NUSCALE POWER™

RAIO-0517-54299

June 8, 2017

Docket No. 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. 13 (eRAI No. 8736) on Topical Report TR-0716- 50351, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces"

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 13 (eRAI 8736)," dated April 10, 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 responses to the following RAI questions from NRC eRAI No. 8736:

- 29611
- 29613
- 29614
- 29615
- 29616

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 13 (eRAI No. 8736). The responses to questions 29611, 29613, 29614, and 29616 contain material considered proprietary by AREVA. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, 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, TWFN-6E55
Samuel Lee, NRC, TWFN-6C20
Bruce Bavol, NRC, TWFN-6C20

Enclosure 1: NuScale Responses to NRC Request for Additional Information eRAI No. 8736 RAI 29611, 29613, 29614, 29615, and 29616 (proprietary)

Enclosure 2: NuScale Responses to NRC Request for Additional Information eRAI No. 8736 RAI 29611, 29613, 29614, 29615, and 29616 (nonproprietary)

Enclosure 3: AREVA Affidavit

Enclosure 1:

NuScale Responses to NRC Request for Additional Information eRAI No. 8736 RAI 29611, 29613, 29614, 29615, and 29616 (proprietary)

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

Enclosure 2:

NuScale Responses to NRC Request for Additional Information eRAI No. 8736 RAI 29611, 29613, 29614, 29615, and 29616 (nonproprietary)

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

Response to Request for Additional Information
Docket No. PROJ0769

eRAI No.: 8736
Date of RAI Issue: 04/10/2017

NRC Question No.: 29611

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, Criterion 2, requires that structures, systems, and components (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 (II)(2) provides review guidance regarding the review of methods used to analyze loads.

Topical Report TR-0716-50351 references topical report ANP-10337-P as the methodology used to obtain the transient response. The staff noted that there are differences between the NuScale development of the damping values presented in TR-0716-50351 Appendix B and Appendix C of ANP-10337-P. The staff is seeking clarification regarding the methodology used to analyze the NuScale fuel assembly design structural response to externally applied loads.

a. Provide a comparison to test data that confirms the viscous damping analytical methodology (the approach used to determine the NuScale zero flow damping values) is correct.

b. Compare the development of NuScale damping values (see Figure B.3-1) to the development of the general PWR damping values documented in ANP-10337 Appendix C (Figure C-2) and provide a discussion of the differences.

c. Explain why the ratio of the proposed BOL damping value (Table B.4-1 of TR-0716-50351-P[SJ2]) to the generic PWR damping value (Section 6.1.3.1 and Figure C-2 of ANP-10337P) is noticeably different than the ratio of the proposed EOL value compared with the generic PWR EOL damping value.

d. Confirm that the vertical axis of Figure B.3-1 is mislabeled and is really critical damping ratio with units of percent.

e. The horizontal axis of Figure B.3-1 is labeled as deflection. Describe the definition of “deflection” as used in this figure. If deflection in this figure is not equivalent to the amplitude of ANP-10337 Figure C-1 then provide the data in a form that is directly comparable to ANP-10337 Figure C-1.

NuScale Response:

The NuScale response is provided below:

a. The methodology that is used to derive NuScale viscous damping values is validated using the MALDIVE fuel assembly test results. The MALDIVE tests were performed with fuel assemblies with a length similar to NuScale. Further description of these tests is provided in Appendix B of Reference 1. The MALDIVE test results were specifically chosen for this validation exercise because they are the most relevant to validate the effect of fuel assembly length on damping characteristics. In this validation, damping ratio values obtained directly from testing of the MALDIVE fuel assembly are compared to damping ratio values predicted for MALDIVE, where the methodology is the same as that used to derive the NuScale damping values.

   The NuScale damping ratios are based on an approach in which **[** **]**. The MASSE and CAMEOL tests are described in Appendix B of Reference 1. As a validation of this approach, the same methodology was applied to the MASSE data to scale it to the MALDIVE configuration. The predicted values for MALDIVE were compared to direct measurements from the MALDIVE tests. Figure 1 shows a comparison between the predicted damping ratios of the MALDIVE fuel assembly and the damping ratio values extracted from the MALDIVE tests for in-water conditions at zero flow velocity.

   A demonstration of the process used to validate this methodology as shown in Figure 1 is described below. For illustration, this discussion will focus on the derivation of the predicted MALDIVE damping when **[** **]**. The small difference observed between measured and predicted values in Figure 1 validates the approach used to derive the NuScale damping ratios.

b. Figure B.3-1 of TR-0716-50351-P (Reference 1) shows the non-irradiated and simulated-irradiated structural damping ratios of the NuScale bundle as a function of the amplitude of deflection as measured **[**

].

The comparable data from ANP-10337P (Reference 2) is in Figure C-1. The data in Figure C-1 shows comparable results to Figure B.3-1 of TR-0716-50351-P (Reference 1). The primary difference between these two data sets is due to the length of the fuel bundles. The NuScale bundle is roughly half the length of the assemblies presented in Figure C-1. As a result, the effect of amplitude in the NuScale test data is more pronounced, as the degree of curvature imposed on a NuScale fuel assembly is greater than that of a longer fuel assembly at a given deflection amplitude.

The ANP-10337P (Reference 2) data sets in Figure C-1 agree well with the data sets presented in Figure B.3-1 of TR-0716-50351-P (Reference 1). The typical range of damping ratios in the non-irradiated condition are similar **[** **]**. Furthermore, the typical range of damping ratios in the simulated-irradiated condition are similar **[** **]**.

Figure C-2 in ANP-10337P (Reference 2), cannot be directly compared to Figure C-1 or Figure B.3-1 from TR-0716-50351-P (Reference 1). Whereas Figure C-1 from ANP-10337P and Figure B.3-1 from TR-0716-50351-P present data from **[** **]**, Figure C-2 from ANP-10337P is based on **[** **]**. Consequently, the definition of amplitude in Figure C-2 is unique from the other figures. Amplitude in Figure C-2 is based on **[**

]. Thus, the data presented in Figure C-2 is more comparable to the damping ratios presented at the low range of amplitudes in Figure C-1.

Note that Figure C-2 in ANP-10337P (Reference 2) only presented non-irradiated data. The equivalent figure for simulated-irradiated conditions is Figure C-5. In Figure C-5, **[**

]. This response is consistent with both the data from Figure C-1 (ANP-10337P) and the NuScale data in Figure B.3-1 (TR-0716-50351-P) (Reference 1).

c. The ratio of the NuScale BOL damping value for the first mode (Table B.4-1 of TR-0716-50351) (Reference 1) to the generic PWR damping value (Section 6.1.3.1 of ANP-10337P) (Reference 2) is **[** **]**. The ratio of the NuScale EOL damping value for the first mode (Table B.4-1 of TR-0716-50351) to the generic PWR damping value (Section 6.1.3.1 of ANP-10337P) is **[** **]**.

In both cases, non-irradiated and irradiated, the NuScale damping values are less than what is presented in ANP-10337P (Reference 2), since the NuScale damping values do not credit the benefits of flow.

The ratio of the non-irradiated damping values **[ ]** indicates good agreement between NuScale and ANP-10337P (Reference 2).

Unlike the ratio of the non-irradiated damping values, the ratio of the irradiated damping values is **[ ]**. The explanation for this **[ ]** is twofold. **[**

].

The discussion above focuses on a comparison of first mode damping values. In the case of the 3rd mode damping, the NuScale value **[**

].

d. The vertical axis of Figure B.3-1 of TR-0716-50351 (Reference 1) is mislabeled. The correct label for this axis should be “First Mode Damping Ratio” with units of percent, which is equivalent to “critical damping ratio” as suggested in the RAI.

e. The horizontal axis of Figure B.3-1 of TR-0716-50351 (Reference 1) is labeled as “Deflection” while the horizontal axis of Figure C-1 of ANP-10337P (Reference 2) is labeled as “Amplitude”. The horizontal axes of both of these plots are equivalent. The definition of “Deflection” and “Amplitude” in both cases is the same and both represent the **[**
]. The data in Figure B. 3-1 of TR-0716-50351 (Reference 1) can be directly compared to the data in Figure C-1 of ANP-10337P (Reference 2).

Impact on Topical Report:

The vertical axis of Figure B.3-1 will be revised to be labeled “First Mode Damping Ratio (%)” as shown in the markup.

References:

1. TR-0716-50351-P, Rev. 0, NuScale Applicability of AREVA Method for Evaluation of Fuel Assembly Structural Response to Externally Applied Forces

2. ANP-10337P, Rev. 0, PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations

Figure 1: Comparison between predicted and experimentally derived damping ratios for the MALDIVE fuel assembly in-water at zero flow conditions (at 70F)

[ ]

Figure B.3-1. Experimental damping ratio values for mode 1 for the non-irradiated and simulated-irradiated NuScale fuel assembly (in-air, free vibration tests)

[ ]

[ ]

[ ]

NuScale Power

Response to Request for Additional Information
Docket No. PROJ0769

eRAI No.: 8736
Date of RAI Issue: 04/10/2017

NRC Question No.: 29613

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 (II)(1) provides review guidance regarding the review of input loads including core plate motions.

Section 3.1 of TR-0716-50351-P provides an analysis of the applicability of the referenced methodology topical report ANP-10337P. Although reactor geometry is discussed, there does not appear to be a comparison provided which discusses the anticipated core plate excitations between the NuScale design and typical PWR designs.

Compare the anticipated core plate excitation of the NuScale to more standard full length PWR excitation. This comparison should include:

a. Frequency spectra of core plate motion associated with design basis seismic load cases for the NuScale and typical PWR reactors.

b. Discussion of the anticipated magnitudes of excitation and justification of the use of the methodology in ANP-10337P for NuScale given the differences in frequency spectra and magnitudes.

NuScale Response:

The NuScale response is provided below:

a. Figures 1 and 2 below present an envelope of the frequency spectra of the upper core plate horizontal motion for NuScale and a typical PWR reactor (B&W). Although the upper core plate motions are used here for illustration, the lower core plate motions exhibit a similar response.

b. A comparison of Figure 1 and Figure 2 shows two distinct differences between NuScale and typical PWRs. Note that the response spectra for the B&W plant is typical for other PWRs.

   The first difference is that the NuScale accelerations are higher than those of typical PWRs, resulting in higher grid impact loads for the NuScale fuel design. The predicted impact response from the spacer grid is within the range of applicability under which the grid model is developed. Per Section 6.1.2.1.2 of Reference 2, the grid model parameters are defined based on test data up to the maximum allowable deformation limit of the grid. As long as the NuScale core plate excitations result in grid impacts that are within this limit, the predicted responses will remain valid.

   The second difference is that the peak acceleration for NuScale has **[**

   ].

Impact on Topical Report:

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

References:

1. TR-0716-50351-P, Rev. 0, NuScale Applicability of AREVA Method for Evaluation of Fuel Assembly Structural Response to Externally Applied Forces

2. ANP-10337P, Rev. 0, PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations

Figure 1: Envelope of NuScale SSE Upper Core Plate Horizontal Acceleration Response Spectra

Figure 2: Envelope of B&W Plant SSE Upper Core Plate Horizontal Acceleration Response Spectra

Response to Request for Additional Information
Docket No. PROJ0769

eRAI No.: 8736
Date of RAI Issue: 04/10/2017

NRC Question No.: 29614

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)(1) provides review guidance regarding the requirement to maintain control rod insertability.

TR-0716-50351-P section 3.1 references Appendix F of ANP-10337P to provide justification for the use of ASME code Level C stress limits to ensure guide tube functionality for the NuScale fuel assembly design. While Appendix F of ANP-10337P defines the Level C stress limit generally and provides discussions demonstrating that Level C stress limits would prevent generalized buckling of the thin walled tubing, it does not define the amount of localized plastic deformation that would occur nor analyze the potential impact on rod insertion times.

Basic ASME Level C stress limits are defined on an elastic basis. The primary membrane plus bending stress limit for Level C is 1.5 times the yield strength, which implies that a guide tube analyzed and evaluated on an elastic basis could experience significant permanent deformation when realistic elastic-plastic material behavior is accounted for. Permanent guide tube deformation of the amount that is permitted by Level C can potentially obstruct the control rod insertion path.

This issue is expected to be design-specific because the amount of permanent deformation that is possible under Service Level C limits and the number of potential deformed shapes is expected to be related to the specific geometry of the guide tube design and the spacer grid locations. This question is specific to the NuScale guide tube geometry and its material properties under temperature and irradiation conditions expected during service in a NuScale reactor.

a. Demonstrate that NuScale control rods remain insertable when the guide tubes are deformed to the most limiting state permitted by ASME BPVC Service level C limits.

b. Confirm that NuScale control rods still meet insertion time limits even under the limiting state permitted by Service Level C.

NuScale Response:

The NuScale response is provided below:

Generic Aspects of the Level C Service Limits

The following statement regarding the *generic* basis for the Level C service limits is made in this RAI:

> "While Appendix F of ANP-10337P defines the Level C stress limit generally and provides discussions demonstrating that Level C stress limits would prevent generalized buckling of the thin walled tubing, it does not define the amount of localized plastic deformation that would occur nor analyze the potential impact on rod insertion times."

This statement is addressed with respect to ANP-10337P in two parts: (1) defining the amount of plastic deformation that occurs at Level C stress limits and (2) addressing the impact on control rod drop times:

1. Plastic Deformation at Level C Stress Limit

While the Level C service limit is implemented in ANP-10337P via a stress limit, the Level C limit is also associated with a fixed strain value. This strain value can be extracted from the benchmarked model of a single guide tube under a bending load, as shown in Figure F-7 of ANP-10337P. After adjusting this model to hot, operating conditions, the load versus strain curve is defined as shown in Figure 1. From this figure, the strain value at Level C is the value corresponding to the design based, Level C limit load. As illustrated in Figure 1, after removal of the Level C limit load, the permanent strain that remains is slightly more than **[ ]**.

2. Impact of Plastic Deformation at Level C Limits on Control Rod Insertion Times

To demonstrate that the level of plastic deformation at Level C maintains acceptable control rod drop times, control rod drop test results are discussed. Control rod drop testing has been performed on 12 and 14 foot 17x17 fuel assemblies with the same cross-sectional geometry as the NuScale design. This testing was performed in a loop circulating coolant with various fuel assembly deflections and flow rates.

Figure 2 presents the drop time results for varying magnitudes of fuel assembly deflection in a pure mode 1 (C-shape). In this figure, the T5 (time to dashpot) data shows **[** **]**. This **[** **]** deflection is to be compared to the *permanent* deflection that remains after the assembly has been subjected to loading up to the Level C limit. Using the model presented in Section F.4 of ANP-10337P, it is demonstrated that after the limiting guide tube has reached the Level C limit under C-shape deflections at reactor operating conditions, the permanent deflection in a 14 foot 17x17 assembly is less than **[** **]**. Thus, under C-shape deformation, the test data demonstrates that control rod drop times are not impacted at levels of deformation up to and beyond the Level C limit.

Given the nature of the inertial induced loading response from seismic and LOCA events, the mode 1, or C-shape, response dominates. Higher mode responses can be present in the dynamic response from a fuel assembly, but with diminished participation when compared to the mode 1 response. It is a practical impossibility that the Level C limit is attained in a pure mode 3 (W-shape) response. Furthermore, the Level C limit is not physically attainable within the restricted confines of the reactor core geometry. Applying the model from Section F.4 of ANP-10337P, reaching Level C in a pure mode 3 configuration requires lateral deflections on the order of **[** **]**. Because fuel assembly deflection occurs in two opposing directions in mode 3, the available space in the core is only half of the accumulated gaps across the longest row. Even across a 17-assembly row of a large PWR, approximately 0.6 inches of space is all that is available to the assembly. This level of deflection is not sufficient to challenge Level C limits under a pure mode 3 deflection.

<u>Applicability of the Generic Level C Service Limits to the NuScale design</u>

This section responds to the NuScale specific requests made in this RAI:

“a. Demonstrate that NuScale control rods remain insertable when the guide tubes are deformed to the most limiting state permitted by ASME BPVC Service level C limits.

b. Confirm that NuScale control rods still meet insertion time limits even under the limiting state permitted by Service Level C.”

Both requests are addressed jointly in this response by showing that Level C limits remain applicable to NuScale through a comparison of critical dimensions.

The strain deformation associated with the Level C service limit was discussed above. The lateral deflection behavior of a fuel assembly is characterized by the fact that the slender components within a grid span follow the deflections of the spacer grids. The individual span deflections sum up to the overall assembly deflection, while the guide tube strain is controlled by the end point deflection and length of each span. The direct consequence of this is that the guide tubes in two fuel assemblies with the same cross-sectional geometry and similar span length will experience the same strains if the ratios of assembly length to deflection are the same for the two designs.

In the case of the NuScale design and the 14 foot 17x17 design for which test data is presented above, both designs have the same cross-sectional geometry and will operate with the same fuel assembly pitch. The longest row across a NuScale core is seven fuel assemblies, compared to 17 assemblies for a core with 14 foot 17x17 assemblies. Thus, the space available to a NuScale fuel assembly for lateral deflection is less than half of that of a full length 17x17 assembly. Comparing to the length of the 14 foot 17x17 fuel design referenced above, the NuScale fuel design is approximately one-half of the length. As a result, the ratio of deflection to fuel assembly length is less than the reference 14 foot 17x17 design. Thus, the NuScale design will not experience strains in excess of those that have been shown to result in a negligible effect on control rod drop times for 12 and 14 foot assemblies.

NuScale has performed preliminary SCRAM drop time testing that included a bowed fuel assembly. The maximum calculated assembly deflection used for the bowed fuel assembly, 0.4”, is based on the accumulated gaps between adjacent fuel assemblies and between fuel assemblies and the core reflector. Under the bowed condition, the full insertion drop time increased approximately 6% from 1.57 sec to 1.66 sec. This result indicates that the SCRAM drop time is not sensitive to the maximum potential guide tube deformation. Additional testing planned for completion in 2018 will confirm that under maximum assembly deflection the measured SCRAM drop time will not challenge the 2.278 sec SCRAM drop time used in NuScale safety analyses.

In summary, the use of Level C limits for NuScale will result in the same strain level that has demonstrated a negligible effect on control rod insertion times for 12 and 14 foot fuel assemblies. The NuScale control rods will remain insertable when the guide tubes are deformed to the Level C limit and the insertion times will remain below the time assumed in the safety analysis.

Impact on Topical Report:

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

References:

1. ANP-10337P, Rev. 0, PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations

Figure 1: Determination of Level-C strain from three-point bending test simulation

Figure 2: CIGARE 1300 - C Shape Drop Time vs. Deflection Amplitude and Flow Rate

<u>Note for Figure 2:</u> *The deflection amplitudes are zero to peak.*

Response to Request for Additional Information
Docket No. PROJ0769

eRAI No.: 8736
Date of RAI Issue: 04/10/2017

NRC Question No.: 29615

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 (II)(2) provides review guidance regarding the review of analytical methods for the load analysis.

TR-0716-50351-P states that the structural analysis code CASAC is used in the analysis but does not specify which version.

What version of CASAC is used in the analysis of the NuScale fuel assembly structural response to externally applied loads?

NuScale Response:

The analysis for NuScale was performed with version 5.4 of CASAC, consistent with the version of CASAC that is referenced in ANP-10337P (Reference 2) and implemented through TR-0716-50351 (Reference 1). Minor CASAC code versions (5.4.1, 5.4.2, etc.) are maintained to be in compliance with ANP-10337P.

Impact on Topical Report:

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

References:

1. TR-0716-50351-P, Rev. 0, NuScale Applicability of AREVA Method for Evaluation of Fuel Assembly Structural Response to Externally Applied Forces

2. ANP-10337P, Rev. 0, PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations

Response to Request for Additional Information
Docket No. PROJ0769

eRAI No.: 8736
Date of RAI Issue: 04/10/2017

NRC Question No.: 29616

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 (II)(1) provides review guidance regarding the review of inputs for the load analysis.

Appendix A of TR-0716-50351-P discusses the applicability of the NuScale fuel characterization test data to the referenced methodology. Figure A.3-1 provides a first-mode frequency versus deflection amplitude plot, but the definition of deflection as used in this plot is not defined in this appendix.

> Define “deflection” as it is appears in Figure A.3-1 (i.e. initial pluck deflection, the peak deflection following pluck release, etc.) and explain how this deflection value is obtained from pluck test data.

NuScale Response:

The horizontal axis of Figure A.3-1 of TR-0716-50351P (Reference 1) is labeled as “Deflection”. “Deflection” in this case represents the **[**

].

Impact on Topical Report:

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

References:

1. TR-0716-50351-P, Rev. 0, NuScale Applicability of AREVA Method for Evaluation of Fuel Assembly Structural Response to Externally Applied Forces

Enclosure 3:

AREVA Affidavit

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 Responses to NRC Request for Additional Information eRAI No. 8736 RAI 29611, 29613, 29614, 29615, and 29616," 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 7th

day of June, 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