonstrate that there is margin between the timing of the module protection system (MPS) trip and the onset of instability. The protection method relies on predicting the flashing phenomenon which is also important to the stability of the power module.

In order to make an affirmative finding NRC staff requests NuScale to provide validation of the riser flashing model.

NuScale Response:

The subject of riser energetics including flashing in the riser was addressed in responses to RAIs 8921, 9018, and 9019. Under the assumption of adiabatic riser, flashing of limited

magnitude was calculated by PIM in a demonstration of a depressurization transient (RAI 8921). The steam quality produced by flashing at the riser exit is estimated as 0.0024 when flow entering the riser inlet is assumed at saturation condition (RAI 9018).

While all the PIM calculations presented in the TR apply the conservative assumption of adiabatic riser, this assumption has been later examined in response to the need to address RAIs. A temperature reduction of 1.2 ºF (0.67 ºC) has been estimated to result from heat exchange across the riser wall (RAI 9019).

The energy available to vapor generation (flashing) is contrasted with the energy loss across the riser wall. The liquid enthalpy change due to flashing is estimated from

$$\Delta h_{flash} = x_{flash}\, h_{fg}$$

The liquid enthalpy reduction due to heat transfer (leakage) across the riser wall is obtained from

$$\Delta h_{leakage} = c_p\, \Delta T_{leakage}$$

where

x_{flash} = 0.0024 is the flashing steam quality

h_{fg} = 1144 kJ/kg is the evaporation enthalpy (latent heat) at pressure of 128 bar

c_p = 7.17 kJ/kg-ºC is the saturated liquid specific heat

$\Delta T_{leakage}$ = 0.67 ºC is the temperature drop from inlet to exit due to heat leakage through riser wall

The ratio of the enthalpy change due to flashing relative to the enthalpy change due to heat leakage is obtained from

$$\frac{\Delta h_{flash}}{\Delta h_{leakage}} = \frac{x_{flash}\, h_{fg}}{c_p\, \Delta T_{leakage}} = \frac{0.0024 \times 1144}{7.17 \times 0.67} = 0.57 < 1$$

It can be therefore concluded that the energy leakage through the riser wall is sufficient to prevent flashing even if the liquid coolant enters the bottom of the riser at saturated condition.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0516-49417, Evaluation Methodology for Stability Analysis of the NuScale Power Module, as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9093
Date of RAI Issue: 09/10/2017

NRC Question No.: 01-38

Title 10, the code of federal regulations (CFR), Part 50, Appendix A, General Design Criterion (GDC) 12- Suppression of reactor power oscillations, requires that oscillations be either not possible or reliably detected and suppressed. The Design-Specific Review Standard (DSRS), 15.9.A, “Design-Specific Review Standard for NuScale SMR Design, Thermal Hydraulic Stability Review Responsibilities,“ indicates that the applicant’s analyses should correctly and accurately identify all factors that could potentially cause instabilities and their consequences. The analyses should also demonstrate that design features that are implemented prevent unacceptable consequences to the fuel. The Standard Review Plan (SRP) 15.0.2 acceptance criteria with respect to accident scenario identification states that the process must include evaluation of physical phenomena to identify those that are important in determining the figure of merit for the scenario.

In section 4.4 “ Phenomena Identification and Ranking Table (PIRT),” of the topical report (TR), TR-0516-49417-P, does not appear to include subcooled boiling in Table 4.1, the PIRT table. However, the table does include the "vapor generation and condensation" as a PIRT table entry.

In order to make an affirmative finding NRC staff requests NuScale to clarify whether the phenomenon of subcooled boiling is intended to be captured under the broader category of vapor generation and condensation in the PIRT.

- If subcooled boiling is not part of the broad category, provide a ranking for this phenomenon and justify the ranking.
- If subcooled boiling is part of the broad category, justify the current ranking considering that the formation of void is expected to play a significant role in the primary instability mode.

NuScale Response:

The phenomenon of subcooled boiling is included in the broad category of vapor generation and condensation, which is ranked high in Table 4-1 of the TR.

The models in PIM capture the effect of subcooled boiling in the core. The models also capture the effect of the voids originated in subcooled boiling in the core as they are transported and condensed in the riser. Additional information regarding the stability effects of subcooled boiling were provided in the response to RAI 9017.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0516-49417, Evaluation Methodology for Stability Analysis of the NuScale Power Module, as a result of this response.

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

eRAI No.: 9093
Date of RAI Issue: 09/10/2017

NRC Question No.: 01-39

Title 10, the code of federal regulations (CFR), Part 50, Appendix A, General Design Criterion (GDC) 12- Suppression of reactor power oscillations, requires that oscillations be either not possible or reliably detected and suppressed. The Design-Specific Review Standard (DSRS), 15.9.A, “Design-Specific Review Standard for NuScale SMR Design, Thermal Hydraulic Stability Review Responsibilities,“ indicates that the applicant’s analyses should correctly and accurately identify all factors that could potentially cause instabilities and their consequences. The analyses should also demonstrate that design features that are implemented prevent unacceptable consequences to the fuel. The Standard Review Plan (SRP) 15.0.2 acceptance criteria with respect to accident scenario identification states that the process must include evaluation of physical phenomena to identify those that are important in determining the figure of merit for the scenario.

In section 4.4 “ Phenomena Identification and Ranking Table (PIRT),” of the topical report (TR), TR-0516-49417-P, the TR states, under the Table 4.1 "Instability in SG tubes" entry, that "Density waves in the SG tubes are excluded by design via the proper inlet throttling of individual tubes as verified by the experimental data."

In order to make an affirmative finding NRC staff requests NuScale to describe the process for demonstrating that the SG tubes are designed with sufficiently tight inlet orifices to preclude density wave instability.

NuScale Response:

Density wave oscillations (DWO) in the steam generator (SG) tubes are not excluded, rather the possibility of their occurrence and consequences thereof are accounted for in Appendix A of the topical report (TR), TR-0516-49417-P, "Evaluation Methodology for Stability Analysis of the NuScale Power Module." However, any destabilizing influence of the SG tube oscillations on the primary side stability is excluded. This clarification is included as an update to TR Table 4-1 "Phenomena Identification and Ranking," in the row entitled, “Instability in SG tubes.”

NuScale has performed extensive tests partly to quantify secondary DWO and to improve the

NRELAP5 stability modeling predictions. These flow fluctuations, if not controlled in magnitude, would impact the mechanical performance of the SG tubes due to thermal stresses, and degrade thermal performance by allowing moisture carryover at the SG exit. These performance issues are maintained within acceptable limits by throttling the inlets of the SG tubes. The maximum acceptable level of secondary flow oscillation magnitude (OM) is limited to 10% about the mean value as determined by the mass flow rate at the SG tube inlet.

NuScale DCA Tier 2 Steam Generator Design Basis Section 5.4.1.3 has been revised to clarify the selection of the current inlet loss coefficient design value as pertaining to these performance issues and not the stability of the primary flow or core power oscillations.

Impact on DCA:

Topical Report Sections 4.1 and 5.2 including Table 4-1 and FSAR Section 5.4.1.3 have been been revised as described in the response above and as shown in the markup provided with this response.

4.0 Phenomenological Description of NuScale Power Module Stability

4.1 Introduction

As described in Section 3.0, the NPM is an integral PWR. The SG is integrated within the RPV and the primary coolant flow is driven by natural circulation, which is an important aspect of its passive design philosophy. The density difference between the relatively high temperature flow exiting the core and the lower temperature flow returning through the downcomer annulus where the SG is the heat sink creates the natural circulation driving head. This configuration presents ~~{{~~ several flow circuits where thermal-hydraulic instabilities are demonstrably excluded during the design stage with regard to causing reactor power oscillations. This section describes these flow circuits and the associated feedback and delay mechanisms to cover the phenomenological aspect of the stability behavior, and to put in perspective the subsequent mathematical and numerical studies that demonstrate NPM stability.

The first flow circuit is the main circulation loop of the core coolant flow, which is subcooled as required for PWR operation. However, in the absence of a recirculation pump, the natural circulation head is dependent on the power level and flow rate, which is a feedback mechanism that may potentially lead to unstable flow oscillations.

The second possible flow path for a potential instability is the closed path between two fuel assemblies or regions in the core also known as the parallel channel mode. In this mode, density waves in one region of the core oscillate out of phase with the flow in another region. This condition would maintain the core pressure drop boundary condition and the power and flow in each fuel assembly may oscillate if the necessary conditions for density wave instability exist.

The third possible flow instability is in the secondary side of the SG where subcooled liquid water is pumped into the helical tubes, boiling occurs, and superheated steam exits at the other end. Density waves, which are common in parallel boiling channels, have been identified as a potential instability mode within the SG tubes and studied experimentally. ~~}}2(a),(c),ECI~~

Various feedback mechanisms are included, and special consideration is given to the possible coupling of the SG dynamics and the flow stability in the primary loop. Feedback coupling between the thermal-hydraulic phenomena and the neutron kinetics is important where coolant and fuel rod temperatures provide reactivity feedback, and the core power response affects the coolant temperature and the density head that drives the flow and influences its stability. Pure neutronic stability, without thermal-hydraulic feedback coupling, is addressed separately in dedicated neutronic analyses in the design certification.

4.2 Background and Past Reactor Stability Studies

Open literature contains extensive studies of the stability of nuclear systems, which is only a subset of the larger body of work when industrial activities in this area by reactor and fuel vendors are included. The primary focus of historical stability work has been for

{{

}}2(a),(c)

5.0 Theory and Model Description of the PIM Code

5.1 Background

The objective of the PIM code is to simulate the dynamics of the flow in the NPM coolant loop with special attention to optimal resolution of its stability. The extensive experience in the field of BWR stability analysis, both numerical and first principle understanding, has been utilized in addressing the new problem of single-phase natural circulation stability that is unique to the NPM. The guiding principle in designing the PIM code is maintaining simplicity, which is essential to the fidelity of stability analysis, while avoiding over simplifications that would sacrifice the level of details needed to ensure the applicability of the model to the actual reactor design and the important phenomena that were identified prior to the stability work.

Based on industry experience, including work in national laboratories and universities in the United States and abroad, it was found that a successful algorithm for thermal-hydraulic stability is that of the RAMONA series of codes (References 12.1.19 and

density head. {{

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

3. Power generation in the core is represented by a point kinetics model. Accordingly, the axial power shape is invariant, which is a reasonable approximation given that only minimal subcooled voiding is possible.
4. {{

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

5. The flow in the primary coolant loop is modeled as non-equilibrium two-phase flow in which a drift flux formulation accounts for mechanical (velocity) differences between the liquid phase and the vapor phase if vapor exists. Thermal non-equilibrium allows the liquid to be in a subcooled, saturated, or superheated state, but the vapor is restricted to the saturation state. Closing relations governing mass, momentum, and energy exchange between the phases and the solid structures are adaptations from commonly used correlations. The algorithms do not account for the possibility of reverse flow.
6. The flow in the secondary side of the SG is modeled {{

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

7. The pressurizer is not modeled. Pressure is specified by input and the dependence of thermodynamic properties on pressure is uniform. This approximation implies that pressure waves cannot be simulated where the sound speed is infinite. Given the long transport times for fluid transit around the primary coolant loop and the low frequency of the oscillations following any perturbation of the steady state, the impact

5.4.1.3 Performance Evaluation

A single RCS natural circulation flow loop is entirely contained within the RPV, thereby eliminating distinct RCS piping loops and the associated potential for a large pipe break (i.e., large break loss-of-coolant accident [LOCA]) event. This design, combined with the intertwined SGs tube bundle configuration, eliminates the potential for asymmetric core cooling and temperatures as a result of a loss of a single SG function. Isolation or other loss-of-heat transfer capability by either of the two intertwined SGs does not introduce asymmetrical cooling in the reactor coolant vessel or system because the tube configuration of the remaining functional SG continues to provide symmetrical heat removal from the reactor coolant flowing in the downcomer of the reactor vessel.

The primary coolant system operates at a higher pressure than the secondary system resulting in the SG tubes being in compression. This configuration reduces the likelihood of a tube rupture and eliminates the potential for pipe whip due to tube-side jetting.

Feedwater enters the SG tubes at their lowest point. As it rises through the tubes, it undergoes a phase change and is heated above saturation temperature before exiting the SG tubes as superheated steam. The configuration keeps the steam-water interface fluid and the superheated steam at the top of the tubes separated from the subcooled liquid at their bottoms. This configuration minimizes the hydraulic instabilities that could introduce potential sources of water hammer.

RAI 01-39

To ~~prevent~~control instabilities in individual SG tubes due to ~~two-phase conditions in the SG tubes as the~~ fluid ~~is~~ brought to boiling conditions as it travels up the tubes, inlet flow ~~resistance is~~restrictors are added at the feedwater inlet ~~plena~~plenum interface. Analysis shows that SG secondary side flow oscillations are decoupled from primary side flow oscillations and thus secondary side flow instabilities do not cause reactor power oscillations. In-phase oscillation of secondary flow does not occur, and the out-of-phase oscillatory flow in individual tubes cancels out, so that the net secondary flow is not oscillating. NRELAP5 analysis shows that an inlet loss coefficient (K) of at least 900 ensures that the tube mass flow rate is stable with fluctuations of less than ±10 percent for all power levels above 5 percent. Additional pressure loss is added to the inlet restriction to provide margin based on a comparison of the NRELAP5 results to test data.

The comprehensive vibration assessment program conforms to the guidance of Regulatory Guide (RG) 1.20, Revision 3. Based on the integral design of the NPM, the SG pressure retaining components are located within the fluid volume of the RPV, along with the reactor internal components. Therefore, the SGs and main steam piping up to and including the MSIVs and the SG tube supports are included in the comprehensive vibration assessment plan.

A set of flow-induced vibration screening criteria were developed for the comprehensive vibration assessment plan as described in Section 3.9. Under normal

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9093
Date of RAI Issue: 09/10/2017

NRC Question No.: 01-40

Title 10, the code of federal regulations (CFR), Part 50, Appendix A, General Design Criterion (GDC) 12- Suppression of reactor power oscillations, requires that oscillations be either not possible or reliably detected and suppressed. The Design-Specific Review Standard (DSRS), 15.9.A, "Design-Specific Review Standard for NuScale SMR Design, Thermal Hydraulic Stability Review Responsibilities," indicates that the applicant's analyses should correctly and accurately identify all factors that could potentially cause instabilities and their consequences. The analyses should also demonstrate that design features that are implemented prevent unacceptable consequences to the fuel. The Standard Review Plan (SRP) 15.0.2 acceptance criteria with respect to accident scenario identification states that the process must include evaluation of physical phenomena to identify those that are important in determining the figure of merit for the scenario.

Section 4.4, "Phenomena Identification and Ranking Table," of the topical report (TR), TR-0516-49417-P the primary side to secondary side heat transfer mechanisms are ranked as of medium importance. The TR disposition indicates that a lower ranking is warranted because errors in the modeling associated with these phenomena would tend to be self-cancelling. However, the physical process of heat transfer from the primary side to the secondary side is the key phenomenon affecting the change in density in the SG annulus of the power module.

In order to make an affirmative finding NRC staff requests NuScale to justify how the heat transfer in the SG warrants an importance ranking lower than high.

NuScale Response:

Heat transfer from the primary to the secondary side of the steam generator (SG) is an important phenomenon for the operation of the NPM. The medium ranking is not meant to diminish the importance of the phenomenon itself but a reflection on the impact of the variation of the net SG heat transfer coefficient (HTC) on stability, which has been judged to be less than high. Under steady state conditions, and regardless of the SG HTC magnitude, the heat transfer from the primary to the secondary side is equal to core thermal power (minus a small component accounting for ambient heat losses and heat loss from normal CVCS recirculation

flow). Under transient conditions, the variation of the HTC would result in a corresponding change on the primary coolant temperature distribution thus moving the center of gravity for density head calculation. However, this is a second order effect. An additional consideration, that was not provided in the TR table 4-1 comment, is the time scale for heat transfer across the SG tube. This effect has been cited in the response to RAI 9098 as having a time scale much smaller than the primary flow oscillation period and therefore has no significant impact on stability.

In conclusion, NuScale agrees that SG heat transfer is an important phenomenon, but the impact of its variation on stability is not high. As with other medium-ranked phenomena, the SG HTC modeling in PIM would not be different if a higher ranking were assigned.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0516-49417, Evaluation Methodology for Stability Analysis of the NuScale Power Module, as a result of this response.

Enclosure 3:

Affidavit of Thomas A. Bergman, AF-1117-57041

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 stability analysis of the NuSale 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. 9093, eRAI No. 9093. 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 11/9/2017.

Thomas A. Bergman