NuScale Power

RAIO-0717-55061

August 1, 2017

Docket No. 52-048

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

**SUBJECT:** NuScale Power, LLC Response to NRC Request for Additional Information No. 50 (eRAI No. 8851) on the NuScale Design Certification Application

**REFERENCE:** U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 50 (eRAI No. 8851)," dated June 02, 2017

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. 8851:

- 04.02-2
- 04.02-3
- 04.02-4
- 04.02-5

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 50 (eRAI No. 8851). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The proprietary enclosure has 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
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8851, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8851, nonproprietary

Enclosure 3: AREVA Affidavit of Nathan E. Hottle

**Enclosure 1:**

NuScale Response to NRC Request for Additional Information eRAI No. 8851, proprietary

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

**Enclosure 2:**

NuScale Response to NRC Request for Additional Information eRAI No. 8851, nonproprietary

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

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

**eRAI No.:** 8851
**Date of RAI Issue:** 06/02/2017

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**NRC Question No.:** 04.02-2

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, Criterion 2, requires that SSCs important to safety are designed to withstand the effects of earthquakes without the loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) the severity of the historical reports, with sufficient margin to cover the limited accuracy, quantity, and time period for the accumulated data, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed. SRP Section 4.2 Appendix A (IV) provides review guidance regarding the review of acceptance criteria used to analyze the fuel structural response to externally applied loads.

Technical Report TR-0816-51127-P Revision 1 provides the fuel and control rod assembly design analysis for the NuFuel-HTP2 fuel assembly. Table 4-8 lists evaluation margins for various fuel assembly components. The staff noted that the allowable stress limits identified are not consistent with ASME NG-3224.1(a). The primary membrane stress limit should be 1.5 $S_m$ and the primary membrane plus bending stress limit should be 2.25 $S_m$. The guide tube $S_m$ implied by Table 4-1 is also not consistent with the stress limits listed in Table 4-8.

In order to make an affirmative finding associated with the above regulatory requirement that accounts for all relevant SSCs important to safety, the NRC staff requests the following information to be provided: Revise the table(s) as necessary to report the correct stress margins and allowable limits. If Table 4-8 of TR-0816-51127-P Revision 1 is correct as is, provide further explanation of how the structural analyses of the guide tube relate to ASME Subsection NG.

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**NuScale Response:**

TR-0716-50351-P, Revision 0 (Reference 2) demonstrates the applicability of the methodology in ANP-10337P, Revision 0 (Reference 3) to NuScale fuel. The guide tube allowable stress limits reported in Table 4-8 of TR-0816-51127-P, Revision 1 (Reference 1) are based on the limits specified in Reference 3.

Section 4.2.2.2 and Section 4.3.2 of Reference 3 specify that the allowable stress limits for guide tubes to ensure control rod insertion under Safe Shutdown Earthquake (SSE) and Loss of Coolant Accident (LOCA) events shall be defined in accordance with the Level C criteria per section NG-3224.1(a) of Section III of the ASME Code. Per Section NG-3224.1(a) (Reference 4) the following stress intensity limits shall apply:

**Primary Membrane Stress: $P_m \leq 1.5 \cdot S_m$**
**Primary Membrane plus Bending Stress: $P_{m+b} \leq 2.25 \cdot S_m$**
**Where $S_m$ is the allowable stress intensity.**

Section F.2 of Appendix F of Reference 3 provides a discussion of these limits and presents the primary membrane plus bending stress limit in a more conservative form, per Section NB-3221.3 of Section III of the ASME Code (Reference 4), as:

**Primary Membrane plus Bending Stress: $P_{m+b} \leq K \cdot 1.5 \cdot S_m$**
**Where K is the plastic shape factor.**

In the case of a structure with a rectangular cross-section, the plastic shape factor is 1.5. Substitution of this value into the equation above yields the relationship specified in Section NG-3224.1(a) of Section III of the ASME code.

For an annular cross-section, the plastic shape factor is less than 1.5 and is defined based on Equation F-1 in Reference 3. Applying the dimensions of the NuScale guide tube, this shape factor becomes 1.32 and the resulting stress limits for NuScale become:

**Primary Membrane Stress: $P_m \leq 1.5 \cdot S_m$**
**Primary Membrane plus Bending Stress: $P_{m+b} \leq 1.32 \cdot 1.5 \cdot S_m = 1.98 \cdot S_m$**

As can be seen, the stress limits for the NuScale guide tube are more conservative than what is strictly specified in Section NG-3224.1(a) of the ASME Code.

Note that the plastic shape factor becomes 1.39 in the dashpot region of the guide tube where the tube wall has a greater thickness.

Per Section II, Part D of the ASME Code (Reference 4), the allowable stress intensity, $S_m$, is defined as the minimum of either 1/3 of the ultimate tensile strength or 2/3 of the yield strength. For the annealed Zirc-4 tubing used for NuScale guide tubes, this limit is set by the ultimate strength at operating temperatures to a value of $S_m$ = **[ ]**.

Thus, the following stress limits are derived:

**Primary Membrane Stress: $P_m \leq 1.5 \cdot S_m$ = [ ]**
**Primary Membrane plus Bending Stress: $P_{m+b} \leq 1.98 \cdot S_m$ = [ ]**

These values are consistent with the allowable stress limits reported in Table 4-8 of Reference 1.

In summary, the stress limits used for the NuScale fuel are consistent with the NRC approved methodology in Reference 3 and are more conservative than those specified in the ASME code.

References:


1. NuScale, LLC, "NuFuel-HTP2™Fuel and Control Rod Assembly Designs," TR-0816-51127-P, Revision 1, January 2017.
2. NuScale, LLC, "NuScale Applicability of AREVA Methods for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351-P, Revision 0, September 2016.
3. AREVA NP, Inc., "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," ANP-10337P, Revision 0, August 2015.
4. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, "Rules for Construction of Nuclear Power Plant Components," 2009 Revision with Addenda, New York.


**Impact on DCA:**
There are no impacts to the DCA as a result of this response.

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

**eRAI No.:** 8851
**Date of RAI Issue:** 06/02/2017

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**NRC Question No.:** 04.02-3

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, Criterion 2, requires that SSCs important to safety are designed to withstand the effects of earthquakes without the loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) the severity of the historical reports, with sufficient margin to cover the limited accuracy, quantity, and time period for the accumulated data, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed. SRP Section 4.2 Appendix A (III) provides review guidance regarding the determination of strength.

Technical Report TR-0816-51127-P Revision 1 provides the fuel and control rod assembly design analysis for the NuFuel-HTP2 fuel assembly. Table 4-8 lists evaluation margins for various fuel assembly components. The staff notes that the guide tube buckling limit is expressed as a pressure in Table 4-8, but units of compressive force (N or $lb_f$) are expected based on the context.

In order to make an affirmative finding associated with the above regulatory requirement that accounts for all relevant SSCs important to safety, the NRC staff requests the following information to be provided:

a. If the units are in error, please correct them
b. If the buckling analysis is really based on compressive pressure loading, provide additional details of the loading conditions and describe the analysis method.

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**NuScale Response:**

a. The units (MPa) for the guide tube buckling limit in Table 4-8 of TR-0816-51127-P (Reference1) are correct.

b. TR-0716-50351-P, Revision 0 (Reference 2) demonstrates the applicability of the methodology in ANP-10337P, Revision 0 (Reference 3) to NuScale fuel. The guide tube buckling analysis summarized in Table 4-8 of Reference 1 is based on the method specified in Section 8.2.1 of Reference 3.

Per Section 8.2.1 of Reference 3, **[**

**]**.

References:


1. NuScale, LLC, "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," TR-0816-51127-P, Revision 1, January 2017.
2. NuScale, LLC, "NuScale Applicability of AREVA Methods for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351-P, Revision 0, September 2016.
3. AREVA NP, Inc., "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," ANP-10337P, Revision 0, August 2015.


**Impact on DCA:**
There are no impacts to the DCA as a result of this response.

NuScale Power™

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

**eRAI No.:** 8851
**Date of RAI Issue:** 06/02/2017

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**NRC Question No.:** 04.02-4

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, Criterion 2, requires that SSCs important to safety are designed to withstand the effects of earthquakes without the loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) the severity of the historical reports, with sufficient margin to cover the limited accuracy, quantity, and time period for the accumulated data, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed. SRP Section 4.2 Appendix A (III) provides review guidance regarding the determination of strength.

Section 4.1.1.1 of TR-0816-51127-P Rev 1 defines a criterion to be used to evaluate fuel rod cladding stress and buckling during the limiting overpressure transient at BOL on page 31. However, it is stated on page 32 that a different (Euler) buckling criterion is met.

In order to make an affirmative finding associated with the above regulatory requirement that accounts for all relevant SSCs important to safety, the NRC staff requests NuScale to provide clarification regarding the criterion used to evaluate fuel rod buckling and update the documentation if necessary.

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**NuScale Response:**

TR-0116-20825-P, Revision 1 (Reference 2) demonstrates the applicability of the fuel rod cladding stress and buckling analysis methodology defined in BAW-10227P-A, Revision 1 (Reference 3) to NuScale fuel. The fuel rod cladding stress and buckling analysis summarized in Section 4.1.1.1 of TR-0816-51127-P, Revision 1 (Reference 1) is based on the method specified in Section 3.3 of Reference 3.

As explained in Section 3.3 of Reference 3, the **[** **]** is used to assess the capability of the fuel rod cladding to resist buckling. This method involves **[**

]. The text in the paragraph immediately following Table 4-2 of Reference 1 explains that the design criterion is met because the differential pressure across the cladding during the limiting overpressure transient at BOL is less than the buckling pressure.

The Euler criterion, although not part of the methodology defined in Reference 3, was also evaluated as an additional buckling check. The evaluation confirms the fuel rod's ability to withstand axial compression loads due to differential thermal expansion between fuel rods and guide tubes.

References:


1. NuScale, LLC, "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," TR-0816-51127-P, Revision 1, January 2017.
2. NuScale, LLC, "Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-P, Revision 1, July 2016.
3. AREVA NP, Inc., "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," BAW-10227P-A, Revision 1, June 2003.


**Impact on DCA:**
There are no impacts to the DCA as a result of this response.

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

**eRAI No.:** 8851
**Date of RAI Issue:** 06/02/2017

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**NRC Question No.:** 04.02-5

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, Criterion 2, requires that SSCs important to safety are designed to withstand the effects of earthquakes without the loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) the severity of the historical reports, with sufficient margin to cover the limited accuracy, quantity, and time period for the accumulated data, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed. SRP Section 4.2 Appendix A (III) provides review guidance regarding the determination of strength.

While the NuFuel-HTP2 fuel design is similar to existing fuel designs used by some currently operating power plants, there are differences which could significantly affect the fuel assembly structural response to externally applied loads. It is unclear to the staff how sensitive the analysis is to these differences.

In order to make an affirmative finding associated with the above regulatory requirement that accounts for all relevant SSCs important to safety, the NRC staff requests NuScale to provide the core plate time histories used to evaluate the fuel assembly structural response to externally applied loads.

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**NuScale Response:**

Summary of Transmitted Time Histories

The time histories defined in Table 1 were made available for audit by the NRC staff on 7/31/2017. The selection of these time histories is based on their correlation to particular limiting margins that are of interest for the spacer grids, bottom nozzle, and the guide tube.

**Table 1: Time Histories**

| No. | Time History | Files |
|---|---|---|
| 1 | CSDRS_TS1_CSD_S7_CR_RXM01 | Lower Core Plate:<br>*DIS_ACC_NID16_C2_CAP_S7_CR_RXM01_TS1*<br>Upper Core Plate:<br>*DIS_ACC_NID17_C2_CAP_S7_CR_RXM01_TS1* |
| 2 | LOCA_2RRV | Lower Core Plate:<br>*DIS_ACC_NID16_2RRV*<br>Upper Core Plate:<br>*DIS_ACC_NID17_2RRV* |
| 3 | CSD_S7_UC_RXM01_TS3 | Lower Core Plate:<br>*DIS_ACC_NID16_C2_CAP_S7_UC_RXM01_TS3*<br>Upper Core Plate:<br>*DIS_ACC_NID17_C2_CAP_S7_UC_RXM01_TS3* |
| 4 | GHF_S7_CR_RXM01_TS2 | Lower Core Plate:<br>*DIS_ACC_NID16_GH_LCN_S7_CR_RXM01_TS2*<br>Upper Core Plate:<br>*DIS_ACC_NID17_GH_LCN_S7_CR_RXM01_TS2* |
| 5 | LOCA_2RVV | Lower Core Plate:<br>*DIS_ACC_NID16_2RVV*<br>Upper Core Plate:<br>*DIS_ACC_NID17_2RVV* |

Sensitivity Studies Already Performed in the NuScale Analysis

The seismic and LOCA analyses for the NuScale fuel have already performed the recommended sensitivity study as defined in Section II.3 of Appendix A of the Standard Review Plan Chapter 4.2. This sensitivity study is required per the applied methodology defined in Section 7.5 of ANP-10337P (Reference 2). The sensitivity studies are only performed for the most limiting cases. For the lateral seismic evaluation, a total of three EOL cases were evaluated for sensitivity. For the vertical seismic evaluation, a total of five BOL cases were evaluated for sensitivity. Sensitivity studies are not performed for cases that will not challenge load limits (e.g. LOCA). As can be seen in the information provided below, some NuScale loads were increased by the application of a sensitivity factor, in instances where the sensitivity analyses warranted it. This sensitivity factor was developed and applied consistent with the definition provided in Reference 2 and SRP 4.2, Appendix A.

Details Regarding Component Margins, Sensitivity Studies, and Corresponding Time Histories

*Limiting Grid Impact Loads:*

From Table 4-7 in Reference 1, the limiting grid impact load occurs in **[** **]**. Table 2 provides more detail regarding the basis for this result including the corresponding time history causing the load and the results of sensitivity studies performed on this case.

**Table 2: Limiting Grid Impact Load, Load Components, Corresponding Time Histories, and Sensitivity Study Results**

| | | |
|---|---|---|
| LOCA | Maximum Impact Load (N) | [ ] |
| | Case Description | [ ] |
| | Time History | [ ] |
| | Sensitivity, Variation in Impact Load | [ ] |
| | Sensitivity Description | [ ] |
| | Sensitivity Factor | [ ] |
| | Maximum Impact Load with Factor (N) | [ ] |
| SSE | Maximum Impact Load (N) | [ ] |
| | Case Description | [ ] |
| | Time History | [ ] |
| | Sensitivity, Variation in Impact Load | [ ] |
| | Sensitivity Description | [ ] |
| | Sensitivity Factor | [ ] |
| | Maximum Impact Load with Factor (N) | [ ] |
| SRSS SSE+ LOCA (N) | | [ ] |
| Grid Load Limit (N) | | [ ] |
| Margin (%) | | [ ] |

[ ]

*Limiting Bottom Nozzle Loads:*

Although the load margins on the bottom nozzle are not limiting, this case is provided because it provides the most direct baseline result for the vertical model.

From Table 4-8 in Reference 1, the limiting bottom nozzle impact load occurs [ ]. The vertical simulation is performed with steady state hydraulic lift loads present on the assembly. When seismic and LOCA results are combined, these steady state loads must be removed from the individual components and then added back to the combined load. This operation is shown in the following equation.

$$V_{combined} = V_{SS} + \sqrt{(V_{SSE} - V_{SS})^2 + (V_{LOCA} - V_{SS})^2}$$

Where: $V_{SS}$ is the steady-state vertical load
$V_{SSE}$ is the peak vertical load under SSE
$V_{LOCA}$ is the peak vertical load under LOCA

The components from the above equation are provided in Table 3 along with the corresponding time history causing the load and the results of sensitivity studies performed on this case.

**Table 3: Limiting Bottom Nozzle Impact Load, Load Components, Corresponding Time Histories, and Sensitivity Study Results**

| | | |
|---|---|---|
| Steady State | Compressive Load (N) | [ ] |
| LOCA | Maximum Impact Load (N) | [ ] |
| | Time History | [ ] |
| | Sensitivity, Variation in Impact Load | [ ] |
| | Sensitivity Description | [ ] |
| | Sensitivity Factor | [ ] |
| | Maximum Impact Load with Factor (N) | [ ] |
| SSE | Maximum Impact Load (N) | [ ] |
| | Time History | [ ] |
| | Sensitivity, Variation in Impact Load | [ ] |
| | Sensitivity Description | [ ] |
| | Sensitivity Factor | [ ] |
| | Maximum Impact Load with Factor (N) | [ ] |
| Combined Vertical Load (N) | [ ] | |
| Load Limit (N) | [ ] | |
| Margin (%) | [ ] | |

[ ]

*Guide Tube Stress:*

From Table 4-8 in Reference 1, the limiting guide tube stress margin occurs [ ]. This limiting guide tube

stress occurs in the span [ ].
This limiting stress state is the result of the worst-case combination of the SSE and LOCA events simulated in lateral and vertical directions. The events that define this worst-case combination are specified in Table 4 below. Note that in the case of the seismic components, the lateral and vertical components of the guide tube stress [ ].

**Table 4: Corresponding Time Histories by Stress Component for the Limiting Guide Tube Stress Margin (Primary Membrane plus Bending)**

| Component | Time History |
|---|---|
| Horizontal SSE | [ ] |
| Horizontal LOCA | [ ] |
| Vertical SSE | [ ] |
| Vertical LOCA | [ ] |

[ ]

The development of the inputs to the seismic analysis is described in the "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs" Technical Report, TR-0816-51127P, Section 4.3.5.1.3. The spectra from the various soil/rock profiles, concrete conditions, module configurations, and resulting time histories, and the assumption of negligible vertical LOCA hydraulic forces, establish the basis (Base Case) for the fuel design and resulting component stress margins reported in Table 4-8 of the Technical Report.

Changes to the seismic analysis inputs and LOCA loads were made by NuScale subsequent to the analyses presented in this Fuel Technical Report. The seismic time histories included in the "NuScale Power Module Seismic Analysis" Technical Report, TR-0916-51502, were developed after the Base Case was evaluated for the fuel design and component margins. Similarly, vertical LOCA hydraulic loads (originally assumed to be negligible) were revised after the final margins for the Base Case were derived. These new seismic and LOCA inputs were evaluated against the Base Case. The analysis demonstrated that the Base Case remained bounding.

To document the process used to evaluate updates to seismic inputs against a Base Case, TR-0816-51127P has been revised as shown in the attached mark-up.

References:


1. NuScale, LLC, "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," TR-0816-51127-P, Revision 1, January 2017.
2. AREVA NP, Inc., "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," ANP-10337P, Revision 0, August 2015.

**Impact on DCA:**

Technical Report TR-0816-51127, NuFuel-HTP2™ Fuel and Control Rod Assembly Designs, has been revised as described in the response above and as shown in the markup provided in this response.

The fuel rod-to-nozzle gap element is defined using dynamic drop test results performed on the EOL fuel assembly. In this test, **[**

**]**

**[**

**]**

#### 4.3.5.1.3 Excitation Inputs

The excitation inputs for the external load analysis are NuScale Power Module (NPM) core plate displacement time histories for the SSE and LOCA events. The core plate displacement time histories include both horizontal and vertical motions.

The SSE input motions are the result of an evaluation of multiple, independent sets of soil-structure interaction parameters and module locations. ~~For the certified~~The seismic ~~design~~ response spectra~~um,~~ consider~~ing~~ six soil/rock profiles, cracked and uncracked concrete conditions, and two bounding reactor module configurations, for a total of 24 variations~~ are examined~~. For the generic high frequency hard rock response spectrum, considering two soil/rock profiles, cracked and uncracked concrete conditions, and two bounding reactor module configurations, a total of eight variations are examined.

To account for the effect of uncertainty in the reactor module dynamic analysis, each time history variation is analyzed with three different scaled time intervals: the reference interval and plus or minus 15%. The frequency shift due to the 15% variation of the time scale is considered to be effectively equivalent to the broadening of spectral peaks that is done when generating in-structure response spectra. Considering the defined variations, a total of 96 time histories are considered in the analysis.

The LOCA time histories are derived from bounding high energy line breaks in the primary coolant system and inadvertent or spurious operation of reactor coolant pressure boundary valves. ~~The development of the short term transient dynamic loads is described in the NuScale Power Module Short Term Transient technical report (Reference 9.1.15).~~ Core plate motions are the combined dynamic response due to asymmetric cavity pressurization of the containment, depressurization of the reactor pressure vessel, and thrust force at the break or valve location. ~~The~~ The analysis is based on the assumption that the LOCA events for the NuScale design result in negligible vertical hydraulic forces acting on the reactor internals.~~ While Reference 9.1.5. discusses the presence of a vertical hydraulic forcing function acting on the fuel in the vertical LOCA analysis, this hydraulic excitation component is not present for the NuScale design because of the negligible vertical forces.~~

The spectrum of soil/rock profiles, concrete conditions, reactor configurations and resulting input time histories, and the assumption of negligible vertical hydraulic forces, described above establish the basis (Base Case) for the fuel design and resulting component stress margins. Subsequent changes to these seismic and LOCA inputs are evaluated against this ~~b~~Base ~~c~~Case to ensure the analysis remains bounding. The seismic inputs and methodology addressed~~included~~ in the NuScale Power Module Seismic Analysis technical report (Reference 9.1.14), and the vertical hydraulic loads discussed in the NuScale Power Module Short Term Transient technical report (Reference 9.1.15), are evaluated relative to the ~~b~~Base ~~c~~Case; the evaluation demonstrates that the fuel loads resulting from the spectra in Reference 9.1.14 in combination with the vertical LOCA loads in Reference 9.1.15 remain bounded.

#### 4.3.5.2 Analysis Results

##### 4.3.5.2.1 Lateral Analysis

The horizontal excitation of the full reactor core is considered in the analysis through a series of two dimensional row models with lengths of three, five, and seven fuel assemblies. (Refer to the reactor core configuration in Figure 3-2.) Excitations in both horizontal directions are considered.

The peak impact loads for the base case seismic inputs, along with margin to the grid impact load limit, from all cases are summarized in Table 4-7. The peak impact loads for SSE and LOCA in a given direction are combined by the square root of the sum of the squares (SRSS) method and margin is calculated against this SRSS impact load. The positive margin for these impact loads confirms that the NuScale spacer grid will not experience plastic deformation that exceeds the limit established in the AREVA methodology (Reference 9.1.5). Thus, the requirements for core coolability and control rod insertion are met.

Table 4-7 Peak grid impact loads and margins

[ ]

##### 4.3.5.2.2 Vertical Analysis

The single assembly vertical model is subjected to vertical core plate displacement time histories corresponding to the SSE and LOCA events. The maximum seismic impact load for the base case inputs, **[** **]** The maximum LOCA impact load is **[** **]** Component loads for the guide tubes, fuel rods, hold-down spring, nozzles, and guide tube connections are extracted from the vertical analysis for further load analysis.

**Enclosure 3:**

AREVA Affidavit of Nathan E. Hottle

# AFFIDAVIT

COMMONWEALTH OF VIRGINIA )
) ss.
CITY OF LYNCHBURG )

1. My name is Nathan E. Hottle. I am Manager, Product Licensing, for AREVA Inc. (AREVA) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in the following document: "NuScale Power, LLC Response to NRC Request for Additional Information No. 50 (eRAI No. 8851) on Technical Report TR-0816-51127, 'NuFuel-HTP2™ Fuel and Control Rod Assembly Designs'," referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(c) and 6(d) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 20th

day of July, 2017.

Ella F. Carr-Payne
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 08/31/17
Reg. # 309873

ELLA F. CARR-PAYNE
NOTARY PUBLIC
REG # 309873
MY COMMISSION EXPIRES 8/31/2017
COMMONWEALTH OF VIRGINIA