May 29, 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. 9466 (eRAI No. 9466) on the NuScale Topical Report, "Non-Loss of Coolant Accident Analysis Methodology," TR-0516-49416, Revision 1

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9466 (eRAI No. 9466)," dated May 07, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 9466 (eRAI No.9466)," dated September 27, 2018
3. NuScale Topical Report, "Non-Loss of Coolant Accident Analysis Methodology," TR-0516-49416, Revision 1, dated August 2017

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 Question from NRC eRAI No. 9466:

- 15.00.02-12

Enclosure 1 is the proprietary version of the NuScale Supplemental Response to NRC RAI No. 9466 (eRAI No. 9466). 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 Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

Michael Melton
Manager, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Rani Franovich, NRC, OWFN-8H12

Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9466, proprietary

Enclosure 2: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9466, nonproprietary

Enclosure 3: Affidavit of Thomas A. Bergman, AF-0519-65734

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9466, proprietary

Enclosure 2:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9466, nonproprietary

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9466
Date of RAI Issue: 05/07/2018

NRC Question No.: 15.00.02-12

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. In addition, GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

TR-0516-49416-P supports the conclusions regarding GDC 10 and 15 in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole. RG 1.203 describes the EMDAP, which the NRC staff considers acceptable for use in developing and assessing EMs used to analyze transient and accident behavior. RG 1.203 discusses the preparation of input and performance of calculations to assess model fidelity in Step 14 (Section 1.4.2) and states:

In particular, nodalization and option selection should be consistent between the experimental facility and similar components in the nuclear power plant. Nodalization convergence studies should be performed to the extent practicable in both the test facility and plant models.

It is important to understand how the test facility nodalization differs from the nodalization proposed for use in the non-LOCA licensing transient analyses since the adequacy of the NPM non-LOCA model is based, in large part, on assessments against test data. Different nodalization could impact the margin to SAFDLs and RCS pressure limits specified in GDC 10 and 15.

The discussions related to the nodalization of the test facility and plant models in TR Sections 5 and 6 are not clear. The nodalization diagrams that have been provided for both the various experimental assessment cases and the plant model are incomplete. In addition, for each of the representative calculations documented in Section 8 of the TR, any changes that may have been made to the nodalization diagrams in Section 6 are not indicated.

Information Requested:

1. Provide a detailed, complete, and legible nodalization diagram of the non-LOCA model used as the base for each of the assessment cases documented in TR Section 5, and update the TR accordingly.

2. Identify and provide the bases for the differences between the NRELAP5 NIST-1 test facility nodalization and the NRELAP5 NPM nodalization. Explain how the NIST-1 assessment results were utilized for NPM model nodalization.

3. For each representative calculation documented in TR Section 8, provide either a detailed and complete nodalization diagram, or explain the specific changes to the base model nodalization as used for each transient calculation. Update the TR accordingly.

NuScale Response:

The original NuScale response as submitted in NuScale correspondence RAIO-0918-61995 and dated September 27, 2018, is augmented with the following information.

Part 3 of the original response described how the pipe break for the main steam line and feedwater line break events for the representative calculations was modeled. A summary of the pipe break modeling was added to Section 7.2 of the Non-LOCA EM LTR (TR-0516-49416) as indicated at the end of this response.

The steam generator modeling commitments in the Non-LOCA EM (TR-0516-49416) are clarified below and summarized in Section 6.1.1 (for the steam generator primary side) and Section 6.1.4 (for the steam generator secondary side) of TR-0516-49416. A cross reference to these sections is added to Section 5.3.5.4 of TR-0516-49416 as indicated at the end of this response.

{{ }}[2(a),(c)] axial nodes are used to model the steam generator helical coils in the non-LOCA transient model. The primary to secondary side nodalization is one to one. The SIET-TF1 and SIET-TF2 nodalization sensitivity calculations with coarser nodalization indicate that uniform nodalization is adequate. The plant nodalization sensitivity calculations indicate that for consistent primary side initial conditions the impact of the steam generator nodalization on primary side transient conditions that affect margin to the MCHFR acceptance criterion are negligible . Therefore, generally, uniform steam generator nodalization will be used, although non-uniform axial nodalization with finer nodes near the inlet may be used if it is determined to be necessary to capture the phase change process in the lower region of the tubes. Consistent with the commitment in Section 6.1 that if finer nodalization is used, the applicable validation cases from SIET-TF1 and SIET-TF2 will be reviewed for consistency in the selected nodallization and revised as necessary.

During review of the steam generator modeling commitments, additional modeling commitment clarifications were identified:

- The non-LOCA plant model {{

  }}[2(a),(c)]
- Consistent with the commitment in Section 6.1 that where precise noding is described, the specified level of detail is considered the minimum level of detail required for the component of interest, the description of the bypass flow channel modeling is updated to specify "at least" {{ }}[2(a),(c)] are required.
- The description of the pool component to which the DHRS heat structures are connected is updated to emphasize that the pool temperature is bounded during the short-term non-LOCA transient analyses.
- The description of the ECCS is updated to reflect currently modeling and it is clarified that the specific modeling of the ECCS valves in the short-term non-LOCA transient does not affect the calculated transient response because the valves remain closed.
- The description of the reactor pool component is updated to emphasize that the pool temperature is bounded during the short-term non-LOCA transient analysis.
- The description of Table 6-2 is characterized as typical, as the module protection system control logic used in the non-LOCA transient analysis will be updated as needed to reflect the design.

Impact on Topical Report:

Topical Report TR-0516-49416, Non-Loss of Coolant Accident Analysis Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

5.3.5.4 Helical Coil Steam Generator Nodalization Sensitivity

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}}2(a),(c)

{{

}}2(a),(c)

Based on these studies, modeling the helical coil SG with {{ }}2(a),(c) nodes is expected to produce reasonably accurate results for the non-LOCA transients. Considering these studies, steam generator modeling requirements are summarized in Sections 6.1.1 and 6.1.4.2 for the primary and the secondary, respectively.

{{

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

Figure 5-186 Coil 1 representative pressure drop for {{ }}2(a),(c) nodes (left) and {{ }}2(a),(c) nodes (right)

The calculated fluid temperature inside the DHRS heat exchanger tubes predicted the trends of the measured data. {{

}}2(a),(c)

The calculated differential pressure across the DHRS steam line {{

}}2(a),(c)

Four different transients were performed for code-to-code benchmarking between NRELAP5 and RETRAN-3D: Reactivity insertion representative of a fast UCRW from full power conditions, reactivity insertion representative of a slow UCRW from full power conditions, negative reactivity insertion to reduce power from 100 percent to 50 percent power, and negative reactivity insertion simulating a dropped rod from 50 percent power. The results from all four of the transients showed that the comparison between the power and the total reactivity were consistently excellent, in that the calculation results of the two codes were nearly identically with one another.

NuScale's LOCA Topical Report (Reference 2) Section 7.3 discusses the validation of NRELAP5 for helical coil SG modeling. The validation was mainly against SIET TF-1 and TF-2 test data. The operating range of the helical coil SG primary and secondary side is demonstrated to be sufficiently covered by the validated range of NRELAP5. It was concluded that NRELAP5 showed reasonable to excellent agreement with test data for all phenomena at conditions important for the non-LOCA analysis.

A nodalization sensitivity of the steam generator for a main steam line break scenario was performed comparing the effect of modeling the SG {{

}}2(a)(c)

Considering the high-ranked phenomena identified from the PIRT process, the NRELAP5 code along with the NPM system model is applicable for calculation of the NPM system response for the non-LOCA short-term transient event progression as part of this EM based on separate effects and integral effects testing, code-to-code benchmarking, and appropriate conservative input for initial and boundary conditions.

6.0 NuScale NRELAP5 Plant Model

This section discusses the NuScale NRELAP5 non-LOCA plant transient model. A summary overview of the plant components and features simulated by the NRELAP5 model is provided, including the reactor primary and secondary (SG) systems, core fuel rods and kinetics, ECCS and DHRS, containment and reactor pool, and trips and controls.

The NRELAP5 plant model is developed to support the non-LOCA analysis methodology described in Section 7.0. The model was developed following the NRELAP5 code manual user guidelines, supplemented by NuScale-specific modeling guidelines. The guidelines describe how to model a NuScale Plant Module using the NRELAP5 code, and include directions on how to select code options, nodalizing the system, and selecting heat transfer correlations.

6.1 Thermal-Hydraulic Volumes and Heat Structures

The NRELAP5 plant model contains multiple hydraulic components, heat structures and junctions. The model simulates the majority of a typical NPM (Figure 6-1) including the RPV and internals, the containment, and the reactor cooling pool. {{

}}2(a),(c) Both DHRS trains are included in the model, along with the ECCS consisting of the RVVs and RRVs. Control system components include variable and logical trips, control blocks and general tables. Figure 6-2 shows a typical nodalization diagram for the primary and secondary systems and is meant to convey the overall model structure rather than show nodalization details of any particular component.

Figure 6-3 shows a cut-away of the typical NPM reactor coolant system and CNV with the key nodalization regions included in the NRELAP5 model. The circled numbers in the figure represent RCS fluid regions and the numbers in squares represent containment regions. Table 6-1 lists the RCS regions and the associated NRELAP5 components.

This NRELAP5 model of the NPM serves as the standalone baseline model for non-LOCA safety analysis, as well as various aspects of plant design support. The information presented herein describes the base model as it is configured for non-LOCA analysis.

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}}2(a),(c)

{{

}}2(a),(c)

Figure 6-2 Typical primary and secondary side nodalization (heat structures and component cell details excluded)

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}}2(a),(c)

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}}2(a),(c)

Figure 6-5 Core and lower plenum nodalization

6.1.4 Secondary System

6.1.4.1 Feedwater System

In the NPM design two feedwater lines penetrate the CNV immediately downstream of the FWIVs. Each feedwater line splits into two lines before connecting to the SGs. {{

}}2(a),(c)

6.1.4.2 Steam Generator Secondary

The NRELAP5 specific helical coil SG component ('hlcoil') is used to simulate the helical coil SG that is characteristic of the NPM (Reference 2 describes the helical coil component). The steam generator nodalization is shown in Figure 6-11.The primary coolant flows through the SG shell side while the feedwater and steam flow through the tube side. The tube and shell side of the SG elevation nodalization schemes are one-to-one. The SG nodes are uniform, or may be finer towards inlet and coarser towards the exit~~. This scheme is chosen~~ if necessary to capture the phase change process in the lower region of the tubes. For non-LOCA transient calculations, finer nodalization is not needed near the tube exits due to the state of the fluid being single phase vapor. Section 5.3.5.4 summarizes SG nodalization sensitivity calculations. Based on these studies, modeling the helical coil SG with {{ }}2(a),(c) nodes is expected to produce reasonably accurate results for the non-LOCA transients. Should additional detail be needed in the future, the relevant benchmarks, sensitivity studies, and transient analyses will be reviewed for continued applicability and updated as necessary to demonstrate the higher level of detail for the component is applicable to the NPM.

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}}2(a),(c)

{{

}}2(a),(c)

Figure 6-11 Steam generator nodalization

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}}2(a),(c)

~~Although i~~In the actual NPM the DHRS heat exchanger is located in the reactor cooling pool.~~,~~ {{

}}2(a),(c)

{{

}}2(a)(c)

Figure 6-14 Not used.~~Decay heat removal system pool nodalization~~

6.1.6 Emergency Core Cooling System

The ECCS hydrodynamic components consist of two reactor recirculation valves (RRVs) and three reactor vent valves (RVVs). {{

}}2(a),(c)

~~{{~~

Figure 6-15 Not used.~~Emergency core cooling system valves~~ ~~}}2(a),(c)~~

6.1.7 Containment Vessel

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}}2(a),(c)

{{

}}2(a),(c)

Figure 6-16 Containment and reactor pool nodalization

6.1.8 Reactor Cooling Pool

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}}2(a),(c)

6.2 Material Properties

Thermal properties (thermal conductivity and volumetric heat capacity) are specified by user input in the NRELAP5 non-LOCA base model for the following materials used in the heat structures. These material properties may be amended or revised as the NPM design evolves:

1. fuel cladding (AREVA's M5® cladding)
2. inconel 690 (SG tubes)
3. uranium dioxide (UO_2)
4. stainless steel (SA-240 304L)
5. fuel-to-cladding gas gap (initially pressurized helium at BOC; mixture of fission product gases and helium after irradiation)
6. carbon steel (SA-508)
7. {{

}}2(a),(c)

6.3 Control Systems

With its combination of trips, control functions, and user-defined tables, NRELAP5 provides flexibility to accurately simulate plant control and protection system responses during both steady-state and transient operation. The NRELAP5 non-LOCA base model contains logic for "normal controls" that simulate normal operational plant response, as well as user-convenience controls that make it easier to initialize the model for particular transients and easier to interpret the transient results. It also contains trip and control logic that accurately simulates the MPS, i.e., the safety-related trips that protect the reactor core and fission product boundaries.

6.3.1.6 Containment Pressure Control (Nonsafety-related)

The containment pressure is established at sub-atmospheric conditions via operation of the containment evacuation system. The impact of this system continuing to operate is considered for the non-LOCA transient analyses.

6.3.2 Module Protection System (Safety-related)

6.3.2.1 Analytical Limits and Delays

The MPS implemented in the NRELAP5 base model is intended for the purposes of performing safety analysis transient simulations. As such, the logic and actuation points are based on the NPM safety analysis analytical limits. Fixed delay times are specified considering different sensor response times. {{

}}2(a),(c) In addition to the sensor delays, a given safety signal is subject to instrumentation string delays, an MPS processing delay, and an actuation delay. The NRELAP5 non-LOCA model incorporates the methodology assumption that a bounding total for these additional delays is applied as a signal delay in addition to the individual sensor delay.

Table 6-2 shows the type of safety signals for the NuScale NPM design. Signals in bold are included in the NRELAP5 non-LOCA model. The control logic for other MPS signals may be added if needed for a particular event analysis or as necessary to maintain consistency with the MPS design.

7.2.4 Steam System Piping Failure Inside or Outside of Containment

The methodology used to simulate a postulated steam system piping failure for the NPM, and an evaluation of the resulting representative plant response against the acceptance criteria listed in Table 7-4, are presented below. Since both split breaks (relatively higher event frequency) and double-ended guillotine breaks (relatively lower event frequency) are analyzed, the more restrictive AOO criteria for system pressures, critical heat flux ratio, and fuel centerline melt applicable to breaks with higher event frequency are used in the evaluation. Radiological dose consequences are assessed as part of the downstream accident radiological dose analysis, documented in a separate report, and compared against the appropriate acceptance criteria.

A description of the event including biases and conservatisms, sensitivity studies, single active failure (SAF) and loss of power (LOP) scenarios, challenging case, and acceptance criteria evaluation are presented in the following sections.

7.2.4.1 General Event Description

The steam line break event ranges from small breaks to double ended ruptures of a main steam line causing an increase in steam flow and an over cooling of the RCS. This event can occur inside or outside the containment vessel (CNV). A break inside CNV will cause a rapid pressurization of the CNV resulting in a reactor trip and CNV isolation with a DHRS actuation. This break location is non-limiting for pressure and MCHFR but challenging to the DHRS as one loop is disabled with the break inside the CNV. A steam line break outside of the CNV will cause an increase in steam flow event that will cause either a low SG pressure trip or a high core power trip due to the reactor power response from the decreased RCS temperature. The break flow will be stopped by the MSIVs closing and depressurization of the steam system piping. Smaller breaks will cause a slower loss of secondary pressure due to the increased steam demand which could cause a high core power trip. These smaller breaks can result in a significant delay in detection time, making the small break cases challenging for MCHFR. Reactor trip and transition to stable DHRS flow eventually terminate the transients and bring the NPM to a safe, stable condition.

For this overcooling event, the high power analytical limit is increased, for example from 120 percent to 125 percent RTP. This is to account for the decalibration of the excore neutron detectors as downcomer density increases in response to a cooldown event. The increase is based on an appropriate decalibration factor (change-in-power-per-change-in-temperature) and considering the downcomer temperature decrease during the overcooling events.

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}}2(a)(c)

7.2.12 Feedwater System Pipe Break Inside or Outside Containment

The methodology used to simulate a postulated feedwater system pipe break for the NPM, and an evaluation of the resulting representative plant response against the acceptance criteria listed in Table 7-4, are presented below. Since both split breaks (relatively higher event frequency) and double-ended guillotine breaks (relatively lower event frequency) are analyzed, the more restrictive AOO criteria for system pressures, critical heat flux ratio, and fuel centerline melt applicable to breaks with higher event frequency are used in the evaluation.

A description of the event including biases and conservatisms, sensitivity studies, single active failure (SAF) and loss of power (LOP) scenarios, challenging case, and acceptance criteria evaluation are presented in the following sections.

7.2.12.1 General Event Description

A feedwater line break can occur inside or outside of containment, and can range in size from a small split crack to a double ended rupture. A feedwater line break inside containment results in a loss of containment vacuum and a high containment pressure MPS signal that actuates a reactor trip, isolates the secondary system and CVCS, and opens the DHRS valves. The steam generator, DHRS piping, and DHRS condenser on the affected side drain through the break. The non-affected steam generator system and DHRS loop provide cooling to the RCS via heat transfer to the reactor pool.

A feedwater line break outside containment causes a loss of feedwater flow to the steam generators and a heatup of the RCS. Larger breaks result in rapid heatup events that pressurize the RCS beyond the high PZR pressure analytical limit. Smaller breaks cause a more gradual heatup, loss of secondary pressure, and reactor trip and DHRS actuation on low steam pressure or high steam superheat. The DHRS provides cooling to the RCS via heat transfer to the reactor pool. Reactor trip and transition to stable DHRS flow terminates the transient with the NPM in a safe, stable condition.

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}}2(a),(c)

Table 7-54 lists the relevant acceptance criteria, SAF, and LOP scenarios.

Breaks outside of containment result in higher system pressures compared to breaks inside containment because of the relatively rapid MPS signal on high containment pressure for breaks inside of containment. The limiting pressure responses occur for breaks outside of containment, when the event is initiated from full power conditions, and the initial conditions are biased in the conservative directions. A loss of normal AC power at the event initiation provides the most challenging peak primary pressure for this event.

Enclosure 3:

Affidavit of Thomas A. Bergman, AF-0519-65734

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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 non-loss of coolant accident analysis methodology.

   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. 9466, eRAI 9466. 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 May 29, 2019.

Thomas A. Bergman