**U.S. NUCLEAR REGULATORY COMMISSION
TR-0716-50351, REVISION 0, "NUSCALE APPLICABILITY OF AREVA METHOD FOR THE EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE TO EXTERNALLY APPLIED FORCES"**

1.0 INTRODUCTION

By letter dated September 30, 2016, NuScale Power, LLC (NuScale), submitted Topical Report (TR)-0716-50351, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," Revision 0, issued September 2016 (Ref. 1), to the U.S. Nuclear Regulatory Commission (NRC or the Commission). NuScale asked the NRC to review and approve the use of AREVA's methodology as described in ANP-10377P-A, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," Revision 0, dated April 30, 2018 (Ref. 2), for the NuScale design.

This safety evaluation report (SER) is based on TR-0716-50351 (Ref. 1) and the applicant's responses to requests for additional information (RAIs). TR-0716-50351 is designed to be referenced as part of a design certification (DC) licensing approval request. TR-0716-50351 examines the applicability of the AREVA fuel assembly structural response analysis methodology (ANP-10337P-A (Ref. 2)) by analyzing the differences between the NuScale reactor and fuel design as compared with the reactor and fuel designs covered by the referenced AREVA methodology. The methodology presented in ANP-10337P-A covers the following areas:

- acceptance criteria
- model architecture
- model parameter and allowable limits definition
- seismic and loss-of-coolant accident (LOCA) analysis
- non-grid component strength evaluation methodology

TR-0716-50351 (Ref. 1) reviews ANP-10337P-A (Ref. 2) in its entirety and determines the applicability of each section to the NuScale fuel assembly and plant design. Additionally, the report identifies NuScale design differences and analyzes potential impacts.

This SER is divided into seven sections. Section 1 is the introduction, Section 2 summarizes applicable regulatory criteria and guidance, Section 3 summarizes the information presented in TR-0716-50351 (Ref. 1), Section 4 gives the technical evaluation of TR-0716-50351, Section 5 presents the conclusions of this review, Section 6 provides the restrictions and limitations on the use of TR-0716-50351, and Section 7 outlines the references.

2.0 REGULATORY EVALUATION

The applicant submitted TR-0716-50351 (Ref. 1) to justify the use and demonstrate the applicability of previously approved AREVA codes and methods (ANP-10337P-A, Ref. 2) for NuScale safety analyses (SAs). These AREVA codes and methodologies are associated with the fuel system design and generally follow the guidance in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," to Section 4.2, "Fuel System Design," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," issued March 2007 (SRP) (Ref. 3).

TR-0716-50351 (Ref. 1), by itself, does not include an SA; instead, a DC application, combined license application, or license amendment request would reference the TR as the basis for an SA in a licensing action. Therefore, TR-0716-50351 does not independently demonstrate compliance with any rules and regulations; instead, it provides tools that an applicant for a license, permit, or certification could use to demonstrate compliance. Based on the intent of TR-0716-50351, the staff does not make any findings about compliance with specific rules or regulations; instead, the staff considers the related rules, regulations, and guidance during its review to determine whether previously approved TRs on AREVA codes and methods apply to NuScale based on the plant design differences. The staff will make findings regarding compliance with specific rules and regulations for the NuScale design in the SER associated with Section 4.2 of the NuScale DCA, Part 2, Chapter 4.

The following sections present the relevant requirements and guidance that the staff used to inform its review.

2.1. Rules and Regulations Evaluation

Title 10 the *Code of Federal Regulations* (10 CFR) 52.47, “Contents of Applications; Technical Information,” requires a standard DC to contain a level of design information sufficient to enable the Commission to judge the applicant’s proposed means of assuring that the construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before granting the certification. Specifically, 10 CFR 52.47(a)(3) requires the DC application to contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents an SA of the structures, systems, and components and of the facility as a whole. It must include, among other things, the design of the facility, including (1) the principal design criteria (PDC) for the facility, (2) the design bases and the relation of the design bases to the PDC, and (3) sufficient information on the materials of construction, general arrangement, and approximate dimensions to provide reasonable assurance that the design will conform to the design bases with an adequate margin for safety.

Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” establishes the minimum requirements for the PDC for water-cooled nuclear power plants similar in design and location to plants for which the Commission had previously issued construction permits, and it provides guidance to applicants in establishing PDC for other types of nuclear power units. General Design Criterion (GDC) 2, “Design Bases for Protection against Natural Phenomena,” requires structures, systems, and components to be designed to withstand the effects of natural phenomena such as earthquakes without loss of capability to perform their safety functions. The design bases must reflect (1) appropriate consideration of the most severe natural phenomena that have been historically reported for the site, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (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.

The focus of TR-0716-50351 (Ref. 1) is to demonstrate the applicability of the referenced codes and methods to NuScale’s licensing actions (e.g., a DC) to analyze the fuel assembly structural response, as required by GDC 2. The staff notes that TR-0716-50351 (Ref. 1) is an applicability TR, which does not develop nor implement a methodology for an SA; instead, TR-0716-50351 justifies extending the applicability of a previously approved methodology to a plant and fuel design not included in the development of the original methodology. Therefore, the staff’s review would not result in a finding against a specific rule or regulation; instead, any approval would allow an applicant to use the referenced methodology to perform a NuScale specific analysis to determine compliance with the applicable regulations.

2.2. Guidance Evaluation

The SRP provides detailed review guidance regarding methods that the staff finds acceptable in meeting the applicable regulatory requirements. Specifically, SRP Section 4.2 Appendix A contains guidance relevant to this review. TR-0716-50351 (Ref. 1) does not contain an actual analysis of the NuScale fuel system design; instead, it provides an applicability analysis of AREVA codes and methods to the NuScale fuel system design. For this reason, the staff used the guidance in SRP Section 4.2, Appendix A, to identify the sensitive parameters to assist reviewers in determining the applicability of ANP-10337P-A (Ref. 2) to the NuScale design.

3.0 SUMMARY OF TECHNICAL INFORMATION

TR-0716-50351 (Ref. 1) analyzes the applicability of the AREVA fuel assembly structural response methodology for the NuScale small modular reactor design. The purpose of TR-0716-50351 is to provide a regulatory basis for the use of ANP-10337P-A (Ref. 2) to support the NuScale DC submittal and specifically the analysis of the fuel assembly structural response as presented in DCD Section 4.2.

3.1. Review of ANP-10337P-A

Section 3 of TR-0716-50351 (Ref. 1) reviews the referenced AREVA methodology (ANP-10337P-A (Ref. 2)) against the NuScale design. Section 3 compares the NuScale fuel assembly design to the designs covered by the AREVA methodology and analyzes the applicability for each chapter of ANP-10337P-A to the NuScale fuel assembly design.

TR-0716-50351 (Ref. 1) identifies three design differences between the NuScale fuel assembly design and the referenced methodology in ANP-10337P-A (Ref. 2):

(1) shorter fuel assembly length
(2) reduced number of relevant mode shapes
(3) reduced axial coolant flow velocity

TR-0716-50351 (Ref. 1) further analyzes each of these differences.
In addition, NuScale's response to RAI No. 9555 (Ref. 4) contains the analysis of the limits and conditions from ANP-10337P-A (Ref. 2) as they pertain to the NuScale fuel assembly design. TR-0716-50351 (Ref. 1) discusses modifications to the fuel assembly modelling that is presented in ANP-10337P-A (Ref. 2) to address the physical differences between the NuScale fuel assembly and the fuel assembly designs evaluated in ANP-10337P-A.

4.0 TECHNICAL EVALUATION

4.1. Fuel Assembly Length

TR-0716-50351 (Ref. 1) addresses the shorter fuel length by adjusting the horizontal model methodology. Although the NuScale model retained one beam element for each grid span, it ignored the top and bottom spacers. The staff performed independent confirmatory analyses to evaluate the impact of the top and bottom spacer grids on the finite element model behavior.

The results confirmed that they have a negligible effect; therefore, the staff agrees that NuScale's modeling is appropriate.

4.2. Relevant Mode Shapes

The relatively shorter length of the NuScale fuel assembly naturally alters its vibration behavior compared to full-length fuel. The methodology presented in ANP-10337P-A (Ref. 2) describes mechanical test protocols used to collect information needed to generate the fuel assembly in-core seismic response finite element models. The normal mechanical testing procedure is used to determine the first five mode shapes of full-length fuel and benchmark the finite element model to the first and third mode behavior. For the shorter NuScale bundle, it is only practical to perform characterization tests for the first, second, and third modes; however, these tests include the key first and third mode vibration frequencies that are necessary to build the finite element model according to ANP-10337P-A. In Section 3.3.2 of TR-0716-50351 (Ref. 1), NuScale justified why it did not consider higher modes in the analysis. The underlying basis presented is that the relative increase in bending stiffness for the NuScale fuel assembly has increased the frequency response of the higher modes such that they are now in frequency ranges that would not appreciably contribute to the overall loadings. The staff reviewed the core plate response spectrum against the fuel assembly natural frequencies and confirmed that mode shapes higher than the third mode would be negligible when calculating the fuel assembly load demands. Based on the specifics of the NuScale fuel assembly natural frequency response and the core plate response spectrum, the staff finds NuScale's use of the lower modes acceptable for analyzing NuScale fuel assemblies.

4.3. Reduced Axial Flow

The methodology presented in ANP-10337P-A (Ref. 2) generically defines coolant flow damping for all pressurized-water reactor (PWR) fuel assemblies. NuScale recognized the differences in coolant flow rates between the NuScale reactor design and standard PWR reactor designs and adjusted the damping methodology to account for these differences. Because of the lower coolant flow rates, NuScale does not credit flow damping in accordance with ANP-10337P-A; instead, it only incorporates structural damping and still-water damping.

The staff reviewed the method used to determine structural damping and confirmed that it followed the methodology presented in ANP-10337P-A (Ref. 2); however, the staff issued RAI No. 8736 on still-water damping as used by NuScale. In its response to RAI No. 8736 (Ref. 5), NuScale provided information that supports the development of the NuScale damping values as presented in TR-0716-50351 (Ref. 1). The staff reviewed this additional information and determined that it supports the still-water damping values in TR-0716-50351.

Based on the elimination of credit for coolant flow damping and the methodology used to determine still-water damping, the staff finds that the damping values presented in TR-0716-50351 (Ref. 1) appropriately account for the reduced coolant flow velocities and, therefore, are acceptable.

4.4. Limits and Conditions Evaluation (ANP-10337P-A)

The referenced Framatome fuel seismic response methodology (ANP-10337P-A (Ref. 2)) contains nine limitations and conditions. NuScale analyzed the reactor and fuel design against these limitations and conditions in response to RAI No. 9555 (Ref. 4), as discussed below.

NuScale's response also provided markups for TR-0716-50351 (Ref. 2), and the staff is tracking the implementation of these markups as **Confirmatory Item CI-01**.

The staff evaluated the NuScale design against the limitations and conditions as provided below.

Limitation No. 1 from ANP-10337P-A

1. Dynamic grid crush tests must be conducted in accordance with Section 6.1.2.1 of ANP-10337P (as amended by RAI No. 16), and spacer grid behavior must satisfy the requirements in the topical report, the key elements of which are:

   a. [ ]

   b. [ ]

   c. [ ]

The staff confirmed that the NuScale grid design is the same HTP™ grid design used as an example in ANP-10337P-A (Ref. 2). All aspects of Limitation No. 1 were demonstrated to be met by this particular grid design during the review of ANP-10337P-A. No additional review was necessary beyond the staff's review of ANP-10337P-A. NuScale documented that the grids were the same design in its response to RAI No. 9555 (Ref. 4).

Limitation No. 2 from ANP-10337P-A

2. For fuel assembly designs where spacer grid applied loads are limited based on allowable grid permanent deformation (as opposed to buckling), the following limits from Table 4-1 of the topical report apply:

   a. For all operating-basis earthquake (OBE), analyses, allowable spacer grid deformation is limited to design tolerances and [ ].

   b. For safe-shutdown earthquake (SSE), LOCA, and combined SSE+LOCA analyses, [ ]

The NuScale fuel assembly design incorporated the same HTP™ grid design used as an example in ANP-10337P-A (Ref. 2). In its response to RAI No. 9555 (Ref. 4), NuScale stated that the HTP™ grids were the same design and that the grid allowable limits are identical to those in ANP-10337P-A. Therefore, the staff finds that the Limitation No. 6 from ANP-10337P-A applies to NuScale.

<u>Limitation No. 3 from ANP-10337P-A</u>

3. The modification or use of the codes CASAC and ANSYS (or other similar industry standard codes) are subject to the following limitations:

   a. CASAC computer code revisions, necessitated by errors discovered in the source code, needed to return the algorithms to those described in ANP-10337P (as updated by RAIs) are acceptable.

   b. Changes to CASAC numerical methods to improve code convergence or speed of convergence, transfer of the code to a different computing platform to facilitate utilization, addition of features that support effective code input/output, and changes to details below the level described in ANP-10337P would not be considered to constitute a departure from a method of evaluation in the safety analysis. Such changes may be used in licensing calculations without NRC staff review and approval. However, all code changes must be documented in an auditable manner to meet the quality assurance requirements of 10 CFR Part 50, Appendix B.

   c. ANSYS or other industry standard codes may be used if they are documented in an auditable manner to meet the quality assurance requirements of 10 CFR Part 50, Appendix B, including the appropriate verification and validation for the intended application of the code.

The NuScale fuel seismic response analysis methodology is based on the use of the CASAC computer code. The staff confirmed that the version of CASAC used in the DC application meets parts a and b of Limitation No. 3. In its response to RAI No. 8736 (Ref. 5), NuScale stated that CASAC meets this limitation. Because NuScale did not use the ANSYS computer code (or any other industry code), part c of Limitation No. 3 does not apply. Therefore, the staff finds that the applicant met the requirements of Limitation No. 3 of the referenced methodology in ANP-10337P-A (Ref. 2).

<u>Limitation No. 4 from ANP-10337P-A</u>

> This methodology is limited to applications that are similar to the current operating fleet of PWR reactor and fuel designs. The core geometry should be comparable to the current fleet, in terms of dimensions, dimension tolerances, fuel assembly row lengths, and the gaps between fuel assemblies. Fuel designs should be comparable to the current fleet, in terms of materials, geometry, and dynamic behavior.

In TR-0716-50351 (Ref. 1), NuScale justified its application of the analysis methodology in ANP-10337P-A (Ref. 2) to the NuScale fuel assembly design. The NuScale reactor core and fuel design parameters contain some differences from the current PWR operating fleet;

however, the staff finds that NuScale has appropriately modified the analysis methodology and appropriately demonstrated that the behavior of the NuScale fuel is similar enough to the operating fleet that the analysis methodology provides a means of reasonably assuring safety. The following three technical topics are at the root of this limitation, and the staff has determined them to be resolved:

(1) Linear Stiffness Model. The linear fuel assembly stiffness model of ANP-10337P-A (Ref. 2) is appropriate for typical deflection range of the operating fleet. The shorter NuScale design raised concerns about the ability of the model to accurately predict lateral deflections of the fuel assembly. NuScale resolved this through the maximum deflections reported in RAI No. 9555 (Ref. 4). The staff's independent confirmatory models closely matched and therefore supported the results reported by NuScale. The staff's concerns were resolved based on the relatively small deflections that are appropriate for the models defined in ANP-10337P-A.

(2) American Society of Mechanical Engineers Level C Stress Limits for Control Rod Insertion. As stated by NuScale in Section 4 of TR-0716-50351-P (Ref. 1), a first bending mode shape dominates the deflection response; which is in the database of insertion test results identified in ANP-10337P-A (Ref. 2). The staff notes that this deflection shape is typical for PWR fuel.

(3) Time Phasing. The staff's review of ANP-10337P-A (Ref. 2) concluded that, based on operational experience, the use of time phasing according to the method defined in ANP-10337P-A is reasonable for typical PWR fuel assemblies. NuScale's response to RAI No. 9555 (Ref. 4) and the staff's independent models confirm that the NuScale deflection behavior is very similar to typical PWR deflection; therefore, time phasing remains reasonable.

Based on the above discussion on the lateral stiffness model, stress limits for control rod insertion, and time phasing, the staff finds that the NuScale dynamic response is comparable to a typical PWR fuel assembly and that this limitation has been met.

Limitation No. 5 from ANP-10337P-A

> ANP-10337P established generic fixed damping values intended to be used for all PWR designs. All applications of this methodology to new fuel assembly designs must consider the continued applicability of the fixed damping values of this methodology. If new materials, new geometry, or new design features of a new fuel assembly design may affect damping, additional testing and/or evaluation to determine appropriate damping values may be required.

NuScale addressed this limitation by defining specific damping values to be used in the NuScale analysis instead of the generic values defined in ANP-10337P-A (Ref. 2). NuScale proposed specific damping values in TR-0716-50351-P (Ref. 1) and provided additional justification in its response to RAI No. 8736 (Ref. 5). The staff performed confirmatory analyses that included a sensitivity study on the damping, which supported NuScale's position that the results are not unusually sensitive to the choice of damping value. The staff finds that the alternate damping values are appropriate and meet the intent of this limitation.

Limitation No. 6 from ANP-10337P-A

> The ANP-10337P methodology includes the generation of fuel rod loads but does not provide a means to demonstrate compliance for fuel rod performance under externally applied loads (to applicable acceptance criteria). Applications of this methodology must provide an acceptable demonstration of fuel rod performance.

TR-0816-51127, "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," issued January 2017 (Ref. 7), evaluates fuel rod performance using limits as determined by the methodology in BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," Revision 1, issued June 2003 (Ref. 8), and evaluates loads from the methodology defined in ANP-10337P-A (Ref. 2). The methodology used to determine the fuel rod limits remains applicable to NuScale because there is no fuel rod length dependence as supported by TR-0116-20825-P-A, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1, dated November 24, 2017 (Ref. 9).

However, fuel rod and assembly length are inherent to the fuel assembly structural response methodology (Ref. 2) that generates the loads for the fuel rod analyses. TR-0716-50351 (Ref. 1) addresses the length differences and the applicability of the methodology for shorter length fuel designs. Because this SER concludes that the load generation methodology applies to the NuScale design, loads can be transferred into the fuel rod analysis methodology for analysis of the overall fuel rod performance. Therefore, the staff finds that this limitation has been met.

Limitation No. 7 from ANP-10337P-A

> As indicated in ANP-10337P when orthogonal deflections from separate core locations are artificially superimposed to calculate component stresses, the component stresses must be compared against the design criteria associated with control rod positions.

In its response to RAI No. 9555 (Ref. 4), NuScale stated that the analysis applied the Service Level C stress limits associated with control rod positions to their structural analysis of fuel assembly components. The staff finds that this meets the criteria of Limitation No. 7.

Limitation No. 8 from ANP-10337P-A

> In accordance with RG 1.92, the combination of loads for non-grid component evaluation should ideally be based on three orthogonal components (two horizontal and one vertical). [
>
>].

In its response to RAI No. 9555 (Ref. 4), NuScale stated that it performed the structural analysis of the fuel assembly using the three-dimensional combination of orthogonal loads. Therefore, the staff finds that the NuScale analysis methodology meets the criteria of Limitation No. 8.

Limitation No. 9 from ANP-10337P-A

[
]

TR-0716-50351 (Ref. 1) states that the NuScale fuel assembly design uses the same HTP™ grid design used as an example in ANP-10337P-A (Ref. 2). Therefore, the staff finds that the grid deformation limits from Limitation No. 9 of ANP-10337P-A remain applicable to the NuScale fuel assembly design.

5.0 STAFF CONCLUSIONS

The staff has completed its review of TR-0716-50351 (Ref. 1) and concludes that the applicant demonstrated that the AREVA fuel assembly structural response methodology described in the TR-0716-50351 can be used, with the stated modifications, to perform NuScale fuel system structural response analyses. The staff reached its conclusions by (1) reviewing the differences between the NuScale plant and fuel designs against those used in the previously approved methodology, (2) reviewing the conditions and limitations of the referenced approved methodology TR, (3) independently verifying that the expected NuScale parameters fall within the validation limits of the respective referenced approved TRs, and (4) evaluating the justification in TR-0716-50351 for all modifications used to address design differences.

Therefore, the staff approves the use of the AREVA fuel assembly structural response analysis methodology (ANP-10337P-A (Ref. 2)) to analyze the NuScale fuel system design, as described in TR-0716-50351 (Ref. 1).

6.0 CONDITIONS AND LIMITATIONS

The staff limited its evaluation of TR-0716-50351 (Ref. 1) to the fuel design and operating parameters as presented in the TR. Any applicant or licensee referencing this TR that wishes to operate with fuel designs different from those presented in TR-0716-50351 would need to address differences in its application or license amendment request. Fuel designs modified under an approved fuel assembly design change process methodology would still be able to apply the referenced methodology to the NuScale design.

7.0 REFERENCES

1. TR-0716-50351, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," Revision 0, NRC Project No. 0769, September 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16274A469).

2. ANP-10337P-A, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," Revision 0, April 30, 2018 (ADAMS Accession No. ML18144A816).

3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," March 2007 (ADAMS Accession No. ML070810350).

4. “NuScale Power—Response to NRC Request for Additional Information eRAI No. 9555,” August 13, 2018 (ADAMS Accession No. ML18226A357).

5. “NuScale Power LLC Response to NRC Request for Additional Information No. 13 (eRAI No. 8736),” June 8, 2017 (ADAMS Accession No. ML17160A169).

6. NP-RT-0612-023, “Gap Analysis Summary Report,” Revision 1, July 2014 (ADAMS Accession No. ML14212A832).

7. TR-0816-51127, “NuFuel-HTP2™ Fuel and Control Rod Assembly Designs,” Revision 1, January 2017 (ADAMS Accession No. ML17007A001).

8. BAW-10227P-A, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel,” Revision 1, June 2003 (ADAMS Accession No. ML17130A709).

9. TR-0116-20825-P-A, “Applicability of AREVA Fuel Methodology for the NuScale Design,” Revision 1, November 24, 2017 (ADAMS Accession No. ML18040B306).