RAIO-1017-56660

October 18, 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. 8776 (eRAI No. 8776) on the NuScale Topical Report, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Revision 0

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 8776 (eRAI No. 8776)," dated August 19, 2017
2. NuScale Topical Report, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Revision 0, dated December 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. 8776:

- 15.06.05-2
- 15.06.05-3
- 15.06.05-4
- 15.06.05-5
- 15.06.05-6

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 8776 (eRAI No. 8776). 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,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8776, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8776, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-1017-56706

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8776, proprietary

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 8776, nonproprietary

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8776
Date of RAI Issue: 08/19/2017

NRC Question No.: 15.06.05-2

Title 10 of the Code of Federal Regulations (10 CFR) Part 52, Section 47 (a)(2) states, "The application must contain a Final Safety Analysis Report (FSAR) that… must include… a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished." Regulatory Guide 1.203 describes a process that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

As stated in RG 1.203, an evaluation model (EM) is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, as illustrated by the following examples:

(1) procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation)
(2) specification of those portions of the analysis not included in the computer programs for which alternative approaches are used
(3) all other information needed to specify the calculational procedure

The entirety of an evaluation model (EM) ultimately determines whether the results are in compliance with applicable regulations. Therefore, the development, assessment, and review processes must consider the entire EM.

The Loss-of-Coolant-Accident (LOCA) model as described in TR-0516-49422, Rev. 0, is used as the containment (CNV) pressure EM. The peak CNV pressure during a LOCA is impacted by heat transfer from containment through the CNV wall to the reactor building pool. The NuScale NRELAP5 LOCA model {{ }}$^{2(a),(c)}$ to model heat transfer from the CNV to the NPM pool bay water during a LOCA. {{ }}$^{2(a),(c)}$

{{

}}2(a),(c) This nodalization implicitly assumes that heat transfer happens instantly from the containment outer surface boundary layer to the bulk of the combined water mass represented by the PIPE. Because reactor building pool mixing may be minimal, the {{

}}2(a),(c) and the associated instantaneous heat transfer, may produce non-conservative peak containment pressures. As little margin exists, the staff is requesting that NuScale determine the impact of {{

}}2(a),(c)

NuScale Response:

The methodology to determine the maximum containment pressure is described in the Containment Response Analysis Methodology Technical Report (Reference 1). The methodology for the containment peak pressure calculation for primary system releases is derived from the Loss-of-Coolant Accident (LOCA) Evaluation Model (EM) described in Reference 2, and changes to the LOCA EM for the purpose of the peak containment pressure calculations are described in Reference 1. In both the peak containment pressure calculations and in the LOCA EM, the NRELAP5 {{

}}2(a),(c).

In order to quantify the impact of including the {{

}}2(a),(c), two sets of sensitivity calculations were performed. The first set was performed on limiting cases from the containment peak pressure analysis described in Reference 1. The second set of sensitivity calculations were performed for a limiting small break LOCA. As summarized in Section 9.7 of Reference 2, the smaller LOCA cases are more limiting with respect to minimum collapsed water level above the top of active fuel. The decay heat removal system (DHRS) heat removal is not credited in the LOCA cases, so the inadvertent actuation block (IAB) in each emergency core cooling system (ECCS) valve delays the opening of the ECCS valves after actuation for an extended period of time. The delayed opening of the ECCS valves allows more liquid inventory to be redistributed from the RCS to the containment vessel (CNV) before the ECCS valves open. This results in a momentary lower calculated level in the reactor coolant system (RCS) when ECCS valves open. For both the limiting containment peak pressure cases and the limiting small break LOCA case, sensitivity calculations are performed {{

}}2(a),(c)

Two cases from Reference 1 are considered for the containment peak pressure sensitivity calculations: the limiting peak containment pressure case, and the limiting primary system pipe

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break case. The limiting peak containment pressure case is for an inadvertent reactor recirculation valve (RRV) opening event with assumed loss of AC and DC power, no single failure, no DHRS, and 1000 psid IAB release pressure. The limiting RRV opening event results in both opening of the ECCS valves and maximum pressure within approximately 2 minutes after the event initiation. In a case with later opening of the ECCS valves, {{

}}2(a),(c)

Table 1 summarizes the results of the containment peak pressure sensitivity calculations. {{

}}2(a),(c)

The results in Table 1 show the {{

}}2(a),(c)

Next, the small break LOCA cases from Reference 2 are considered for impact on minimum level above top of active fuel. In development of the LOCA EM, the break spectrum calculations evaluated break location, break size, DHRS availability, single active failures, and loss of power. The case chosen here for the sensitivity calculation is a 5% break in the charging line, no DHRS, loss of AC power, and a single failure of a RVV.

{{

}}2(a),(c)

{{

}}2(a),(c)

Table 2 summarizes the results of the small break LOCA sensitivity calculations. Figure 3 and Figure 4 show the sensitivity calculation results for the minimum collapsed level, and containment pressure, respectively. For the sensitivity calculations, {{

}}2(a),(c)

The results shown in Table 2 for minimum collapsed liquid level in the riser show that {{

}}2(a),(c)

The results of these sensitivities show that the modeling approach in the LOCA EM and in the peak containment pressure calculations is adequate. {{

}}2(a),(c)

References

1. TR-0516-49084, Revision 0, "Containment Response Analysis Methodology Technical Report."
2. TR-0516-49422, Revision 0, "Loss-of-Coolant Accident Evaluation Methodology."

{{

}}2(a),(c)

Figure 1 - RRV Case Sensitivity Results, Maximum Containment Pressure

{{

}}2(a),(c)

Figure 2 - Injection Line Break Sensitivity Results, Maximum Containment Pressure

{{

}}2(a),(c)

Figure 3 - Small LOCA Case Sensitivity Results, Minimum Collapsed Level

{{

}}2(a),(c)

Figure 4 - Small LOCA Case Sensitivity Results, Maximum Containment Pressure

{{

}}2(a),(c)

Figure 5 - Small LOCA Case Sensitivity Results, {{ }}2(a),(c) Liquid Temperature

{{

}}2(a),(c)

Figure 6 - Small LOCA Case Sensitivity Results, {{ }}2(a),(c) Vapor Quality

Table 1 - CNV Pressure Sensitivity Results

{{

}}2(a),(c)

Table 2 - Small LOCA Case Sensitivity Results for Minimum Collapsed Level

{{

}}2(a),(c)

Impact on Topical Report:
There are no impacts to the Topical Report TR-0516-49422, Loss-of-Coolant Accident Evaluation Model, as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8776
Date of RAI Issue: 08/19/2017

NRC Question No.: 15.06.05-3

Title 10 of the Code of Federal Regulations (10 CFR) Part 52, Section 47 (a)(2) states, "The application must contain a Final Safety Analysis Report (FSAR) that… must include… a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished." Regulatory Guide 1.203 describes a process that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

As stated in RG 1.203, an EM is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, as illustrated by the following examples:

(1) procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation)
(2) specification of those portions of the analysis not included in the computer programs for which alternative approaches are used
(3) all other information needed to specify the calculational procedure

The entirety of an EM ultimately determines whether the results are in compliance with applicable regulations. Therefore, the development, assessment, and review processes must consider the entire EM.

NRC Information Notice 92-02 and 92-02 Supplement 1 indicate that whenever the RELAP5 code is applied to situations in which the pressure drops significantly between cells, the energy in the downstream volume may be underestimated. To address this issue, NuScale modified {{ }}$^{2(a),(c)}$ described in Equation 2.10-190 (Report SwUM-0304-17023, Rev 4, "NRELAP5 Code Manual Theory Manual: Models, Correlations and Solution Methods). The staff understands that this feature is invoked by {{ }}$^{2(a),(c)}$ and that it is used in conjunction with Henry-Fauske and Moody critical flow

models (c=3) for LOCA Appendix K applications. However, the impact of changing {{ }}2(a),(c) pressure was not discussed by the applicant in either the topical report or the theory manual.

The staff is requesting that NuScale demonstrate how (1) the {{ }}2(a),(c) with the presence of the Moody critical flow correlation or the modified Henry-Fauske critical flow model addresses the issues presented in NRC Information Notice 92-02 and (2) the total enthalpy is preserved across the junction modeling the break location.

NuScale Response:

As described in Section 6.0 of the Loss-of-Coolant Accident Evaluation Model Topical Report (Reference 1), RELAP5-3D© version 4.1.3 was used as the baseline development platform for the NRELAP5 code. A commercial grade dedication was performed by NuScale to establish the baseline NRELAP5 code. Subsequently, features were added and changes made to NRELAP5 to address the unique aspects of the NuScale design and licensing methodology. Those aspects of NRELAP5 that are new or revised specifically for the Loss-of-Coolant Accident application are discussed in Section 6.0 of Reference 1. NRELAP5 does contain a {{ }}2(a),(c) as noted; however, NuScale did not {{ }}2(a),(c) as stated. Rather, this modification was already part of the commercial grade dedicated code RELAP5-3D obtained from the Idaho National Lab, which subsequently became NRELAP5.

Since the RELAP5 series of codes do not explicitly account for the {{

}}2(a),(c)

{{

}}2(a),(c)

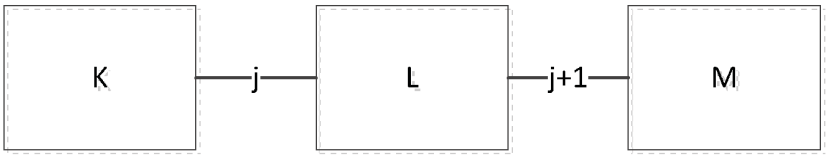

Figure 1. Typical NRELAP5 nodal diagram.

The impact of neglecting the {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 2. Energy conservation at an orifice located, obtained from Reference 4.

{{

}}2(a),(c)

Figure 3. Temperature up and downstream of an orifice, obtained from Reference 4.

Additional discussion and verification of the {{

}}2(a),(c)

The choking models are considered separately and activation of {{

}}2(a),(c)

References

1. TR-0516-49422, Revision 0, "Loss-of-Coolant Accident Evaluation Model."
2. SwUM-0304-17023, "NRELAP5 Theory Manual."
3. Stewart, M., “Total Energy Balance Calculation Across a Break”, NRC 189, RELAP5 Improvements for AWLRs, fin L2537, Subtask, Energy Balance out the Break, 12/14/92.
4. Stewart, M., Riemke, R., “Modification to Improve Energy Conservation at an Abrupt Geometry Change in Relap5,” 1993 Relap5 International Users Seminar, Sheraton Boston, Boston, Mass., August 6-9, 1993.
5. RELAP5/MOD3 Code Manual Volume 1, Idaho National Laboratory, NUREG/CR-5535, INEL-95/0174.
6. Wang, W., Mortensen, G., “Energy Conservation in Relap5,” SCIE-NRC-392-99, November, 1999.

Impact on Topical Report:
There are no impacts to the Topical Report TR-0516-49422, Loss-of-Coolant Accident Evaluation Model, as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8776
Date of RAI Issue: 08/19/2017

NRC Question No.: 15.06.05-4

Title 10 of the Code of Federal Regulations (10 CFR) Part 52, Section 47 (a)(2) states, “The application must contain a Final Safety Analysis Report (FSAR) that… must include… a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished.” Regulatory Guide 1.203 describes a process that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

As stated in RG 1.203, an evaluation model (EM) is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, as illustrated by the following examples:

(1) procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation)
(2) specification of those portions of the analysis not included in the computer programs for which alternative approaches are used
(3) all other information needed to specify the calculational procedure

The entirety of an EM ultimately determines whether the results are in compliance with applicable regulations. Therefore, the development, assessment, and review processes must consider the entire EM.

One of the key parameters that the Loss-of-Coolant Accident EM must be able to calculate to demonstrate compliance with design requirements during a LOCA is the peak containment pressure. On pages 252-254. The applicant concludes "Overall, NRELAP5 predicted the HP-06b data with reasonable-to-excellent agreement." However, Figure 7-102, "NIST-1 HP-06b containment vessel pressure comparison" on page 254 of LOCA EM Topical Report, TR-0516-49422, shows that NRELAP5 underestimated {{ }}$^{2(a),(c)}$

{{ }}[2(a),(c)] when compared to NIST test results. This suggests uncertainty in the code's predictive capability for a full-scale and full-pressure actual plant with only 49 psi of margin. Also, the topical report does not describe the NRELAP5 code uncertainty quantification with regard to the peak containment pressure.

The staff is requesting that NuScale provide additional detail regarding how the {{ }}[2(a),(c)] data is included and accounted for in the LOCA EM and how the NRELAP5 code containment pressure uncertainty has been determined.

NuScale Response:

1.0 Background

The peak containment vessel (CNV) pressure is identified as one of the Figures-of-Merit (FOMs) for the development of the Loss-of-Coolant Accident (LOCA) evaluation model (EM). However, as clarified in Section 4.3 of Reference 1, the design-basis analysis for peak CNV pressure is performed using a separate methodology presented in Reference 2. This methodology uses an NRELAP5 model that is similar to the model used in the LOCA EM with additional conservatisms that maximize the CNV pressure and temperature response to primary and secondary system release events by maximizing the mass and energy release to the CNV and minimizing the performance of the CNV heat removal system. The validations of NRELAP5 against the separate effects test (SET) and the NIST-1 integral effects tests (IET) data presented in Reference 1 are also applicable to the CNV response analysis methodology.

2.0 NRELAP5 Assessments of NIST-1 Integral Effects Test Data

Sections 7.5.6 through 7.5.8 of Reference 1 summarize the assessment of NRELAP5 against the NIST-1 IET data. The NIST-1 IETs considered for the validation of NRELAP5 include a {{

}}[2(a),(c)] In addition to the IETs used to validate NRELAP5 in Reference 1, NuScale performed an additional NIST test (HP-43) with the same conditions as HP-09, but with a higher decay heat power. Results from the HP-43 test and NRELAP5 assessment will be discussed in this response to provide additional support to the conclusions drawn from the NRELAP5 validation tests from Reference 1. The acceptance criteria used in Reference 1 for characterizing the results of these assessments are described in Table 1-2 of Reference 1. These criteria are based on the definitions provided in Regulatory Guide (RG) 1.203 and take into consideration the ability of NRELAP5 to predict the overall data trends as well as the magnitude of the data itself.

As described in RG 1.203, “reasonable” agreement applies when the code exhibits minor deficiencies. Overall, the code provides a reasonable prediction. All major trends and

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phenomena are correctly predicted. Differences between the calculation and data are greater than deemed necessary for excellent agreement. The calculation will frequently lie outside but near the specified or inferred uncertainty bands of the data. However, the correct conclusions about trends and phenomena would be reached if the code was used in similar applications.

The use of “excellent” agreement applies when the code exhibits no deficiencies in modeling a given behavior. Major and minor phenomena and trends are correctly predicted. The calculated results are judged to agree closely with the data. The calculation will, with few exceptions, lie within the specified or inferred uncertainty bands of the data. The code may be used with confidence in similar applications.

The results presented in Sections 7.5.6 through 7.5.8 of Reference 1 show that {{

}}2(a),(c)

3.0 Uncertainties in NIST-1 Assessments

The NRELAP5 assessments against the NIST-1 IETs were performed using the best estimate approach. The boundary conditions and initial conditions for the assessments were specified based on the measured values during the experiment. Examples of the boundary and initial conditions assigned based on the experimental measurements include:

- {{

}}2(a),(c)

Furthermore, these assessments also used the best estimate models and the model input parameters. Examples of such models include:

- {{

}}2(a),(c)

Table 1 shows the comparison of measured and NRELAP5-calculated peak CNV pressures for

different NIST-1 IETs. The maximum difference between the measured and predicted peak CNV pressures is {{

}}2(a),(c)

Table 1 Comparison of measured and predicted peak NIST-1 IET CNV pressures

{{

}}2(a),(c)

In addition to the uncertainties in the CNV pressure measurement instrumentation, uncertainties also exist in NRELAP5 calculations for specifying the initial and boundary conditions based on the measured data and assigning the values for some of the model input parameters. NRELAP5 sensitivity calculations were performed for various NIST-1 IET assessments to analyze the impact of these uncertainties on predictions of important parameters. These calculations show that the uncertainties affecting the calculation of peak CNV pressure are:

{{

}}2(a),(c)

3.1 Uncertainties in {{ }}[2(a),(c)]

The {{

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

{{

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

{{

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

Figure 1 Comparison of CNV pressure in {{

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

{{

}}2(a),(c)

Figure 2 Comparison of CNV pressure in {{

}}2(a),(c)

In the LOCA EM, the uncertainties in modeling {{

}}2(a),(c)

{{

}}2(a),(c)

3.2 Uncertainties in {{

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

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

The HTP area in the NIST-1 facility is scaled to represent the NPM CNV wall heat transfer area. As described in response to eRAI 8777, Question 15.06.05-1, the presence of {{

}}2(a),(c),ECI were discussed in response to eRAI-8777.

The data for NIST-1 IETs HP-06, HP-07, HP-09, and HP-06b shows that {{

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

{{

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

As shown in Figure 3, a linear fit was used between {{

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

{{

}}2(a),(c)

Figure 3 HP-43 pre-break {{ }}2(a),(c), ECI

{{

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

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

NIST-1 {{

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

{{

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

{{

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

Figure 5 HP-43 CNV pressure sensitivity to {{ }}2(a),(c), ECI

The above sensitivity calculations show that the {{

}}2(a),(c),ECI which accounts for uncertainty in specifying these boundary conditions.

3.3 Uncertainties in NIST-1 {{ }}2(a),(c)

{{

}}2(a),(c)

{{

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

{{

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

Figure 6 HP-43 CNV pressure sensitivity to {{ }}2(a),(c), ECI

4.0 NPM Sensitivity calculations

The initial and boundary conditions in the NPM CNV performance analysis methodology are biased to the conservative ends of the operating range (e.g. PZRpressure, PZR level, RCS average temperature, reactor power, initial CNV pressure, reactor pool temperature, reactor pool level, RCS flow) to maximize the pressure and temperature response of the CNV. A sensitivity study was performed to quantify the conservatism in the CNV peak pressure results provided by biasing these initial conditions.

Table 2 shows {{

}}2(a),(c)

Table 2 - Initial Condition Sensitivity

{{

}}2(a),(c)

Table 3 shows the comparison of the base case (i.e., biased initial and boundary conditions) and the sensitivity case (i.e., nominal conditions) results. The peak CNV pressure for the biased initial condition case is {{ }}2(a),(c) than that of the nominal initial condition case for the same limiting inadvertent RRV opening scenario. Note that the transient in these

calculations begins at 50 seconds.

Table 3 Limiting Inadvertent RRV Opening Case (Biased vs. Nominal Initial Conditions)

{{

}}2(a),(c)

5.0 Summary

The validation of NRELAP5 against the NIST-1 data shows that the {{

}}2(a),(c)

Furthermore, the uncertainty in modeling of {{

}}2(a),(c) Moreover, as shown Section 4 of this response, the initial and boundary conditions in the NuScale CNV performance analysis methodology are {{

}}2(a),(c)

There are further conservatisms included in the LOCA EM and the CNV performance analysis methodologies such as the 10CFR50 Appendix K assumptions along with {{ }}[2(a),(c)] in the LOCA EM methodology. There is a high confidence that these conservatisms bound uncertainty in code calculated peak CNV pressure.

References

1. TR-0516-49422, Revision 0, "Loss-of-Coolant Accident Evaluation Model."
2. TR-0516-49084, Revision 0, "Containment Response Analysis Methodology Technical Report."

Impact on Topical Report:
There are no impacts to the Topical Report TR-0516-49422, Loss-of-Coolant Accident Evaluation Model, as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8776
Date of RAI Issue: 08/19/2017

NRC Question No.: 15.06.05-5

Title 10, Part 50, of the Code of Federal Regulations (10 CFR Part 50), “Domestic Licensing of Production and Utilization Facilities,” Appendix K, “ECCS Evaluation Models,” and Section C.1, “Break Characteristics and Flow,” specify the requirement to analyze a spectrum of possible pipe breaks.

In Chapter 3, Section 3.6 of the NuScale Final Safety Analysis Report (FSAR), the applicant has indicated that all piping ends at the welding point between the vessel nozzle and the connecting pipes or valve components. Following this approach, both the Reactor Vent Valve (RVV) and Reactor Recirculation Valve (RRV) break locations should be at the welding points between the vessel nozzle and the valve components. It is not clear that the applicant has identified the break locations at those welding points in FSAR Section 15.6.5 or in the LOCA topical report TR-0516-49422. Consistent with BTP 3-4, Revision 2 (as referenced on Page 3.6-5 of NuScale FSAR), NRC staff considers the welding points between the vessel nozzle and the valve components to be potential break points.

The NRC staff is requesting NuScale to provide additional justification for why (1) the existing break spectrum analyses have adequately considered the welding locations between RVV and RRV components and the vessel nozzles and (2) breaks at these locations do not have to be considered as part of the LOCA break spectrum analysis. If larger break sizes are identified, provide additional necessary LOCA analyses and results.

NuScale Response:

FSAR Section 3.6 discusses the criteria used to postulate piping break locations for the purposes of pipe rupture protection pursuant to General Design Criterion (GDC) 4, for which NuScale conforms to the guidance provided in BTP 3-4. GDC 4 requires that SSCs important to safety "be appropriately protected against dynamic effects . . . that may result from equipment failures." GDC 35, as implemented by 10 CFR 50.46, is the regulatory requirement governing emergency core cooling capability (the NuScale design conforms to PDC 35, which includes differences not pertinent to this discussion). The scope of 10 CFR 50.46 is limited to "postulated

loss-of-coolant accidents," which are explicitly defined at subsection (c) as those "from breaks in pipes in the reactor coolant pressure boundary." 10 CFR 50 Appendix K paragraph C.1 further states "a spectrum of possible pipe breaks shall be considered." Standard Review Plan 15.6.5 does not cite BTP 3-4 as an acceptance criterion or reference, and does not reference SRP Section 3.6 as a review interface. Accordingly, NuScale does not consider BTP 3-4 applicable to the selection of an acceptable spectrum of postulated pipe break locations for the purposes of demonstrating emergency core cooling capability.

"Pipes" and "piping" are not defined by 10 CFR Part 50. Pursuant to 10 CFR 50.55a(a), however, the components of the NuScale RCPB are designed to "meet the requirements for Class 1 components in Section III of the ASME BPV Code." BPVC Section III, NCA-9200 defines a "component" as "a vessel, . . . pressure relief valve, line valve, storage tank, piping system, <u>or</u> core support structure" [emphasis added]. It then defines "piping system" as "an assembly of piping, piping supports, components, <u>and</u>, if applicable, component supports" [emphasis added]. Therefore, pursuant to Section III, a component is classified as either a vessel or a piping system. Further, while a piping system may include non-piping components such as a valve, a piping system must at least include piping. Section III NB-1131 provides that "the Design Specification shall define the boundary of a component to which piping or another component is attached."

In the NuScale design, there is no piping between the Reactor Pressure Vessel (RPV) nozzles and Reactor Vent Valves (RVVs)/Reactor Recirculation Valves (RRVs). The welds between these valves and the RPV are classified as part of the ASME Class 1 vessel component.The welds between the nozzles and valves are not classified as ASME Class 1 or 2 piping in their respective design specifications. Therefore, pursuant to ASME Section III, there is no piping system but rather only two non-piping components welded together.

NuScale is unaware of any statement in FSAR Section 3.6 to the contrary, specifically there are no statements in FSAR Section 3.6 that are intended to imply that welds between a vessel nozzle and a connecting valve component are to be identified as postulated break locations. Even if the statement "that all piping ends at the welding point between the vessel nozzle and the connecting pipes or valve components" were present in the FSAR, this would imply that the piping also begins. Absent any piping beginning, as explained above, there can be no piping end.

Although BTP 3-4 is not applicable to postulated break locations for safety analysis purposes, NuScale arrives at the same conclusion in applying it to the welds in question. BTP 3-4 contains no statements suggesting that a weld between a pressure vessel nozzle and an attached valve should be a postulated break location. Pipe break postulation criteria contained in BTP 3-4 is only given for ASME Class 1 and 2 piping. Again, the welds between the Reactor Pressure Vessel (RPV) nozzles and Reactor Vent Valves (RVVs)/Reactor Recirculation Valves (RRVs) are not classified as ASME Class 1 or 2 piping in their respective design specifications.

BTP 3-4 addresses locations where a piping system connects to a vessel nozzle. However, this

is not applicable to the configuration of the RVVs or the RRVs, which consist of a valve attached to a pressure vessel nozzle with no intermediate or downstream piping attached (½ inch tubing which is connected to the valve has a negligible effect on the valve and nozzle). The term "nozzle" is used within BTP 3-4 as repeated below:

> The staff's observation of actual piping failures has indicated that they generally occur at high stress and fatigue locations, such as at the terminal ends of a piping system at its connection to the nozzles of a component.

Accordingly, the guidance in the BTP generally requires the postulation of pipe breaks at piping terminal ends, with exceptions given for piping within containment penetration areas. A piping "terminal end" is defined in Footnote 3. It states:

> This term is defined as the extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection to a main piping run is a terminal end of the branch run, except where the branch run is classified as part of a main run in the stress analysis and is shown to have a significant effect on the main run behavior. . .

As indicated above, the BTP 3-4 guidance regarding terminal ends of piping runs, which may or may not be a vessel nozzle, addresses piping thermal expansion and piping system motion. Absent piping, there can be no piping run, and therefore no piping system or terminal end thereof. This reading is supported by design and analytical considerations. Piping systems can include long pipe runs which, when heated, experience thermal expansion. Long pipe runs may also move significantly during a seismic event. The highest stressed location due to thermal expansion and seismic events can be a vessel nozzle location, which acts as a restraint in six degrees of freedom. Additionally, relative motion between a nozzle connection and the closest support in the piping system during a seismic event (i.e., seismic anchor movement) causes stress at the nozzle location. These characteristics do not apply to the configuration of the NuScale RVVs or the RRVs, where the valve is welded directly to a nozzle with no piping downstream of the valve. There are no thermal expansion loads due to an attached piping system and the displacement of the valve is less than that of a more massive and more flexible piping system during a seismic event. Also, the valve is supported only by the nozzle, eliminating the concern of seismic anchor movement loads.

Precedent supports NuScale's position. As stated in the AP1000 Design Control Document, Revision 19, Section 5.1.3.34, "The two reactor coolant pump suction nozzles are welded directly to the outlet nozzles on the bottom of the steam generator channel head." The NRC safety evaluation concluded that this nozzle-to-nozzle weld "eliminates the cross-over leg of the coolant loop piping…" (NUREG-1793, 5.1.3.3). NuScale did not find any indication that the welds between these two ASME Class 1 components were considered piping terminal ends as either potential high energy line break (GDC 4) or LOCA (GDC 35) locations. Rather, DCD Tables 3.6-2 and 3.6-3 indicate these welds are not locations of postulated pipe ruptures, and

Figure 3E-3 indicates that the welds are not considered high energy piping.

See NuScale's response to eRAI 8785, Question 15.06.05-1 for further justification of the NuScale break spectrum excluding the RVVs and RRVs.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0516-49422, Loss-of-Coolant Accident Evaluation Model, as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8776
Date of RAI Issue: 08/19/2017

NRC Question No.: 15.06.05-6

Title 10 of the Code of Federal Regulations (10 CFR) Part 52, Section 47 (a)(2) states, "The application must contain a Final Safety Analysis Report (FSAR) that… must include… a description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished." Regulatory Guide 1.203 describes a process that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

As stated in RG 1.203, an evaluation model (EM) is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, as illustrated by the following examples:

(1) procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation)
(2) specification of those portions of the analysis not included in the computer programs for which alternative approaches are used
(3) all other information needed to specify the calculational procedure

The entirety of an EM ultimately determines whether the results are in compliance with applicable regulations. Therefore, the development, assessment, and review processes must consider the entire EM.

During a LOCA, the high energy fluid released from the Reactor Pressure Vessel (RPV) largely condenses on the containment wall near the top of the Containment Vessel (CNV). The condensate will form a film and flow down the CNV wall. With the condensate film insulating the steam from the wall, less condensation is expected at the lower CNV elevations. In addition, temperature differences between the RPV outside surface and the CNV inside surface will cause heat transfer on both surfaces to be different in the radial direction. With the spatial

distribution of the RPV component structure and break location, eventually the steam inside the containment annulus will form a three-dimensional pattern. The current {{

}}[2(a),(c)] The staff is requesting the applicant to provide additional justification as to why a one- dimensional model adequately captures the 3-D effects of the fluid flow within containment.

Although the initial containment vacuum is maintained, a certain amount of the non-condensable gas is present and the dissolved gas in the primary system coolant may appear during the blow-down process. In the LOCA topical report, TR-0516-49422, Section 8.2.8.2, "Technical Evaluation," the applicant states, {{

}}[2(a),(c)] The staff is requesting the applicant to (1) provide additional information to support this statement and (2) evaluate the potential effect of dissolved reactor coolant system non-condensable gases on two-phase core water level and containment heat transfer upon emergency core cooling system actuation.

NuScale Response:

1.0 Background

{{

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

This response provides a conservative estimation of the total amount of non-condensable gas that is initially present inside the NPM containment and the reactor coolant system (RCS), including the amount of non-condensable gas dissolved in the RCS fluid. The results of analytical and NRELAP5 sensitivity calculations and the analysis of NIST-1 experimental data are discussed to show that during Phase 1a and 1b of the NPM LOCA:

---

1. {{

}}2(a),(c)

As described in Section 4.2 of Reference 1, Phase 1a of the LOCA spans the postulated break to ECCS actuation and Phase 1b spans the ECCS actuation to when recirculation flow is established and the pressures and levels in the RPV and CNV approach equilibrium.

2.0 Estimation of amount of non-condensable gas

The NPM CNV is maintained under a vacuum during normal operation. In addition to non-condensable gas that may be initially present in the CNV vessel during normal operation, non-condensable gas that is initially present in the RCS may be transported to the CNV during a LOCA and following ECCS actuation. The sources of non-condensable gas in the RCS include {{

}}2(a),(c)

- {{

}}2(a),(c)

Table 1 Mass of Non-Condensable Gas in CNV and RCS

{{

}}2(a),(c)

For the liquid space breaks (e.g., chemical and volume control system (CVCS) charging line break), only the fraction of RCS {{

}}2(a),(c)

{{

}}2(a),(c)

{{

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

Figure 1 Mole fraction of non-condensable gas as a function of containment pressure

3.0 Relative Importance of CNV Heat Transfer Mechanisms

3.1 Analytical calculations

The analytical calculations presented in this section were performed as part of NIST-1 facility scaling analysis. Figure 2 schematically shows the heat transfer path from the CNV to ultimate heat sink. The steam vented from the RCS condenses rapidly on the inside surface of the CNV. CNV heat is transferred to the CNV inside surface, conducted though the CNV wall to the outside surface of the CNV, and then transferred by natural convection to the reactor pool.

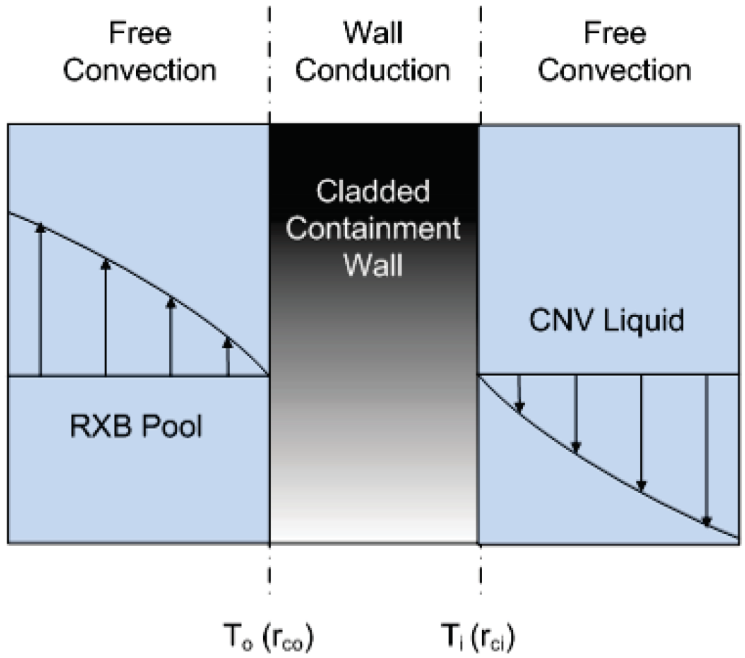

Figure 2 Containment Heat Transfer Mechanisms during a LOCA

Under steady state conditions, the rate of heat transfer from CNV to pool is represented by the following general equation:

{{

}}2(a),(c)

{{

}}2(a),(c

{{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

Figure 3 Containment Overall Heat Transfer Coefficient for Steady-State Conditions as a function of Containment Pressure

The above analysis is applicable for steady state heat transfer from the CNV to the reactor pool. The time constant for the {{

}}2(a),(c)

3.2 NRELAP5 Sensitivity Calculations

Figure 4 shows the percentage contribution of total heat transfer resistance from various CNV heat transfer mechanisms for a transient initiated by an inadvertent opening of a reactor vent valve (RVV). Similar results for a CVCS discharge pipe break LOCA are shown in Figure 5. The contribution of {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 4 CNV heat transfer resistance contribution in NPM NRELAP5 calculation for a transient initiated by inadvertent opening of a RVV

{{

}}2(a),(c)

Figure 5 CNV heat transfer resistance contribution in NPM NRELAP5 calculation for a typical CVCS discharge pipe LOCA

{{

}}2(a),(c)

Figure 6 CNV inside and outside heat transfer coefficients in NPM NRELAP5 calculation for a typical transient initiated by inadvertent opening of a RVV

Sensitivity calculations for an inadvertent opening of an RVV evaluate the impact of {{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

Figure 7 CNV heat transfer resistance contribution in NPM NRELAP5 calculation for transient initiated by inadvertent opening of a RVV with {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 8 CNV heat transfer resistance contribution in NPM NRELAP5 calculation for transient initiated by inadvertent opening of a RVV with {{

}}2(a),(c)

{{

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

Figure 9 Impact of reduced {{

}}[2(a),(c)] in NPM NRELAP5 calculation for transient initiated by inadvertent opening of an RVV

Similar sensitivity calculations are performed for the limiting peak CNV pressure scenario. The limiting peak CNV pressure scenario identified in Reference 5 is an inadvertent RRV opening case that assumes: loss of AC and DC power, minimum initial RCS flow, no single failure, no DHRS, and inadvertent actuation valve 1000 psid release pressure.

The convective heat transfer coefficient at the CNV inside surface is reduced by a range of multipliers (or fouling factors) in the NRELAP 5 heat structure model input. Table 2 shows the results of these sensitivity calculations. Note that all calculations are performed with the event initiation occurring at 50 seconds.

Table 2 Results of CNV inside surface heat transfer coefficient sensitivity calculations

{{

}}2(a),(c)

Table 2 shows that the peak CNV pressure increases with the degradation in {{

}}2(a),(c)

4.0 Adequacy of 1-D NRELAP5 Model to Calculate CNV Condensation

The adequacy of NRELAP5 to model the condensation in NPM CNV is shown in Reference 1 by assessment of NRELAP5 against the NIST-1 HP-02 separate effects condensation test data (see Section 7.5.4 of Reference 1). This section describes the adequacy of NIST-1 to represent the condensation phenomenon in the NPM CNV and provides further analysis of the NIST-1

HP-02 separate effects condensation test data.

4.1 Steam Condensation in NPM and NIST-1 CNV

The heat transfer at the NPM CNV inside wall surface is due to {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 10 Condensation heat transfer coefficients at inside surface of CNV, as a function of CNV pressure, calculated using {{ }}2(a),(c)

4.2 NRELAP5 Model for Steam Condensation and Assessment against Data

As described in Section 8.2.8 of Reference 1, the {{

}}2(a),(c)

{{

}}2(a),(c)

5.0 Effect of non-condensable gas on two-phase level in RPV

This request for additional information requests an evaluation of the potential effect of dissolved RCS non-condensable gases on two-phase core water level. The collapsed liquid level above the top of active fuel (TAF) is used as a FOM in the LOCA EM, while the ECCS actuation is based on two-phase (or mixture) level in RCS and CNV. All ECCS valves are equipped with an inadvertent actuation block (IAB) design feature that mechanically prevents spurious opening of the ECCS valves at full operating pressure. The IAB prevents the valves from opening when the differential pressure between the RPV and CNV is greater than the IAB threshold pressure setpoint. After the IAB has blocked a spurious opening of the ECCS valve, it allows the valve to open only after the differential pressure between the RPV and CNV has decreased below the IAB release pressure setpoint.

The NRELAP5 sensitivity calculations presented in this response indicate that any {{

}}2(a),(c)

{{

}}2(a),(c)

6.0 Summary

The presence of non-condensable gas has a degrading effect on the condensation rate of steam. As the condensation progresses, non-condensable gases from the gas mixture tend to concentrate near the condensate film surface. This results in reduction of the partial pressure of steam near the film surface and lower condensation rate.

The initial amount of non-condensable gas in the NPM CNV is small as the NPM implements a vacuum during normal operation. Considering this and other sources of non-condensable gas initially present in the RCS, as shown in Section 2 of this response, the concentration of non-condensable gas in the NPM CNV decreases rapidly relative to the concentration of steam as the CNV pressure increases to its peak value following a LOCA and/or actuation of ECCS.

The analytical calculation results as well as NRELAP5 sensitivity calculations show that the {{

}}2(a),(c)

Furthermore, due to the finite time constant for the heat transfer process from the CNV to the reactor pool, the instantaneous CNV pressure response under transient conditions after opening of ECCS/break is primarily a function of the total energy release rate to the CNV from the RCS and the free CNV volume. NRELAP5 sensitivity calculations show that even a {{

}}2(a),(c)

The NRELAP5 model used for the calculation of film condensation heat transfer coefficient for the NPM and NIST CNV is conservative because it uses a {{

}}2(a),(c)

Considering this conservative approach for calculation of heat transfer coefficient, the results of sensitivity calculations, the assessment of NRELAP5 against the NIST-1 IETs and high pressure condensation SET, and the fact that the concentration of non-condensable decreases significantly as the CNV pressurizes, the current 1-D modeling approach for NPM CNV is adequate.

The evaluation of impact of non-condensable gas or a 3-D flow distribution in the CNV on the collapsed liquid level in the RPV shows that any degradation of heat transfer from the CNV to pool would result in increase of the minimum collapsed liquid level in RPV. Furthermore, two-phase level swell in the RPV level due to release of dissolved gas would not impact the progression of limiting small liquid-space break LOCA scenario.

References

1. TR-0516-49422, Revision 0, "Loss-of-Coolant Accident Evaluation Model."
2. Butterworth, D., 1983, Film Condensation of a Pure Vapour, In Heat Exchanger Design Handbook, edited by E. U. Schlunder, Hemisphere Publishing Company, New York, NY, 1989.
3. Churchill, S. and Chu H, “Correlating Equations for laminar and turbulent free convection from a vertical plate,” International Journal of Heat Mass Transfer, Vol. 18(11), pp. 1323-1329, 2009.
4. Minkowycz, W.J. and Sparrow, E.M., “Local Nonsimilar Solutions for Natural Convection on a Vertical Cylinder,” Journal of Heat Transfer, 96(2), 178-183, May 1974.
5. TR-0516-49084, Revision 0, "CNV Response Analysis Methodology Technical Report."
6. Shah, Mohammed M., “An Improved and Extended General Correlation for Heat Transfer During Condensation in Plain Tubes”, HVAC&R Research, Vol. 15, No. 5, September 2009.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0516-49422, Loss-of-Coolant Accident Evaluation Model, as a result of this response.

Enclosure 3:

Affidavit of Zackary W. Rad, AF-1017-56706

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

1. I am the Director, 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 methods by which NuScale develops its loss-of-coolant accident analysis of the NuScale power module.

   NuScale has performed significant research and evaluation to develop a basis for this methods 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. 8776, eRAI No. 8776. 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 10/18/2017.

Zackary W. Rad