May 1, 2020 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 Submittal of Approved Version of NuScale Topical Report "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351, Revision 1

REFERENCE: Letter from Bruce Bavol (NRC) to Zackary Rad (NuScale), "Final Safety Evaluation for NuScale Power, LLC Topical Report TR-0716-50351, Revision 1, 'NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces,' " dated February 27, 2020 (ML20056C907)

By the referenced letter dated February 27, 2020, the NRC issued a final safety evaluation report documenting the NRC Staff conclusion that NuScale Topical Report "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351, Revision 1, is acceptable for referencing in licensing applications for the NuScale design. The referenced letter requested that NuScale publish the approved version of TR-0716-50351, Revision 1, within three months of receipt of the letter.

Accordingly, Enclosure 1 to this letter provides the approved version of the topical report, designated TR-0716-50351-P-A, Revision 1. The enclosure includes the February 27, 2020 NRC letter and its final safety evaluation report, the NuScale responses to NRC requests for additional information, and the final topical report submittal (Revision 1).

Enclosure 1 contains proprietary information. 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) pertains to Framatome Inc. (formerly AREVA Inc.) proprietary information to be withheld from the public. Framatome proprietary information is denoted by bolded straight brackets (i.e., "**[ ]**"). Enclosure 2 is the nonproprietary version of the approved topical report package.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please feel free to contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com if you have any questions.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8H12
Michael Dudek, NRC, OWFN-8H12
Bruce Bavol, NRC, OWFN-8H12

Enclosure 1: "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351-P-A, Revision 1, proprietary version
Enclosure 2: "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351-NP-A, Revision 1, nonproprietary version
Enclosure 3: Framatome Affidavit of Gayle Elliott

Enclosure 1:

"NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351-P-A, Revision 1, proprietary version

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 Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces,” TR-0716-50351-NP-A, Revision 1, nonproprietary version

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

Section A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 27, 2020

Mr. Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC.
1100 Circle Boulevard, Suite 200
Corvallis, OR 97330

SUBJECT: FINAL SAFETY EVALUATION FOR NUSCALE POWER, LLC TOPICAL REPORT TR-0716-50351, REVISION 1, "NUSCALE APPLICABILITY OF AREVA METHOD FOR THE EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE TO EXTERNALLY APPLIED FORCES"

Dear Mr. Rad:

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, to the U.S. Nuclear Regulatory Commission (NRC). 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, for the NuScale design. By letter dated December 19, 2019, NuScale submitted Revision 1, TR-0716-50351, which incorporated changes from request for additional information responses.

The NRC staff has evaluated TR-0716-50351, Revision 1, and found that it is acceptable for referencing licensing applications for the NuScale small modular reactor design to the extent specified and under the conditions and limitations delineated in the enclosed safety evaluation report (SER).

The NRC staff requests that NuScale publish the applicable version(s) of the SER listed above within three months of receipt of this letter. The accepted version of the TR shall incorporate this letter and the enclosed SER and add "-A" (designated accepted) following the report identification number.

CONTACT: Bruce M. Bavol, NRR/DNRL
301-415-6715

If the NRC staff's criteria or regulations change, and its acceptability conclusion in the SER is invalidated, NuScale and/or the applicant referencing the SER will be expected to revise and resubmit its respective documentation, or submit justification for continued applicability of the SER without revision of the respective documentation.

Prior to placing the public version of this document in the publicly available records component of NRC's Agencywide Documents and Access Management System (ADAMS), the NRC staff requests that NuScale perform a final review of the SER for proprietary or security related information not previously identified. If you believe that any additional information meets the criteria, please identify such information line-by-line and define the basis pursuant to the criteria established in Title 10 of the *Code of Federal Regulations,* Part 2, Section 3.90, "Public inspections, exemptions, requests for withholding."

If after a 10-day period, you do not request that all or portions of the SER be withheld from public disclosure, the SER will be made available for public inspection through the publicly available records component of NRC's ADAMS.

If you have any questions or comments concerning this matter, please contact Bruce Bavol at 301-415-6715 or via e-mail address at Bruce.Bavol@nrc.gov.

Sincerely,

/RA/

Anna H. Bradford, Director
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation

Docket No. 52-048

Enclosures:
1. TR-0716-50351 SER (Public)
2. TR-0716-50351 SER (Proprietary)

cc: DC NuScale Power, LLC Listserv (w/o Enclosure 2)

SUBJECT: FINAL SAFETY EVALUATION FOR NUSCALE POWER, LLC TOPICAL REPORT TR-0716-50351, REVISION 1, "NUSCALE APPLICABILITY OF AREVA METHOD FOR THE EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE TO EXTERNALLY APPLIED FORCES" DATED: February 27, 2020

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U.S. NUCLEAR REGULATORY COMMISSION
TR-0716-50351, REVISION 1, "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, (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
- nongrid 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.

By letter dated December 19, 2019, NuScale submitted Revision 1, TR-0716-50351, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," (Ref. 10) which incorporate changes from request for additional information (RAI) responses.

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.

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.

Limitation No. 3 from ANP-10337P-A

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).

Limitation No. 4 from ANP-10337P-A

> 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 December 2019 (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).

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Section B

Licensing Topical Report

NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces

April 2020

Revision 1

Docket: PROJ0769

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Department of Energy Acknowledgement and Disclaimer

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Abstract

This topical report demonstrates the applicability of *PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations*, ANP-10337P-A (Reference 7.0-1) to the NuScale fuel assembly designed for use in the NuScale Power Module. The generic methodology described in ANP-10337P-A is used to evaluate the structural response of the NuScale fuel assembly to dynamic loads applied during seismic and loss-of-coolant accident (LOCA) events. Additional justification is provided for areas of the methodology that require special consideration due to unique features of the NuScale fuel assembly.

NuScale Power, LLC (NuScale) is submitting this topical report for Nuclear Regulatory Commission (NRC) approval to apply the methodology described in ANP-10337P-A to the NuScale fuel assembly design and for approval of the NuScale design-specific damping values. A summary of the application results are being submitted for review and approval as part of the NuScale Design Certification Application.

Executive Summary

This report demonstrates the applicability of *PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations*, ANP-10337P-A (Reference 7.0-1) to the NuScale fuel assembly designed for use in the NuScale Power Module. The generic methodology described in ANP-10337P-A is used to evaluate the structural response of the NuScale fuel assembly to dynamic loads applied during seismic and loss-of-coolant accident (LOCA) events, consistent with the guidance in Standard Review Plan Section 4.2 Appendix A, *Evaluation of Fuel Assembly Structural Response to Externally Applied Forces*, NUREG-0800.

The report provides an evaluation of each chapter of ANP-10337P-A with regard to NuScale applicability, including additional justification for areas impacted by NuScale-specific design parameters. The application of the methodology for the NuScale fuel design is being provided in the Design Certification Application.

The ANP-10337P-A methodology is applicable to any pressurized water reactor (PWR) fuel assembly geometry that can be characterized by the defined testing and model benchmarking processes and any reactor core geometry for which the seismic and LOCA boundary condtions can be defined. This report systematically demonstrates that the NuScale fuel design can be modeled and analyzed utilizing this methodology.

The report identifies three specific NuScale design differences and provides further justification or adaptation to demonstrate applicability:

1. The NuScale fuel assembly is shorter than typical PWR designs. To address this difference, an evaluation is provided to demonstrate that the modeling and benchmarking process adequately characterizes the NuScale fuel design.
2. The experimental characterization of the frequency response of the NuScale fuel design is limited to the first three natural frequencies. This difference is shown to have no impact on the resulting fuel assembly model.
3. The contribution of axial coolant flow to the NuScale fuel assembly damping is expected to be much less than that for other operating PWRs. To address this difference, NuScale-specific fuel assembly damping values are established.

To meet the requirements of ANP-10337P-A, the report provides an explicit demonstration of the applicability of grid impact modeling elements used in the generic methodology.

Based on the justifications and evaluations provided in this report, the ANP-10337P-A methodology with NuScale design-specific damping values can be applied to structural response analyses of the NuScale fuel design as part of the NuScale Design Certification Application.

1.0 Introduction

1.1 Purpose

This report demonstrates the applicability of *PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations*, ANP-10337P-A (Reference 7.0-1), to the NuScale fuel assembly designed for use in the NuScale Power Module. The generic methodology described in ANP-10337P-A is used to evaluate the structural response of the NuScale fuel assembly to dynamic loads applied during seismic and loss-of-coolant accident (LOCA) events, consistent with the guidance in Standard Review Plan Section 4.2 Appendix A, *Evaluation of Fuel Assembly Structural Response to Externally Applied Forces*, NUREG-0800 (Reference 7.0-2).

NuScale Power, LLC is submitting this topical report for Nuclear Regulatory Commission (NRC) approval to apply the methodology described in ANP-10337P-A to the NuScale fuel assembly design.

1.2 Scope

In order to demonstrate the applicability of ANP-10337P-A to the NuScale fuel assembly, this report:

- Provides an evaluation of each chapter of ANP-10337P-A with regard to NuScale applicability.
- Provides additional justification for areas of the methodology that require special consideration due to unique features of the NuScale fuel assembly.
- Summarizes selected elements of the NuScale fuel assembly modeling to enhance the applicability justification.

This report does not provide results of the full suite of characterization tests and the structural response analyses of the NuScale fuel. A summary of the application results are being submitted for review and approval as part of the NuScale Design Certification Application.

1.3 Abbreviations

Table 1-1. Abbreviations

Term	Definition
BOL	Beginning of Life
CFR	Code of Federal Regulations
EOL	End of Life
°F	Degrees Fahrenheit
Ft/s	Feet per Second
GDC	General Design Criteria
ID	Inner Diameter
Kg U	Kilograms of uranium
kW/m	Kilowatts per meter
LOCA	Loss-of-Coolant Accident
MWt	Megawatt thermal
NRC	Nuclear Regulatory Commission
OD	Outer Diameter
Psia	Pounds per Square Inch – Absolute
Psig	Pounds per Square Inch – Gauge
PWR	Pressurized Water Reactor
R^2	Coefficient of determination
RCS	Reactor Coolant System
RFT	reactor flange tool
SER	Safety Evaluation Report
SRP	Standard Review Plan

2.0 Background

ANP-10337P-A defines a generic methodology to evaluate the structural response of pressurized water reactor (PWR) fuel assembly designs subjected to dynamic loads under seismic and LOCA events. This report demonstrates that this method is applicable to the NuScale fuel design.

To assist in the review of the applicability of the methodology, a brief summary of the NuScale fuel design is provided below.

Welded Fuel Assembly Structure

The NuScale 17x17 fuel assembly design is a reduced-height version of AREVA's 17x17 PWR fuel designs for Westinghouse-type reactors. The total, nominal height of the fuel assembly is 94 inches (not including hold-down springs). Due to the reduced height and the use of span lengths between spacer grids that are typical for operating PWR plants, the assembly has a total of five spacer grids. The HTP™ grids are welded to the guide tubes, while the HMP™ grid is captured by rings welded to the guide tubes. The design includes features as described in the following subsections.

Fuel Rod with Alloy M5® Fuel Rod Cladding

The fuel rod design features M5® cladding. The seamless M5® cladding encapsulates ceramic UO_2 pellets that are cylindrically shaped with a spherical dish at each end. The fuel rod has an internal spring system that axially restricts the position of the fuel stack within the rod, preventing the formation of gaps during shipping and handling while allowing for the expansion of the fuel stack during operation. The lower end cap has a bullet-nose shape to provide a smooth flow transition in addition to facilitating insertion of the rods into the spacer grids during assembly. The upper end cap has a grippable shape that allows for the removal of the fuel rods from the fuel assembly if necessary, which is typical of AREVA fuel for operating PWR plants.

The nominal density of the pellets is 96 percent theoretical density with a possible enrichment up to 4.95 weight percent ^{235}U.

Zircaloy-4 HTP™ upper and intermediate spacer grids

The four HTP™ spacer grids that occupy the top four grid positions are formed from interlocking strips that are welded at all intersections and welded to the side plates. Each grid strip includes a pair of strips welded back-to-back to produce flow channels. The design creates a flow path that is slanted at its outlet, thus causing a vortex flow pattern under normal PWR operating conditions. The spacer grid design creates line contacts with the fuel rod, which provide resistance to grid-to-rod fretting relative to traditional point-contact spacer grid designs. The HTP™ grids on the NuScale design are identical to those used on AREVA's 17x17 PWR product.

Alloy 718 HMP™ Lower Spacer Grid

The HMP™ spacer grid resembles the HTP™ spacer grid with respect to spring design, rod-to-grid surface contact, and manufacturing. The HMP™ spacer grid, however, has enhanced strength and relaxation characteristics, and straight (non-mixing) flow channels. The HMP™ grid on the NuScale design is identical to those used on AREVA's 17x17 PWR product.

Bottom Nozzle with Mesh Filter Plate

The 304 stainless steel bottom nozzle consists of a cast frame of ribs connecting the guide tube locations. A high strength A-286 alloy mesh filter plate is pinned to the top of the frame and held in place by shoulder screws at each of the 24 guide tube locations.

Zircaloy-4 MONOBLOC™ Guide Tubes

The MONOBLOC™ guide tubes have a constant outer diameter and a reduced inner diameter that forms the guide tube dashpot. The added thickness in the dashpot of the MONOBLOC™ guide tube increases the lateral stiffness of the fuel assembly. The MONOBLOC™ feature is common to the AREVA 17x17 PWR fuel designs.

Reconstitutable Top Nozzle

The top nozzle consists of a 304-stainless steel frame that is attached to the fuel assembly with quick disconnect features at each of the 24 guide tube locations. The NuScale top nozzle design has different requirements with respect to the through-hole in the grillage to accommodate the NuScale top-entry incore detectors. The top nozzle design incorporates four sets of two-leaf hold-down springs made of Alloy 718.

Table 2-1 provides additional information on the NuScale fuel design and identifies the differences between the NuScale and AREVA 17x17 PWR fuel design. The only difference pertinent to the fuel assembly structural response is the fuel assembly height.

Table 2-1. NuScale fuel design parameters

Parameter	**NuScale Fuel Design**	**AREVA 17x17 PWR**
Fuel rod array	17 x 17	17 x 17
Fuel rod pitch (inch)	0.496	0.496
Fuel assembly pitch (inch)	8.466	8.466
Fuel assembly height (inch)*	94.0	159.45
Number of guide tubes per bundle	24	24
Dashpot region inner diameter (inch)	0.397	0.397
Dashpot region outer diameter (inch)	0.482	0.482
Inner diameter above transition (inch)	0.450	0.450
Outer diameter above transition (inch)	0.482	0.482
Number of instrument tubes per bundle	1	1
ID (inch)	0.450	0.450
OD (inch)	0.482	0.482
Number of fuel rods per bundle	264	264
Cladding outer diameter (inch)	0.374	0.374 and 0.376
Cladding inner diameter (inch)	0.326	0.326
Length of total active fuel stack (inch)*	78.74	144
Fuel pellet OD (inch)	0.3195	0.3195
Fuel pellet density (% theoretical density)	96	96
Spacer grid span lengths (inch)	20.1	20.6
Fuel rod internal pressure (psig)	215	315

* Height is measured from the seating surface of the bottom nozzle to the top of the post on the top nozzle. Dimension does not include hold-down springs.

Table 2-2 provides the operating conditions representative of the NuScale design.

Table 2-2. NuScale operating conditions

Parameter	**NuScale Design Value**	**AREVA 17x17 PWR Value**
Rated thermal power (MWt)	160	3455
Average coolant velocity (ft/s)	3.1	16
System pressure (psia)	1850	2280
Core tave (°F)	547	584
Linear heat rate (kW/m)	8.2	18.0
RCS inlet temperature (°F)	503	547
RCS Reynolds Number	76,000	468,000
Fuel assemblies in core	37	193
Fuel assembly loading (kgU)	249	455
Core loading (kgU)	9,213	87,815

In addition to the operating condition given in Table 2-2, the NuScale design also includes the condition where the reactor core is placed on the reactor flange tool (RFT) during refueling activities. When the core is in this configuration, the fuel assemblies remain in the core cavity with the same lower and upper core plate engagement as during operation. The core is submerged in water at a temperature range of 65 degrees F to 110 degrees F and the control rods are completely inserted.

2.1 Regulatory Requirements

Section 3.0 of ANP-10337P-A identifies the regulatory requirements and guidance addressed by the generic methodology. Specifically, the methodology addresses the following requirements as they relate to the structural requirements of a fuel assembly subjected to externally applied loads from earthquakes and postulated pipe breaks:

- GDC 2 – Design bases for protection against natural phenomena
- GDC 27 – Combined reactivity control systems capability
- GDC 35 – Emergency core cooling
- 10 CFR Part 50 Appendix S – Earthquake engineering criteria for nuclear power plants
- 10 CFR 50.46 – Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors

The methodology addresses these regulatory requirements consistent with guidance in SRP Section 4.2 (Reference 7.0-2).

This topical report demonstrates that the regulatory requirements and demonstration methods defined in ANP-10337P-A are applicable to the NuScale fuel design.

3.0 Review of ANP-10337P-A Topical Report

PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations, ANP-10337P-A defines the methodology for performing the evaluation of the fuel assembly structural response to externally applied forces (i.e., seismic and LOCA excitations) for PWRs. The methodology corresponds to the NRC guidance provided in Chapter 4.2, Appendix A of the Standard Review Plan (NUREG-0800). This section addresses the applicability of ANP-10337P-A to the NuScale fuel assembly.

3.1 Applications

ANP-10337P-A defines the AREVA methodology that can be applied generically to evaluate the structural response of fuel assemblies to externally applied forces. This evaluation methodology starts with the execution of design characterization testing in order to define fundamental, design-specific fuel assembly response characteristics such as natural frequencies, stiffness, damping, etc. These characteristics are applied to a generic structural model architecture in order to build design-specific models capable of simulating the fuel assembly structural response. Thus, the methodology is independent of the design being analyzed and can be used to represent the structural response of any PWR fuel assembly. Plant-specific applications are achieved through the geometry of the model boundary conditions (core dimensions, etc.) and inputs (e.g. core plate motion time histories).

An evaluation of the content of each chapter of ANP-10337P-A and its applicability to the NuScale fuel assembly is provided below. Items for which applicability to the NuScale design requires additional justification have been highlighted and are addressed in Section 3.3.

Chapter 1 Introduction

This chapter identifies the purpose and provides an overview of ANP-10337P-A. As stated in this chapter, the method is generically applicable to PWR fuel assembly designs.

Chapter 1 of ANP-10337P-A is applicable to the NuScale fuel assembly in its entirety.

Chapter 2 Applicability

This chapter establishes the generic applicability of the methods to PWR fuel assembly designs. As noted, “PWR fuel designs exhibit a similar geometry and structure that is appropriately represented by the modeling architecture defined in this topical report.” The chapter provides a generic physical description of PWR fuel assemblies and components that is consistent with the design of the NuScale fuel assembly as summarized in Section 2.0 of this report.

To provide further justification, the following key statements from ANP-10337P-A Chapter 2.0 are evaluated with respect to the applicability of this method to NuScale fuel.

1. "… this methodology is applicable to any fuel assembly geometry that can be characterized by the testing and model benchmarking processes defined in this topical."

   The NuScale fuel design is subjected to the entire testing program defined in Table 6-4 of ANP-10337P-A. Similarly, the model benchmarking activity follows the process defined in ANP-10337P-A. These testing and benchmarking activities are suitable and sufficient to characterize the structural response of the NuScale fuel assembly as demonstrated below. Thus, the conditions of the statement from the methodology topical report are satisfied and the methodology can be used to simulate NuScale fuel. A more detailed demonstration of an application of this testing and modeling to NuScale fuel is provided in Section 3.3.4.

2. "… this methodology can be applied to any reactor core geometry for which the seismic and LOCA boundary conditions can be defined."

   The reactor geometry for NuScale (i.e., upper and lower core plates and heavy reflector) is consistent with a typical PWR with regard to the boundary conditions addressed in Section 5.2.2.1 of ANP-10337P-A. The seismic and LOCA boundary conditions will be defined in the application of the method.

3. "Section 2.1 provides a discussion of typical PWR fuel assembly structure, but the specific fuel designs discussed are for illustration only and do not limit the range of applicability of this methodology."

   The NuScale fuel assembly structure is consistent with the description of the structure provided in Chapter 2 of ANP-10337P-A. The fuel assembly is constructed in the same manner and using the same components described in this chapter. Furthermore, the cross-sectional geometry is the same as one of the representative designs presented in Chapter 2. As such, the modeling and analytical approaches defined throughout ANP-10337P-A are directly relevant and applicable to the NuScale fuel, with only a few exceptions addressed in this report. In terms of specific fuel design differences, the NuScale fuel design is outside of the range of PWR designs presented in Table 2-1 of ANP-10337P-A only in terms of overall length and number of spacer grids. The modeling approach defined in ANP-10337P-A can still be applied to represent the NuScale geometry because of the consistency in components and cross-sectional geometry. The effect of a shorter bundle and fewer spacer grids is addressed in Section 3.3 of this report. The effect of these design differences will be measured and characterized as an outcome of the required design testing.

4. "Section 2.2 does identify specific spacer grid behavior that must be satisfied for the methodology defined herein to be applicable to a given PWR fuel design."

   In Section 2.2, ANP-10337P-A defines one generic restriction to the use of the method in that [

   ] is demonstrated in more

detail in Section 3.3.4 and Appendix A. Since the HTP™ grid used in the NuScale design is identical to the grid cited in the sample problem for ANP-10337P-A, this potential limitation on the applicability of the methodology is satisfied.

In summary, the NuScale design is consistent with the general conditions defined in Chapter 2 of ANP-10337P-A. Therefore, the range of applicability defined in this chapter encompasses the NuScale design. The differences in assembly length and number of spacer grids are addressed in Section 3.3 of this report. Compliance to the requirement defined in Section 2.2 of ANP-10337P-A is demonstrated in Section 3.3.4 and Appendix A.

Chapter 3 Regulatory Requirements

Chapter 3 reviews the regulatory requirements that are relevant to this methodology. These requirements include Appendix A (GDC 2, 27, and 35) and Appendix S of 10 CFR Part 50 and 10 CFR 50.46. In addition, this chapter reviews the NRC guidance from the Standard Review Plan Section 4.2 that pertains to these requirements (primarily Appendix A).

The requirements of Appendices A (GDC 2, 27, and 35), and S of 10 CFR Part 50, 10 CFR 50.46 are directly applicable to the NuScale design. The guidance of SRP Section 4.2 Appendix A is directly applicable and is implemented in ANP-10337P-A.

The same regulatory requirements identified in ANP-10337P-A Chapter 3 are applicable to the NuScale Power Module; therefore, this chapter is applicable to the NuScale design.

When the reactor is placed in the RFT during refueling operations, some of the regulatory requirements are already inherently met. In this case, the LOCA event is not considered and control rod insertion and coolable geometry are already satisfied since the reactor is at ambient temperature conditions. The requirement that fuel rod mechanical fracture will not occur due to seismic loads remains applicable.

Chapter 4 Acceptance Criteria

Chapter 4 establishes the appropriate selection of acceptance criteria in order to satisfy the regulatory requirements specified in Chapter 3. In general, this chapter establishes criteria to evaluate spacer grid impact loads and allowable stresses for non-grid components.

Like the regulatory requirements in Chapter 3, these criteria are generic to PWR fuel. The NuScale fuel design uses the same components and structure as the PWR designs presented in Chapter 2 of ANP-10337P-A; therefore, the criteria defined in this chapter can be applied to the NuScale design to demonstrate compliance to the regulatory requirements with the following minor clarification:

- Section 4.2.2 of ANP-10337P-A justifies excluding the evaluation of hold-down springs for accident condition loads based on specific designs of fuel assemblies and reactor core internals. The justification for Westinghouse designs applies to the NuScale design because both fuel types use the same design of top nozzle and hold-down spring system.

Chapter 5 Model Architecture

Chapter 5 provides a description of the generic modeling approach that is used to represent any fuel design in seismic and LOCA analyses. Independent horizontal and vertical models are defined to simulate the fuel assembly response in those directions. The chapter presents in detail the models and identifies the required model inputs for both horizontal and vertical response analyses.

As noted in the evaluation of Chapter 2, this modeling approach is generally applicable to the NuScale design due to the fact that the NuScale fuel is constructed with the same components and with the same cross-sectional geometry as designs considered in ANP-10337P-A. Inspection of Table 5-1, Table 5-2, and Figures 5-1 through 5-7 reveals that the models, boundary conditions, and required model inputs apply equally well to NuScale fuel. With the exception of the specific geometry differences noted in the Chapter 2 evaluation (i.e., overall length and number of spacer grids), only one additional distinction can be made from this chapter in applying the methodology to the NuScale design. ANP-10337P-A Section 5.2.1.1 presents three mechanisms that contribute to fuel assembly damping in the horizontal model. In the NuScale design, because of the low flow rate (Table 2-2), the contribution of axial coolant flow to the fuel assembly damping is expected to be much less than for other operating PWRs. As a result, the damping definition provided in ANP-10337P-A is not applicable to NuScale. This difference is addressed in Section 3.3.

The reactor geometry for the NuScale design (e.g. upper and lower core plates and heavy reflector) is consistent with the boundary conditions addressed in Section 5.2.2.1 of ANP-10337P-A.

With the exception noted above, Chapter 5 of ANP-10337P-A is applicable to the NuScale design.

Chapter 6 Model Parameter and Allowable Limits Definition

Chapter 6 establishes the means of defining the criteria introduced in Chapter 4 and the modeling parameters introduced in Chapter 5 using design-specific values. These parameters relate to geometric or material properties that can be established directly from design definition documents, or that require characterization testing to define. The NuScale fuel assembly utilizes identical components, with the same cross-sectional geometry, as designs considered in ANP-10337P-A; therefore, the modeling parameters associated with geometric and material properties are unchanged for the NuScale application. The remaining parameters require characterization testing in order to be defined. The full design characterization testing program identified in Table 6-4 of Chapter 6 in ANP-10337P-A will be applied to the NuScale fuel design. Differences in fuel assembly behavior due to the shorter length or fewer number of spacer grids, as noted in Chapter 2, will be characterized and accommodated as a result of the testing defined in Chapter 6.

For the seismic analysis when the reactor core is placed in the RFT, Chapter 6 of ANP-10337P-A is applicable, but no conversion of the model parameters to operating temperature is needed since the temperature in the RFT condition is taken as 70 degrees F, which is in the specified range of temperatures. The lateral and vertical models, which are benchmarked at ambient temperature, are used directly. Model parameters that are defined or measured at operating temperature, such as the BOL spacer grid through grid properties, are converted to ambient conditions.

One item from Chapter 6 is potentially affected by the application to NuScale fuel.

- Section 6.1.3 presents damping values to be used in the horizontal model. As noted in the review of Chapter 5, the contribution of axial coolant flow to the fuel assembly damping is expected to be much less than for other operating PWRs. As a result, the damping definition provided in Section 6.1.3 of ANP-10337P-A is not applicable to NuScale. This difference is addressed in Section 3.3.3.

Chapter 6 of ANP-10337P-A is applicable to the NuScale fuel design with the exception of the definition of damping values presented in Section 6.1.3. This item is addressed in Section 3.3.

Chapter 7 Seismic and LOCA Analysis

Chapter 7 defines the process of applying appropriate forcing functions representing seismic or LOCA events to the models described in Chapter 5. Chapter 7 also defines the method of accounting for the combined effect of seismic and LOCA loads. In the horizontal analysis, the model calculates the time-varying displacements and impact forces for assemblies across the core. The results of this analysis are also used for calculating the resulting loads and stresses in the assembly. Similarly, the vertical model calculates a time-varying response from the fuel assembly that is used to evaluate the loading on fuel assembly components.

It was demonstrated above that the development of design-specific models and boundary conditions in accordance with ANP-10337P-A is applicable to the NuScale design. The process to apply these models to determine the fuel assembly structural response to seismic and LOCA events is design independent. Therefore, this chapter is applicable to the NuScale fuel design.

Chapter 8 Non-Grid Component Strength Evaluation Methodology

Chapter 8 defines the process of performing the structural component stress analysis using the loads and deflections generated by the seismic and LOCA analyses described in Chapter 7. Like the modeling approach addressed in Chapters 5 and 6, the analysis approach in Chapter 8 is applicable to NuScale fuel because the fuel design uses the same components and structure, and has the same cross-sectional geometry as designs addressed in ANP-10337P-A.

The process defined in this chapter is generic and remains applicable to the NuScale fuel design.

Chapter 9 References

Chapter 9 lists the references cited throughout ANP-10337P-A and is not evaluated separately.

Appendix A: CASAC Code Description

Appendix A provides a description of the structural code, CASAC, used to carry out the modeling and analysis defined in Chapters 5, 6, and 7 of ANP-10337P-A. The application of CASAC in the analysis of NuScale fuel is appropriate given that the NuScale fuel design uses the same components and structure as the PWR fuel assemblies described in ANP-10337P-A. Therefore, Appendix A is applicable to the NuScale design.

Appendix B: Sample Problem Summary

Appendix B provides the results of the sample problem that accompanies ANP-10337P-A. The sample problem is executed for a 17x17 HTP™ fuel design for Westinghouse reactors. This fuel design has the same cross-sectional geometry as the NuScale fuel design. This section provides insight on the application of the method to a specific fuel design and reactor, but is not directly relevant to the NuScale design.

Appendix C: Test Results Supporting the Fuel Assembly Damping Formulation

Appendix C defines generic fuel assembly damping values that credit operational flow rates for PWRs. However, the reduced flow rate and the shorter fuel assembly length of the NuScale design will necessitate the establishment of a specific damping value for the NuScale fuel design, as addressed in Section 3.3.

Appendix C is not applicable to the NuScale design.

Appendix D: Simulation of the Effects of Irradiation on Dynamic Crush Characteristics with Zirconium Alloy Spacer Grids

Appendix D addresses the treatment of the effects of irradiation in the testing and analysis of zirconium-alloy spacer grids. The effects of irradiation on zirconium-alloy materials are generic, regardless of the fuel or reactor design. Therefore, the conclusions regarding the effect of irradiation on grid behavior, and furthermore, the conclusions regarding how to simulate these effects in testing, are applicable to NuScale fuel. The NuScale fuel design utilizes the 17x17 HTP™ spacer grid that is currently in use in operating plants and for which the effects of irradiation are addressed using the same methods defined in Appendix D.

This appendix is applicable to the NuScale fuel design.

Appendix E: Justification for the Use of Level C Stress Limits to Ensure Guide Tube Functionality

Appendix E provides the basis for the acceptability of using Level C stress limits to ensure guide tube functionality (i.e., control rod insertability) following a seismic or LOCA event. The discussion and data presented in Appendix E are generic to any guide tube geometry. The characterization of guide tube stress states and the definition of the Level C service limit in relation to guide tube geometry are generic. Furthermore, the testing discussed in Appendix E is performed on guide tubes of the same cross-sectional geometry as the NuScale design.

This appendix is applicable to the NuScale fuel design.

3.2 Topical Report Restrictions

This section addresses the Limitations and Conditions (L&Cs) of ANP-10337P-A in the context of the application of this methodology to the NuScale Fuel Assembly design.

L&C #1 Discussion:

L&C #1 imposes requirements on the tested behavior of grids in order to be compliant with the ANP-10337P-A methodology:

1. *Dynamic grid crush tests, must be conducted in accordance with Section 6.1.2.1 of ANP-10337P (as amended by RAI 16), and spacer grid behavior must satisfy the requirements in the TR, the key elements of which are:*
    a. **[ ]**
    b. **[ ]**
    c. **[ ]**

The acceptability of the NuScale grids under L&C #1 has been addressed in *ANP-3712P-000, Framatome Responses to NRC RAI No. 9555 regarding NuScale Topical Report TR-0816-51127* (Reference 7.0-3).

L&C #2 Discussion:

L&C #2 imposes requirements on the maximum allowable deformation of spacer grids:

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 TR apply:*

a. *For all OBE analyses, allowable spacer grid deformation is limited to design tolerances and* **[ ]***.*
b. *For SSE, LOCA, and combined SSE+LOCA analyses,* **[**

]

The acceptability of the NuScale grids under L&C #2 has been addressed in Reference 7.0-3.

L&C #3 Discussion:

L&C #3 imposes controls and quality requirements on the computer programs implementing the methodology of ANP-10337P-A:

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 analyses use the same code versions employed in the analytical method demonstration in Appendix B of ANP-10337P-A. Therefore, L&C #3 is not a concern.

L&C #4 Discussion:

L&C #4 limits the un-restricted use of the ANP-10337P-A methodology to fuel designs and applications consistent with the operating fleet. Markedly new designs have to be assessed:

4. *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.

L&C #4 has been addressed in Reference 7.0-3. While in absolute terms the NuScale fuel assembly is different from the operating fleet, this design has been demonstrated to be similar, on a scale basis to the generic fuel assembly of ANP-10337P-A. The available lateral deflections in core when scaled by fuel assembly lengths are smaller for NuScale than those for the generic assembly in ANP-10337P-A. This justifies the extension, on a scale basis, of all considerations and acceptability measures from the generic fuel assembly of ANP-10337P-A to the NuScale assembly. This observation substantiates the conclusion that the NuScale assembly is similar to the generic assembly in the approved methodology of ANP-10337P-A, and the time-phasing method is appropriate in this case.

L&C #5 Discussion:

L&C #5 limits the applicability of the lateral damping formulation to existing designs, and requires an applicability justification or a new formulation for new designs:

5. *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.*

The NuScale fuel assembly is much shorter than the current fleet designs, and the lateral fuel assembly damping has been re-formulated to account for specific test results on short assemblies, the particulars of the axial flow, and the phenomena governing the dynamics of these designs. This formulation is addressed within this document in Appendix 2 and also in *ANP-3591P, Revision 0, AREVA Responses to NRC RAI 8736 (Questions 29611, 29613-29616) regarding TR-0716-50351, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces"* (Reference 7.0-4) (Question 29611).

L&C #6 Discussion:

L&C #6 requests that the fuel rod assessment under faulted conditions be demonstrated.

6. *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.*

The fuel rod analysis is part of the component stress evaluation that was performed for the NuScale fuel design.

L&C #7 Discussion:

L&C #7 requires that when bounding stress analysis of the non-grid components is used, without regard to specific core location, the more stringent limits for control rod locations must be used:

7. *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.*

The margin calculations for the NuScale fuel assembly guide tubes were performed using ASME Service Level C stress limits, which are applicable to control rod locations, therefore L&C #7 is fulfilled.

L&C #8 Discussion:

L&C #8 requires that, in the case when **[** **]**:

8. *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).* **[** **]**.

The NuScale component stress analysis was performed using a 3-D load combination as discussed in Reference 7.0-3. Therefore, L&C #8 is not a concern.

L&C #9 Discussion:

L&C #9 places a restriction over the range of applicability of **[** **]**:

9. **[** **]**.

This point has been addressed in Reference 7.0-3. The NuScale grid design is the same as the grid in the generic fuel assembly used in the Sample Problem (Appendix B) of ANP-10337P-A. The limitation of L&C #9 has been met.

3.3 NuScale Design Differences and Requirements

To extend the applicability of ANP-10337P-A to include NuScale fuel, the following design differences are addressed:

- NuScale fuel assembly is shorter than typical PWR designs presented in Table 2-1 of ANP-10337P-A.

- The contribution of axial coolant flow to the NuScale fuel assembly damping is expected to be much less than that for other operating PWRs, and thus, the damping values presented in Section 6.1.3 of ANP-10337P-A are not applicable to NuScale fuel.
- In addition to the evaluation of the fuel during operation in the reactor, the NuScale analysis includes a seismic evaluation in which the core is residing in the RFT during refueling operations.

3.3.1 Fuel Assembly Length and Number of Spacer Grids

Although not stated as defining a range of applicability, Table 2-1 of ANP-10337P-A illustrates typical PWR designs to which ANP-10337P-A can be expected to be applied. The NuScale fuel assembly is outside the range of parameters in Table 2-1 in terms of fuel assembly length (shorter) and number of grids (fewer).

- The shorter assembly length of the NuScale fuel design will result in unique dynamic properties of the fuel assembly (i.e., higher stiffness and higher natural frequencies). However, this difference in design is captured in the method defined in ANP-10337P-A because the method requires that fuel assembly models be built to match design-specific experimental dynamic characterization of the fuel design. The expected differences in the dynamic properties of the NuScale fuel assembly due to its shorter length are directly characterized through full-scale prototype testing and the models built to match this tested behavior had negligible error. The application of this method to NuScale with its shorter length is discussed in more detail in Section 3.3.4 and Appendix A.
- The designs in Table 2-1 of ANP-10337P-A have between five and nine intermediate spacer grids, whereas the NuScale fuel design has three intermediate spacer grids. The NuScale fuel assembly has a total of five spacer grids, but following the modeling architecture defined in Section 5.2.1 of ANP-10337P-A, the uppermost and lowermost end grids are not modeled explicitly [

  ] As a result, the NuScale fuel assembly model will be represented as a single beam with three rotational nodes at the intermediate grid locations. With three non-fixed degrees of freedom, the model is only capable of accurately representing the fuel assembly response up to the third mode, consistent with the limitations of the experimental testing of the NuScale fuel assembly, in which it is only practical to characterize assembly frequencies up to the third mode (see Section 3.3.2 below). [

  ] The application of this fuel assembly model, with three rotational nodes, is demonstrated in Section 3.3.4 and Appendix A of this report and shows negligible error to tested results. Additional studies have demonstrated that results from this modeling approach reflect an appropriate level of mass participation in the dynamic response.

Therefore, with regard to the shorter fuel assembly length and fewer spacer grids, ANP-10337P-A remains applicable to NuScale fuel without modifications.

3.3.2 Deleted

This section is no longer needed.

3.3.3 Fuel Assembly Damping

Section 6.1.3 of ANP-10337P-A defines fuel assembly damping values that are generically applicable to standard PWR fuel designs. However, relative to a standard PWR, the NuScale design will operate with a shorter fuel assembly and reduced flow rates. For these reasons, the damping values defined in Section 6.1.3 of ANP-10337P-A are not applicable to the NuScale design.

Section 5.0 and Appendix B provide details regarding the establishment of NuScale-specific fuel assembly damping values. For the NuScale design, the maximum fuel assembly damping ratio values to be used for the analysis of seismic and LOCA events in place of those defined in Section 6.1.3 of ANP-10337P-A are defined in Table 3-1. These damping values do not credit the additional contribution of damping in flowing water.

Table 3-1. NuScale Fuel Assembly Damping Ratio Values

[ ]

3.3.4 Reactor Flange Tool Seismic Analysis

3.3.4.1 Model Adjustments

As mentioned in Section 3.1 above, when the reactor core is installed in the RFT, the fuel assembly seismic analysis is performed at ambient temperature. No LOCA analysis is performed because the LOCA accident does not exist in the RFT condition.

The horizontal seismic analysis of the RFT follows the same general procedure as defined in ANP-10337P-A. The single fuel assembly model is benchmarked at room temperature based on the free vibration and forced vibration experimental data (Section 6.1.1 of ANP-10337P-A). The equivalent stiffness (K_{EQ}) and equivalent damping (C_{EQ}) for the spacer grid impact spring are benchmarked at room temperature (Section 6.1.2.2 of ANP-10337P-A). The benchmarked assembly model and impact spring properties are thus unchanged for the RFT seismic analysis. These benchmarked models are applied directly in the analysis without further scaling for higher temperatures, as defined throughout Section 6 of ANP-10337P-A.

Relative to Section 6.1.2.1.1 of ANP-10337P-A, [

]

The fuel assembly damping ratios used in the RFT analysis are consistent with the process defined in Section 5.0 and Appendix B. The in-water damping ratios derived from experiment at [

] The structural damping ratios were measured at 70 degrees F, so no adjustment for temperature is necessary. Flowing water is not credited when the fuel is in RFT conditions and therefore, like the damping values presented in Section 3.3.3, the RFT damping values presented in Table 3-2 only include structural and quiescent water components.

Table 3-2. NuScale Fuel Assembly Damping Ratio Values for RFT Analysis at 70°F

[ ]

The vertical seismic analysis for RFT also follows the same general procedure as defined in ANP-10337P-A. The vertical model is benchmarked at room temperature to axial drop experimental data, and the model parameters are unchanged for the RFT analysis. Therefore, the temperature scaling operations defined in Section 6.2 of ANP-10337P-A are not required for the RFT analysis.

3.3.4.2 Regulatory Requirements

The regulatory requirements defined in Section 3 of ANP-10337P-A are primarily concerned with shutting down the reactor safely and maintaining a safe-shutdown condition. When placed in the RFT, the reactor is already near ambient temperature with control rods inserted; thus, the coolability and control rod insertion requirements are met. Therefore, no design margin calculations are made for the spacer grid impact force, guide tube buckling, or guide tube stress.

The remaining regulatory requirement addresses mechanical fracture of the fuel rod by external forces. The fuel rod loads from the horizontal and vertical seismic events are calculated per Section 8 of ANP-10337P-A for inclusion in the fuel rod faulted stress evaluation.

4.0 NuScale Fuel Characterization

Two areas of fuel characterization are reviewed to provide an explicit demonstration of the application of ANP-10337P-A to NuScale fuel. These two items are reviewed in detail in Appendix A, but a summary is provided in this section.

- Section 2.2 of ANP-10337P-A requires an explicit demonstration of the applicability of grid impact modeling elements.
- Section 3.3 of this document notes that the NuScale fuel assembly is outside the range of typical PWR designs to which ANP-10337P-A is applied in terms of both overall fuel assembly length and the number of spacer grids. The application of the single fuel assembly model described in Section 5.2.1 of ANP-10337P-A to the NuScale design, with a shorter overall length and fewer spacer grids, is demonstrated.

4.1 Spacer Grid Behavior

Under lateral impacts over the range of application, Section 2.2 of ANP-10337P-A specifies that [

]

The NuScale fuel design utilizes the same 17x17 HTP™ spacer grid that is currently in use in operating plants. Thus, this behavior is well established for this existing grid design. Figure A.2-1 and Figure A.2-2 in Appendix A demonstrate [

]

4.2 Single Fuel Assembly Model

The applicability of the single fuel assembly model, as defined in Section 5.2.1 of ANP-10337P-A, to the NuScale fuel design is addressed in this section. This section shows the ability of a benchmarked fuel assembly model to replicate a frequency that characterizes test data from free and forced vibration testing.

The free vibration test is performed in order to characterize the primary, or first mode, natural frequency of the fuel assembly. The NuScale fuel assembly was tested over a range of deflections from [

] For the non-irradiated (BOL) assembly, the frequency ranges from a [ ] Likewise, the simulated-irradiated (EOL) assembly frequency ranges from a [ ] The EOL condition is more representative of the bulk of the fuel in the NuScale core at any given time, other than the initial startup.

The forced vibration test provides complementary information to the free vibration test by providing a measurement of natural frequencies at higher modes, as well as the primary frequency, albeit, at smaller deflection amplitudes. For the NuScale fuel assembly, the forced vibration test indicates [ ]

Individual BOL and EOL single fuel assembly models are benchmarked to match [ ] following the process defined in Section 6.1.1 of ANP-10337P-A. Table A.3-1 and Table A.3-2 in Appendix A of this report demonstrate that the resulting single fuel assembly models are capable of replicating [ ] measured from the fuel assembly dynamic testing with negligible errors. Therefore, the single fuel assembly model described in Section 5.2.1 of ANP-10337P-A is applicable to the NuScale fuel design.

5.0 NuScale Damping Characteristics

During an external excitation (seismic or LOCA), a PWR fuel assembly experiences the following three sources of energy dissipation when in-core:

- In-air, structural damping. This loss of energy is related primarily to the energy dissipation at the contact interface between fuel assembly components, including the boundary conditions, and is measured in air.
- Quiescent, viscous water damping. This loss is due to irrecoverable pressure losses during the movement of the fuel assembly through water.
- Axial coolant flow damping. This energy dissipation source is related to the hydrofoil effect observed with lateral structural motions in axial flow conditions.

These sources of damping are quantified for the NuScale fuel design in both the non-irradiated and irradiated conditions to replace the generic damping values defined in Section 6.1.3 of ANP-10337P-A.

5.1 Definition of NuScale Damping Values

The different sources of damping can be independently measured by performing free vibration and forced vibration tests on fuel assemblies in air and in water. For NuScale, these sources of damping are based on in-air tests performed on NuScale prototype fuel assemblies, and four additional in-water tests performed on geometrically similar fuel assemblies. Analytical formulations are used to extend in-water test results to the NuScale fuel design.

The final value for the damping ratio of the NuScale fuel assembly is based on the following two components: in-air, structural damping and quiescent, viscous water damping. Each of these components is addressed in more detail below.

<u>In-air, structural damping</u>. The damping ratio is calculated using the free vibration and forced vibration tests that were performed on NuScale non-irradiated and simulated-irradiated fuel assemblies. [

]

[

]

<u>Quiescent, viscous water damping (drag)</u>. Data from in-water tests performed on assemblies that have the same cross-sectional geometry as the NuScale fuel design are used to derive the damping ratio value of the NuScale fuel assembly for in-water

conditions (zero flow). [

]

[ ]

[

]

Applying this approach, the in-water damping ratio component [

] These values are based on experiments performed at [
].

The methodology used to establish damping values for the NuScale fuel assembly does not credit the effect of flowing water. This conservatism in the method has been quantified using the same method applied for the quiescent water damping and can be shown to under-represent the damping ratio. Specifically, this component can be shown to [

]

[

]

5.2 Summary of NuScale Damping Values

A summary of the damping ratio values [ ] is given in Table 5-1. [ ]

Table 5-1. Summary of NuScale fuel assembly damping ratios

[ ]

Table 5-2. Summary of NuScale fuel assembly damping ratios for the RFT at 70°F

[ ]

The damping ratio for the NuScale fuel assembly can be dependent on the amplitude at which the fuel assembly is oscillating. In general, the quiescent water damping tends to increase with amplitude while the in-air damping component is non-linear. [ ]

[

]

6.0 Summary and Conclusions

ANP-10337P-A defines a generic methodology for performing the evaluation of the fuel assembly structural response to externally-applied forces (i.e., seismic and LOCA) that is generically applicable to all PWRs. This methodology is applicable to the NuScale design with the following modifications:

- The methodology uses NuScale specific fuel assembly damping values. This modification has been defined and justified within this report.
- In addition to the evaluation of the fuel during operation in the reactor, the NuScale analysis includes a seismic evaluation in which the core is residing in the RFT during refueling operations.

With this modification, ANP-10337P-A is applicable to the NuScale fuel design.

7.0 References

7.0-1 AREVA Inc., “PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations,” ANP-10337P-A Rev. 0, April 2018.

7.0-2 U.S. Nuclear Regulatory Commission, “Standard Review Plan, Fuel System Design,” NUREG-0800, Chapter 4, Section 4.2, Rev. 3, March 2007.

7.0-3 ANP-3712P-000, Framatome Responses to NRC RAI No. 9555 regarding NuScale Topical Report TR-0816-51127.

7.0-4 ANP-3591P, Revision 0, AREVA Responses to NRC RAI 8736 (Questions 29611, 29613-29616) regarding TR-0716-50351, “NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces.”

Appendix A. Review of NuScale Fuel Characterization Test Data Applicability to ANP-10337P-A

A.1 Introduction

The purpose of this appendix is to provide a review of NuScale fuel characterization test data in order to demonstrate the behavior necessary to confirm the applicability of ANP-10337P-A. Specifically, Section 2.2 of ANP-10337P-A requires an explicit demonstration of the applicability of grid impact modeling elements. In addition, Section 3.3 of this report notes that the NuScale fuel assembly is outside the range of typical PWR designs to which ANP-10337P-A is applied in terms of both overall fuel assembly length and the number of spacer grids. This appendix also demonstrates the application of the single fuel assembly model described in Section 5.2.1 of ANP-10337P-A to the NuScale design, with a shorter overall length and fewer spacer grids.

A.2 Spacer Grid Behavior

Under lateral impacts over the range of application, Section 2.2 of ANP-10337P-A specifies that [

]

The NuScale fuel design utilizes the same 17x17 HTP™ spacer grid that is currently in use in operating plants. Thus, this behavior is well established for this existing grid design. The data to be reviewed here are generic and not specific to NuScale.

Figure A.2-1 and Figure A.2-2 demonstrate [

] Figure A.2-1 and Figure A.2-2 present this relationship for single grids, but they are representative of the same behavior seen over the total population of tested spacer grids. [

]

Figure A.2-1. [
]

Figure A.2-2. [
]

A.3 Single Fuel Assembly Model

The applicability of the single fuel assembly model, as defined in Section 5.2.1 of ANP-10337P-A, to the NuScale fuel design is addressed in this section. The desired result from this demonstration is to show the ability of a benchmarked fuel assembly model to replicate a frequency that characterizes test data from free vibration and forced vibration testing.

The cross-sectional geometry of the NuScale fuel design is the same as that of other designs that are currently operating. The only unique characteristics of the NuScale fuel design are its shorter length and fewer spacer grids.

The NuScale fuel assembly was subjected to the program of characterization testing as summarized in Table 6-4 of ANP-10337P-A. The free vibration and forced vibration tests are of particular importance to the creation of a lateral single fuel assembly model. These tests provide information regarding the natural frequencies of the fuel assembly. A plot of first mode frequency versus deflection amplitude from free vibration testing is presented for non-irradiated (BOL) and simulated-irradiated (EOL) assemblies in Figure A.3-1.

Figure A.3-1. NuScale fuel assembly first-mode frequency versus deflection amplitude, non-irradiated (BOL) and simulated-irradiated (EOL), ambient conditions

As can be seen from Figure A.3-1, the NuScale fuel assembly was tested [

] For the BOL assembly, the frequency ranges from a [

]

In comparison to the BOL assembly behavior, the EOL assembly [

] Furthermore, the EOL condition is more representative of the bulk of the fuel in the NuScale core at any given time, other than the initial startup. For the EOL assembly, the frequency ranges from [

]

The establishment of the mid-point frequency as the benchmark target is consistent with the process illustrated in Figure 6-1 of ANP-10337P-A. The deflection amplitudes corresponding to the BOL and EOL mid-point frequencies are different, but the mid-point frequency results in a more representative characterization of the fuel assembly natural frequency.

Forced vibration testing was also performed on the NuScale fuel assembly. This testing was performed at deflection amplitudes that were smaller than those tested in free vibration, due to the large amount of energy required to simultaneously excite multiple modes of the fuel assembly. However, while the free vibration test only provides a measurement of the first mode frequency, the forced vibration testing provides a measurement of natural frequencies at higher modes, as well as the primary frequency. Specifically, the forced vibration testing demonstrated [

]

Individual BOL and EOL single fuel assembly models were benchmarked to match [

] following the process defined in Section 6.1.1 of ANP-10337P-A. The results of the benchmarking process are presented in Table A.3-1 and Table A.3-2.

Table A.3-1. NuScale fuel assembly frequency benchmark results, non-irradiated condition

[ ]

Table A.3-2. NuScale fuel assembly frequency benchmark results, simulated-irradiated condition

[ ]

Table A.3-1 and Table A.3-2 demonstrate that the resulting single fuel assembly models are capable of replicating [ ] measured from the fuel assembly dynamic testing with negligible errors. Therefore, the single fuel assembly model described in Section 5.2.1 of ANP-10337P-A is applicable to the NuScale fuel design.

A.4 Conclusions / Summary

This appendix has demonstrated the applicability of both the grid impact model and the fuel assembly dynamic model from ANP-10337P-A to the NuScale fuel design.

Appendix B. NuScale Damping for Lateral Accident Condition Analysis

B.1 Introduction

During an external excitation (seismic or LOCA), a PWR fuel assembly experiences the following three sources of energy dissipation:

- In-air, structural damping. This loss of energy is related primarily to the energy dissipation at the contact interface between fuel assembly components, including the boundary conditions, and is measured in air.
- Quiescent, viscous water damping. This loss is due to irrecoverable pressure losses during the movement of the fuel assembly through water.
- Axial coolant flow damping. This energy dissipation source is related to the hydrofoil effect observed with lateral structural motions in axial flow conditions.

This appendix determines these damping ratio values for the NuScale fuel design in both the non-irradiated and irradiated conditions.

B.2 Methodology

The different sources of damping can be independently measured by performing free vibration and forced vibration tests on fuel assemblies in air and in water. For NuScale, these sources of damping are based on in-air tests performed on NuScale prototype fuel assemblies and in-water tests performed on fuel assemblies that have the same cross-sectional geometry. Analytical formulations are used to extend in-water test results to the NuScale fuel design.

A broad set of experimental datasets are used to establish damping ratios for the NuScale fuel assemblies. A description of these experiments is provided below:

- The NuScale tests involve two NuScale 17x17 prototype fuel assemblies with HTP™ spacer grids (non-irradiated and simulated-irradiated) subjected to free vibration and forced vibration tests in air. The results from these tests provide a direct measurement of in-air, structural damping ratios for NuScale fuel.
- The MASSE tests were performed using [ ] Both free vibration and forced vibration tests were performed in both air and water, including the effects of flowing water. These tests are used to support the in-water damping ratio of the NuScale fuel assembly in the non-irradiated condition.
- The CAMEOL tests were performed using [ ] Forced vibration tests were performed in both air and water, including the effects of flowing water.

These tests are used to support the in-water damping ratio of the NuScale fuel assembly in the irradiated condition.

- The MALDIVE tests were performed using [ ] Both free vibration and forced vibration tests were performed in both air and water, including the effects of flowing water.
- The MARITIME tests were performed using [ ] Both free vibration and forced vibration tests were performed in air. These test results were reviewed to provide additional insight on the structural damping ratios of short assemblies and validate the NuScale fuel assembly damping ratios.

The final value for the damping ratio of the NuScale fuel assembly is based on the following two components, and the method used to quantify these components is provided:

- In-air, structural damping. The damping ratio is calculated using the free vibration and forced vibration tests that were performed on NuScale non-irradiated and simulated-irradiated fuel assemblies. [ ]
- Quiescent, viscous water damping (drag). [ ]

[ ]

where

[

]

The methodology used to derive damping for the NuScale fuel assembly does not credit the effect of flowing water. This conservatism in the method has been quantified and can be shown to under-represent the damping ratio of the fuel by more than **[**

]

The damping ratio for the NuScale fuel assembly can be dependent on the amplitude at which the fuel assembly is oscillating. **[**

]

[

]

B.3 Results

B.3.1 Structural damping ratio

The structural damping ratio measured from the experiment for the NuScale fuel assembly (BOL and EOL) is shown in Figure B.3-1.

[ ]

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)

[ ]

[ ]

[ ]

[ ]

[

]

[ ]

B.3.2 Quiescent, viscous water damping (zero flow)

Through the experimental campaigns defined in Section B.2, it has been demonstrated that when a fuel assembly is immersed in water and is subjected to an external excitation, an additive damping component can be measured due to the movement of the fuel through a dense fluid. This additional damping component will increase with an increase in the amplitude of oscillation.

[

]

[ ]

[

]

[ ]

[ ]

[ ]

[ ]

B.3.3 Axial Coolant Flow Damping

Beyond the damping components discussed thus far, there is an additional source of damping that is associated with the axial flow of coolant past the bundle. This additional damping has been demonstrated through numerous experimental campaigns and can be quantified for NuScale. Using the MASSE, CAMEOL, and MALDIVE test data, and the same method employed in Section B.3.2, a damping ratio can be quantified for the NuScale flow conditions. This component can [ ] However, for conservatism, the effects of coolant flow will not be credited in the total damping ratio for the NuScale fuel assembly.

B.4 Summary of Fuel Assembly Damping Ratios for NuScale

A summary of the damping ratio values [ ] is given in Table B.4-1. [ ]

Table B.4-1. Summary of NuScale fuel assembly damping ratios

[ ]

Table B.4-2. Summary of NuScale fuel assembly damping ratios for the RFT at 70°F

[ ]

[ ]

[

]

Section C

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 2:

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

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.

NUSCALE POWER™

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)

[ ]

[ ]

[ ]

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

Note for Figure 2: *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

NuScale Power

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 **[**

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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

April 30, 2020

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 Submittal of Corrected Response to NRC Request for Additional Information No. 284 (eRAI No. 9225) on the NuScale Design Certification

REFERENCES:

1. U.S Nuclear Regulatory Commission, "Request for Addition Information No. 284 (eRAI No. 9225)," dated November 22, 2017 (ML1335A107)

2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 284 (eRAI No. 9225)," dated July 19, 2019 (Letter - ML19200A195, Package – ML19200A194)

The purpose of this letter is to provide the NuScale Power, LC (NuScale) corrected response to the referenced NRC Request for Additional Information (RAI), as discussed with Bruce Bavol of the NRC Staff on April 29, 2020.

The enclosure to this letter contains NuScale's corrected response to the following RAI Questions from NRC eRAI No. 9225:

- 04.02-8

The corrected response does not modify any of the language in the response to the RAI, but corrects proprietary markings and redacted information in the RAI response. This corrected response replaces the response provided in Reference 2. Therefore, NuScale requests that the NRC remove the RAI 9225 package located at ML19200A194 with this corrected response.

Enclosure 1 is the proprietary version of the Corrected NuScale Response to NRC RAI No. 284 (eRAI No. 9225). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.90. The enclosed affidavit (Enclosure 3) pertains to Framatome proprietary information to be withheld from the public. Framatome proprietary information is denoted by straight brackets (i.e. "[ ]"). Enclosure 2 is the nonproprietary version of the NuScale response.

The technical report "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," TR-0816-51127 contained export information. The markup pages in the enclosed RAI response for TR-816-51127 are therefore labeled "Export Controlled," although these markup pages do not contain any export controlled information.

NuScale Power, LLC
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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, please contact Matthew Presson at 541-452-7531 or at MPresson@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Prosanta Chowdhury, NRC, OWFN-8H12
Bruce Bavol, NRC, OWFN-8H12

Enclosure 1: NuScale Corrected Response to NRC Request for Additional Information eRAI No. 9225, proprietary version
Enclosure 2: NuScale Corrected Response to NRC Request for Additional Information eRAI No. 9225, nonproprietary version
Enclosure 3: Framatome Affidavit of Gayle Elliott

Enclosure 2:

NuScale Corrected Response to NRC Request for Additional Information eRAI No. 9225, nonproprietary version

Note: The RAI response to RAI 9225 in NuScale letter RAIO-0420-69855 contained markups to the "NuFuel - HTP2^{TM} Fuel and Control Rod Assembly Designs" technical report, TR-0816-51127 and DCA, Section 4.2, "Fuel System Design." The additional markups are not relevant to this approved topical report and therefore, have not been provided.

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Power for all humankind

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

eRAI No.: 9225

Date of RAI Issue: 11/22/2017

NRC Question No.: 04.02-8

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 used to analyze the loads.

In RAI 8769 Question 1, the staff requested information regarding fuel assembly structural response when in approved locations outside of an operating bay. As part of its response to this question, NuScale provided a Table which compared upper core plate motions when the fuel is located in the RFT as compared with the motions when the fuel was located in an operating bay. While the results indicate that the upper core plate motions in an operating bay bound those found when the fuel is located in the RFT, the staff notes that the RFT design is not currently finalized, as indicated by staff RAI 8838. Therefore, the staff does not consider Table 1 of the response to RAI 8769 Question 1 to be final.

Update the core plate motions provided in the response to RAI 8769 Question 1 Table 1 with values derived from a final RFT design, or provide justification to explain how the values presented are bounding for all potential RFT designs.

NuScale Response:

A separate fuel seismic analysis was performed to assess the fuel assembly structural response when located in the reactor flange tool (RFT).

As a first step to considering the response of fuel to a seismic event while in the (RFT, it is important to understand the unique conditions of the RFT relative to the NuScale Power Module (NPM). These unique conditions must be considered not only to determine how to appropriately model the fuel dynamic response in the RFT, but also to determine the safety requirements and the acceptance criteria for the fuel while it resides in the RFT.

Seismic Analysis Methodology for the RFT Condition

Regarding the dynamic behavior of the fuel, the key similarity between the RFT and NPM conditions is that the fuel remains in the core cavity, completely submerged in water. In this condition, the fuel is held fixed at the upper and lower core plates exactly as it would be when the reactor is in operation in the NPM. Overall, the dynamic representation of the fuel in the RFT is identical to the operating condition in the NPM, with the only difference being the lower temperature and very low flow. Therefore, it is appropriate to analyze the fuel using the same methods applied for the NPM analysis with consideration of the change in environmental conditions.

The dynamic models of the fuel, as defined in Reference 1 and 2, are adjusted for the reduction in temperature from the NPM (nominally 546 degrees F) to the RFT (65 degrees F to 110 degrees F) to account for changes in the elastic modulus of the structure, changes in thermal expansion, and changes in fuel assembly damping. The primary effects of the lower temperature in the RFT are a contraction of the core causing smaller gaps and a stiffening of the assembly. Under the same level of excitation, these effects lead to higher loads in the simulations. Therefore, the RFT simulations are performed at the low end of this temperature range at 70 degrees F. This temperature coincides with the conditions of the dynamic tests upon which the fuel assembly models are based. The decrease in temperature also results in a small increase in the fuel assembly damping, based on the same temperature scaling relationship defined in Reference 2 (Equation C-1). Like the NPM evaluation, the RFT analysis only considers the effects of still water; increased damping associated with flowing water is not credited.

Regulatory Requirements and Acceptance Criteria

Following the regulatory framework established by 10 CFR Part 50, Appendix A and Appendix S, and the guidance provided in NUREG-0800, Chapter 4.2, Appendix A, the objective of performing an evaluation of the fuel in response to a safe shutdown earthquake is to demonstrate the integrity of the reactor coolant boundary, the capability to shut down the reactor, maintain it in a safe-shutdown condition, and the capability to prevent or mitigate the consequences of accidents that could result in offsite exposures. This last element includes the preservation of a coolable geometry of the fuel assembly for post-seismic conditions. In contrast to the NPM condition, the coolable geometry and the shutdown condition are already inherently satisfied in the RFT condition. For this reason, the evaluation of the fuel in the RFT condition centers on satisfying the ability to prevent or mitigate the consequences of accidents that could result in offsite exposure. This is satisfied in the RFT analysis by focusing on the structural integrity of the fuel rod cladding during and after a seismic event. For this purpose, the same acceptance criteria applied to the fuel rod cladding for the NPM analysis will be conservatively applied to the RFT analysis. Use of the same criteria is conservative because it does not credit the added strength at colder conditions and structural integrity of the fuel rod cladding can be met with an acceptance criteria greater than what is used for the NPM analysis.

Analysis Results

Evaluating the RFT time histories in the dynamic model provides grid impact loads, vertical impact loads at the bottom nozzle, and bending stresses induced by fuel assembly deflections. These results are processed to calculate a combined stress state for the fuel rod cladding as shown in Table1.

Table 1: RFT Fuel Rod Stress Margins

[

]

Note [1]: The fuel rod cladding temperature is significantly lower in the RFT than the NPM. Correspondingly, the material strength at the lower temperatures of the RFT will be higher than in the NPM. The RFT margins are conservatively calculated using the lower strength value based on the NPM temperature.

Although loads are not directly comparable between the RFT and NPM cases because of the difference in temperature, the results of the lateral and vertical seismic analyses for the RFT indicate a condition that is generally less severe than the fuel response in the NPM. In addition, all fuel assembly components will benefit from an increase in material strength at the lower temperature in the RFT, thereby furthering the observation of a more benign seismic event in the RFT relative to the NPM.

Conclusions

The evaluation of the fuel assembly structural response to a seismic event while residing within the RFT has demonstrated that the structural integrity of the fuel rod cladding is maintained, thus satisfying the regulatory requirement to prevent or mitigate the consequences of accidents that could result in offsite exposures. In general, the results of the RFT analysis demonstrate a more benign event for the fuel relative to the condition analyzed when the fuel is in the NPM.

References

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

2. ANP-10337P-A, Rev. 0, “PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations,” April 2018

Impact on DCA:

FSAR section 4.2 and Topical Report, TR-0716-50351, NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces, and related Technical Report TR-0816-51127, NuFuel-HTP2 Fuel and Control Rod Assembly Designs, have been revised as described in the response above and as shown in the markup provided with this response.

1.3 Abbreviations

Table 1-1. Abbreviations

Term	Definition
BOL	Beginning of Life
CFR	Code of Federal Regulations
EOL	End of Life
°F	Degrees Fahrenheit
Ft/s	Feet per Second
GDC	General Design Criteria
ID	Inner Diameter
Kg U	Kilograms of uranium
kW/m	Kilowatts per meter
LOCA	Loss-of-Coolant Accident
MWt	Megawatt thermal
NRC	Nuclear Regulatory Commission
OD	Outer Diameter
Psia	Pounds per Square Inch – Absolute
Psig	Pounds per Square Inch – Gauge
PWR	Pressurized Water Reactor
R^2	Coefficient of determination
RCS	Reactor Coolant System
RFT	reactor flange tool
SER	Safety Evaluation Report
SRP	Standard Review Plan

Table 2-1. NuScale fuel design parameters

Parameter	NuScale Fuel Design	AREVA 17x17 PWR
Fuel rod array	17 x 17	17 x 17
Fuel rod pitch (inch)	0.496	0.496
Fuel assembly pitch (inch)	8.466	8.466
Fuel assembly height (inch)*	94.0	159.45
Number of guide tubes per bundle	24	24
Dashpot region inner diameter (inch)	0.397	0.397
Dashpot region outer diameter (inch)	0.482	0.482
Inner diameter above transition (inch)	0.450	0.450
Outer diameter above transition (inch)	0.482	0.482
Number of instrument tubes per bundle	1	1
ID (inch)	0.450	0.450
OD (inch)	0.482	0.482
Number of fuel rods per bundle	264	264
Cladding outer diameter (inch)	0.374	0.374 and 0.376
Cladding inner diameter (inch)	0.326	0.326
Length of total active fuel stack (inch)*	78.74	144
Fuel pellet OD (inch)	0.3195	0.3195
Fuel pellet density (% theoretical density)	96	96
Spacer grid span lengths (inch)	20.1	20.6
Fuel rod internal pressure (psig)	215	315

* Height is measured from the seating surface of the bottom nozzle to the top of the post on the top nozzle. Dimension does not include hold-down springs.

Table 2-2 provides the operating conditions representative of the NuScale design.

Table 2-2. NuScale operating conditions

Parameter	NuScale Design Value	AREVA 17x17 PWR Value
Rated thermal power (MWt)	160	3455
Average coolant velocity (ft/s)	3.1	16
System pressure (psia)	1850	2280
Core tave (°F)	547	584
Linear heat rate (kW/m)	8.2	18.0
RCS inlet temperature (°F)	503	547
RCS Reynolds Number	76,000	468,000
Fuel assemblies in core	37	193
Fuel assembly loading (kgU)	249	455
Core loading (kgU)	9,213	87,815

In addition to the operating condition given in Table 2-2, the NuScale design also includes the condition where the reactor core is placed on the Reactor Flange Tool (RFT) during refueling activities. When the core is in this configuration, the fuel assemblies remain in the core cavity with the same lower and upper core plate engagement as during operation. The core is submerged in water at a temperature range of 65°F to 110°F and the control rods are completely inserted.

2.1 Regulatory Requirements

Section 3.0 of ANP-10337P identifies the regulatory requirements and guidance addressed by the generic methodology. Specifically, the methodology addresses the following requirements as they relate to the structural requirements of a fuel assembly subjected to externally applied loads from earthquakes and postulated pipe breaks:

- GDC 2 – Design bases for protection against natural phenomena
- GDC 27 – Combined reactivity control systems capability
- GDC 35 – Emergency core cooling
- 10 CFR Part 50 Appendix S – Earthquake engineering criteria for nuclear power plants
- 10 CFR 50.46 – Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors

The methodology addresses these regulatory requirements consistent with guidance in SRP Section 4.2 (Reference 2).

This topical report demonstrates that the regulatory requirements and demonstration methods defined in ANP-10337P are applicable to the NuScale fuel design.

design is identical to the grid cited in the sample problem for ANP-10337P, this potential limitation on the applicability of the methodology is satisfied.

In summary, the NuScale design is consistent with the general conditions defined in Chapter 2 of ANP-10337P. Therefore, the range of applicability defined in this chapter encompasses the NuScale design. The differences in assembly length and number of spacer grids are addressed in Section 3.3 of this report. Compliance to the requirement defined in Section 2.2 of ANP-10337P is demonstrated in Section 3.3.4 and Appendix A.

Chapter 3 Regulatory Requirements

Chapter 3 reviews the regulatory requirements that are relevant to this methodology. These requirements include Appendix A (GDC 2, 27, and 35) and Appendix S of 10 CFR Part 50 and 10 CFR 50.46. In addition, this chapter reviews the NRC guidance from the Standard Review Plan Section 4.2 that pertains to these requirements (primarily Appendix A).

The requirements of Appendices A (GDC 2, 27, and 35), and S of 10 CFR Part 50, 10 CFR 50.46 are directly applicable to the NuScale design. The guidance of SRP Section 4.2 Appendix A is directly applicable and is implemented in ANP-10337P.~~ The discussion of 10 CFR Part 50 Appendix S is applicable with the following minor clarification:~~

- ~~10 CFR 52.47(a)(2)(iv) specifies the requirements for offsite radiological consequence analyses, including exposure limits, for design certification applicants, as opposed to 10 CFR Part 100, 10 CFR 50.34, and 10 CFR 50.67 as stated in ANP-10337P. However, the conclusion that fuel rod failures are permitted during postulated accidents and must be accounted for in the dose analysis remains applicable. This minor clarification does not affect the applicability of the method.~~

The same regulatory requirements identified in ANP-10337P Chapter 3 are applicable to the NuScale Power Module; therefore, this chapter is applicable to the NuScale design.

When the reactor is placed in the RFT during refueling operations, some of the regulatory requirements are already inherently met. In this case, the LOCA event is not considered and control rod insertion and coolable geometry are already satisfied since the reactor is at ambient temperature conditions. The requirement that fuel rod mechanical fracture will not occur due to seismic loads remains applicable.

Chapter 4 Acceptance Criteria

Chapter 4 establishes the appropriate selection of acceptance criteria in order to satisfy the regulatory requirements specified in Chapter 3. In general, this chapter establishes criteria to evaluate spacer grid impact loads and allowable stresses for non-grid components.

Like the regulatory requirements in Chapter 3, these criteria are generic to PWR fuel. The NuScale fuel design uses the same components and structure as the PWR designs presented in Chapter 2 of ANP-10337P; therefore, the criteria defined in this chapter can application. The remaining parameters require characterization testing in order to be defined. The full design characterization testing program identified in Table 6-4 of Chapter 6 in ANP-10337P will be applied to the NuScale fuel design. Differences in fuel assembly behavior due to the shorter length or fewer number of spacer grids, as noted in Chapter 2, will be characterized and accommodated as a result of the testing defined in Chapter 6.

For the seismic analysis when the reactor core is placed in the RFT, Chapter 6 of ANP-10337P is applicable, but no conversion of the model parameters to operating temperature is needed since the temperature in the RFT condition is taken as 70°F, which is in the specified range of temperatures. The lateral and vertical models, which are benchmarked at ambient temperature, are used directly. Model parameters that are defined or measured at operating temperature, such as the BOL spacer grid through grid properties, are converted to ambient conditions.

~~Two~~One item~~s~~ from Chapter 6 is~~are~~ potentially affected by the application to NuScale fuel.

- ~~Section 6.1.1.2 states that the objective of the forced vibration tests is to obtain at least the first five natural frequencies. In the case of the NuScale design, due to the shorter length and the presence of only three intermediate spacer grids, it is not necessary, nor practical, to obtain characteristics beyond the first three frequencies and mode shapes. [~~

  ~~] This difference is addressed in Section 3.3.~~
- Section 6.1.3 presents damping values to be used in the horizontal model. As noted in the review of Chapter 5, the contribution of axial coolant flow to the fuel assembly damping is expected to be much less than for other operating PWRs. As a result, the damping definition provided in Section 6.1.3 of ANP-10337P is not applicable to NuScale. This difference is addressed in Section 3.3.3~~3.3~~.

Chapter 6 of ANP-10337P is applicable to the NuScale fuel design with the exception of ~~(1) the request to experimentally characterize fuel assembly up to the first five natural frequencies, and (2)~~ the definition of damping values presented in Section 6.1.3. ~~Both of these~~This item~~s~~ is~~are~~ addressed in Section 3.3.

Chapter 7 Seismic and LOCA Analysis

Chapter 7 defines the process of applying appropriate forcing functions representing seismic or LOCA events to the models described in Chapter 5. Chapter 7 also defines the method of accounting for the combined effect of seismic and LOCA loads. In the horizontal analysis, the model calculates the time-varying displacements and impact forces for assemblies across the core. The results of this analysis are also used for calculating the resulting loads and stresses in the assembly. Similarly, the vertical model dynamics of these designs. This formulation is addressed within this document in Appendix 2 and also in Reference 4 (Question 29611).

L&C #6 Discussion:

L&C #6 requests that the fuel rod assessment under faulted conditions be demonstrated.

6. *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.*

The fuel rod analysis is part of the component stress evaluation that was performed for the NuScale fuel design.

L&C #7 Discussion:

L&C #7 requires that when bounding stress analysis of the non-grid components is used, without regard to specific core location, the more stringent limits for control rod locations must be used:

7. *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.*

The margin calculations for the NuScale fuel assembly guide tubes were performed using ASME Service Level C stress limits, which are applicable to control rod locations, therefore L&C #7 is fulfilled.

L&C #8 Discussion:

L&C #8 requires that, in the case when [ ]:

8. *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).* [ ].

The NuScale component stress analysis was performed using a 3-D load combination as discussed in Reference 3. Therefore, L&C #8 is not a concern.

L&C #9 Discussion:

L&C #9 places a restriction over the range of applicability of [ ]:

9. [ ]

[

]

This point has been addressed in Reference 3. The NuScale grid design is the same as the grid in the generic fuel assembly used in the Sample Problem (Appendix B) of ANP-10337P (Reference 1). The limitation of L&C #9 has been met.

3.3 NuScale Design Differences and Requirements

To extend the applicability of ANP-10337P to include NuScale fuel, the following design differences are addressed:

- NuScale fuel assembly is shorter than typical PWR designs presented in Table 2-1 of ANP-10337P.
- ~~The experimental characterization of the frequency response of the NuScale fuel design is limited to the first three natural frequencies, as opposed to the first five natural frequencies as requested in Section 6.1.1.2 of ANP-10337P.~~
- The contribution of axial coolant flow to the NuScale fuel assembly damping is expected to be much less than that for other operating PWRs, and thus, the damping values presented in Section 6.1.3 of ANP-10337P are not applicable to NuScale fuel.
- In addition to the evaluation of the fuel during operation in the reactor, the NuScale analysis includes a seismic evaluation in which the core is residing in the RFT during refueling operations.

3.3.1 Fuel Assembly Length and Number of Spacer Grids

Although not stated as defining a range of applicability, Table 2-1 of ANP-10337P illustrates typical PWR designs to which ANP-10337P can be expected to be applied. The NuScale fuel assembly is outside the range of parameters in Table 2-1 in terms of fuel assembly length (shorter) and number of grids (fewer).

- The shorter assembly length of the NuScale fuel design will result in unique dynamic properties of the fuel assembly (i.e., higher stiffness and higher natural frequencies). However, this difference in design is captured in the method defined in ANP-10337P because the method requires that fuel assembly models be built to match design-specific experimental dynamic characterization of the fuel design. The expected differences in the dynamic properties of the NuScale fuel assembly due to its shorter length are directly characterized through full-scale prototype testing and the models built to match this tested behavior had negligible error. The application of this method to NuScale with its shorter length is discussed in more detail in Section 3.3.4 and Appendix A.
- The designs in Table 2-1 of ANP-10337P have between five and nine intermediate spacer grids, whereas the NuScale fuel design has three intermediate spacer grids. The NuScale fuel assembly has a total of five spacer grids, but following the modeling architecture defined in Section 5.2.1 of ANP-10337P, the uppermost and lowermost end grids are not modeled explicitly [ ]

[

] As a result, the NuScale fuel assembly model will be represented as a single beam with three rotational nodes at the intermediate grid locations. With three non-fixed degrees of freedom, the model is only capable of accurately representing the fuel assembly response up to the third mode, consistent with the limitations of the experimental testing of the NuScale fuel assembly, in which it is only practical to characterize assembly frequencies up to the third mode (see Section 3.3.2 below). [

] The application of this fuel assembly model, with three rotational nodes, is demonstrated in Section 3.3.4 and Appendix A of this report and shows negligible error to tested results. Additional studies have demonstrated that results from this modeling approach reflect an appropriate level of mass participation in the dynamic response.

Therefore, with regard to the shorter fuel assembly length and fewer spacer grids, ANP-10337P remains applicable to NuScale fuel without modifications.

3.3.2 ~~Frequency Response of the NuScale Fuel~~Deleted

~~Section 6.1.1.2 of ANP-10337P establishes a requirement that the dynamic characterization testing provides the first five frequencies and mode shapes of the fuel assembly. For the NuScale fuel assembly, because of its shorter length and increased lateral stiffness, it is only practical to characterize the first three natural frequencies. In general, because of the increased lateral stiffness of the NuScale fuel assembly, the higher mode frequencies have shifted beyond the range of interest for the dynamic events that are analyzed. [~~

~~]~~

~~Therefore, with regard to the representation of the NuScale fuel, ANP-10337P remains applicable to NuScale fuel without modifications.~~ This section is no longer needed.

3.3.3 Fuel Assembly Damping

Section 6.1.3 of ANP-10337P defines fuel assembly damping values that are generically applicable to standard PWR fuel designs. However, relative to a standard PWR, the NuScale design will operate with a shorter fuel assembly and reduced flow rates. For these reasons, the damping values defined in Section 6.1.3 of ANP-10337P are not applicable to the NuScale design.

Section 5.0 and Appendix B provide details regarding the establishment of NuScale-specific fuel assembly damping values. For the NuScale design, the maximum fuel assembly damping ratio values to be used for the analysis of seismic and LOCA events in place of those defined in Section 6.1.3 of ANP-10337P are defined in Table 3-1.

These damping values do not credit the additional contribution of damping in flowing water.

Table 3-1. NuScale Fuel Assembly Damping Ratio Values

3.3.4 Reactor Flange Tool Seismic Analysis

3.3.4.1 Model Adjustments

As mentioned in Section 3.1 above, when the reactor core is installed in the RFT, the fuel assembly seismic analysis is performed at ambient temperature. No LOCA analysis is performed because the LOCA accident does not exist in the RFT condition.

The horizontal seismic analysis of the RFT follows the same general procedure as defined in ANP-10337P. The single fuel assembly model is benchmarked at room temperature based on the free vibration and forced vibration experimental data (Section 6.1.1 of ANP-10337P). The equivalent stiffness (K_{EQ}) and equivalent damping (C_{EQ}) for the spacer grid impact spring are benchmarked at room temperature (Section 6.1.2.2 of ANP-10337P). The benchmarked assembly model and impact spring properties are thus unchanged for the RFT seismic analysis. These benchmarked models are applied directly in the analysis without further scaling for higher temperatures, as defined throughout Section 6 of ANP-10337P.

Relative to Section 6.1.2.1.1 of ANP-10337P, [

]

The fuel assembly damping ratios used in the RFT analysis are consistent with the process defined in Section 5.0 and Appendix B. The in-water damping ratios derived from experiment at [ ] The structural damping ratios were measured at 70°F, so no adjustment for temperature is necessary. Flowing water is not credited when the fuel is in RFT conditions and therefore, like the damping values presented in Section 3.3.3, the RFT damping values presented in Table 3-2 only include structural and quiescent water components.

Table 3-2. NuScale Fuel Assembly Damping Ratio Values for RFT Analysis at 70°F

The vertical seismic analysis for RFT also follows the same general procedure as defined in ANP-10337P. The vertical model is benchmarked at room temperature to axial drop experimental data, and the model parameters are unchanged for the RFT analysis. Therefore, the temperature scaling operations defined in Section 6.2 of ANP-10337P are not required for the RFT analysis.

3.3.4.2 Regulatory Requirements

The regulatory requirements defined in Section 3 of ANP-10337P are primarily concerned with shutting down the reactor safely and maintaining a safe-shutdown condition. When placed in the RFT, the reactor is already near ambient temperature with control rods inserted; thus, the coolability and control rod insertion requirements are met. Therefore, no design margin calculations are made for the spacer grid impact force, guide tube buckling, or guide tube stress.

The remaining regulatory requirement addresses mechanical fracture of the fuel rod by external forces. The fuel rod loads from the horizontal and vertical seismic events are calculated per Section 8 of ANP-10337P for inclusion in the fuel rod faulted stress evaluation.

5.2 Summary of NuScale Damping Values

A summary of the damping ratio values [ ] is given in Table 5-1. [ ]

Table 5-1. Summary of NuScale fuel assembly damping ratios

[ ]

Table 5-2. Summary of NuScale fuel assembly damping ratios for the RFT at 70°F

[ ]

The damping ratio for the NuScale fuel assembly can be dependent on the amplitude at which the fuel assembly is oscillating. In general, the quiescent water damping tends to increase with amplitude while the in-air damping component is non-linear. [ ]

[

]

6.0 Summary and Conclusions

ANP-10337P defines a generic methodology for performing the evaluation of the fuel assembly structural response to externally-applied forces (i.e., seismic and LOCA) that is generically applicable to all PWRs. This methodology is applicable to the NuScale design with the following modifications:

- The methodology uses NuScale specific fuel assembly damping values. This modification has been defined and justified within this report.
- In addition to the evaluation of the fuel during operation in the reactor, the NuScale analysis includes a seismic evaluation in which the core is residing in the RFT during refueling operations.

~~one modification regarding the use of fuel assembly damping values specific to the NuScale design. This modification has been defined and justified within this report.~~ With this modification, ANP-10337P is applicable to the NuScale fuel design.

Figure A.2-1. [
]

Figure A.2-2. [
]

A.3 Single Fuel Assembly Model

The applicability of the single fuel assembly model, as defined in Section 5.2.1 of ANP-10337P, to the NuScale fuel design is addressed in this section. The desired result from this demonstration is to show the ability of a benchmarked fuel assembly model to replicate a frequency that characterizes test data from free vibration and forced vibration testing.

The cross-sectional geometry of the NuScale fuel design is the same as that of other designs that are currently operating. The only unique characteristics of the NuScale fuel design are its shorter length and fewer spacer grids.

The NuScale fuel assembly was subjected to the program of characterization testing as summarized in Table 6-4 of ANP-10337P. The free vibration and forced vibration tests are of particular importance to the creation of a lateral single fuel assembly model. These tests provide information regarding the natural frequencies of the fuel assembly. A plot of first mode frequency versus deflection amplitude from free vibration testing is presented for non-irradiated (BOL) and simulated-irradiated (EOL) assemblies in Figure A.3-1.

Figure A.3-1. NuScale fuel assembly first-mode frequency versus deflection amplitude, non-irradiated (BOL) and simulated-irradiated (EOL), ambient conditions

Table B.4-1. Summary of NuScale fuel assembly damping ratios

Table B.4-2. Summary of NuScale fuel assembly damping ratios for the RFT at 70°F

[

]

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RAIO-0818-61383

August 13, 2018

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. 9555 (eRAI No. 9555) on the NuScale Topical Report, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351, Revision 0

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9555 (eRAI No. 9555)," dated June 15, 2018
2. NuScale Topical Report, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351, Revision 0, dated September 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 Question from NRC eRAI No. 9555:

- 04.02-7

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 9555 (eRAI No. 9555). 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) pertains to the AREVA proprietary information to be withheld from the public. Framatome proprietary information is denoted by straight brackets (i.e., "[ ]"). 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 Paul Infanger at 541-452-7351 or at pinfanger@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. 9555, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9555, nonproprietary

Enclosure 3: Affidavit of Nathan E. Hottle

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Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9555, proprietary

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9555, nonproprietary

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9555
Date of RAI Issue: 06/15/2018

NRC Question No.: 04.02-7

Title 10 of the Code of Federal Regulations, Part 50, Appendix A, General Design Criterion (GDC) 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 used to analyze the loads.

Topical report TR-0716-50351-P references the Framatome topical report ANP-10377P "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations" as the methodology for analyzing the NuScale fuel assembly structural response in the NuScale reactor and ensuring compliance with GDC 2. Any licensee referencing ANP-10377P must comply with the nine conditions and limitations listed in Section 5 of the staff's safety evaluation report for ANP-10337P. The staff needs additional information to determine if the NuScale design complies with these conditions and limitations.

1. Provide information to demonstrate that the grid behavior of the NuScale fuel assembly meets ANP-10337P limitation and condition numbers 1, 2 and 9
2. Topical report ANP-10337P limitations 4, 7, and 8 are related to the methods used to calculate fuel assembly component stresses in 3D nonlinear structural models based on deflections calculated in the dynamic models.
    1. Clarify how the limiting NuScale deflection cases were selected for evaluation in 3D structural models.
    2. ANP-10337P limitation and condition number 4 limits the use of time phasing for identification of the most limiting deflection stress states to applications that are similar to the current fleet of fuel assemblies and core designs. The NuScale design is outside the current range of operational experience, so unusual dynamic behavior and unusual deflection shapes are a concern to the staff. If time phasing was used for NuScale, describe how time phasing was used to find the most limiting fuel assembly component stress states. If time phasing was not used, confirm that limitation and condition number 7 was followed.

3. Confirm that the NuScale 3D load combination was performed consistent with ANP-10337P limitation and condition number 8.

3. Topical report ANP-10337P limitation number 4 also addresses strain energy in the horizontal dynamic model. It also restricts the application of the methodology to the current fleet of fuel and core designs, and the NuScale design is outside this limitation. Provide the following information necessary for the staff to review the strain energy in the horizontal dynamic model of the NuScale fuel assembly.
    1. Identify the maximum lateral deflection and strain energy calculated over the full set of design basis horizontal analyses.
    2. Provide force-deflection data from the lateral stiffness test described in 5.1.1 of TR-0816-51127-P, Revision 1. Provide force-deflection data that encompasses the maximum deflection calculated in the model, or as high of a deflection as was recorded in the lateral stiffness test.
    3. Compare the maximum lateral deflection and strain energy calculated in the horizontal dynamic finite element model to the supporting mechanical test data. Confirm that the calculated lateral deflection is within a range where model behavior agrees with test data, or justify the use of the model outside the range where it agrees with test data.

NuScale Response:

The response is organized along the three main categories in the RAI:

1. Grid behavior compatibility with ANP-10337PA ([3]).

2. Stress analysis method compatibility with ANP-10337PA.

3. Stress analysis model validation to test data.

1.1 RAI 9555 - GRID BEHAVIOR COMPATIBILITY WITH ANP-10337PA

<u>Question 1:</u> The NRC requests clarification on the NuScale grid design and conformance with Limitations and Conditions (L&C) # 1, 2, and 9.

For convenience these three L&Cs to ANP-10337PA ([3]) are reproduced below:

L&C #1 imposes requirements on the tested behavior of grids in order to be compliant with the ANP-10337PA ([3] methodology.

1. *Dynamic grid crush tests, must be conducted in accordance with Section 6.1.2.1of ANP-10337P (as amended by RAI 16), and spacer grid behavior must satisfy the requirements in*

the TR, the key elements of which are:

a. ***[***

]

b. ***[***

]

c. ***[***

]

L&C #2 imposes requirements on the maximum allowable deformation of spacer grids:

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 TR apply:

a. For all OBE analyses, allowable spacer grid deformation is limited to design tolerances and ***[*** ***]****.*

b. For SSE, LOCA, and combined SSE+LOCA analyses, ***[***

]

L&C #9 places a restriction over the range of applicability of **[**

]

9. ***[***

]*.*

Response:

The NuScale grid design is the W-17 HTP ([2], Section 2.0), which is exactly the same grid design used in ANP-10337PA ([3]) to formulate all the grid behavior requirements. Further, this is the same grid used in the Fuel Assembly described and analyzed in the Sample Problem (Appendix B) of ANP-10337PA ([3]). As such, the NuScale spacer grids are fully compatible with the ANP-10337PA requirements and linear visco-elastic grid element validity limits. Therefore the NuScale design meets the requirements of L&C 1, 2, and 9. The fuel technical report ([1]), Table 4-7 indicates positive margin to the peak grid impact loads confirming that the plastic deformation of less than 1 mm is met for the NuScale design.

1.2 *RAI 9555 - STRESS ANALYSIS METHOD COMPATIBILITY WITH ANP-10337PA*

Question 2.1: Clarify how the limiting NuScale deflection cases were selected for evaluation in 3D structural models.

Response:

The selection process uses time-phasing and follows the process outlined in Section 8.1.2 of ANP-10337PA ([3]):

- For each time history **[** **]** and the maximum values are reported
- The maximum **[** **]**
- Given that the calculated stresses do not take into account the core location, a Level C service limit is imposed as allowable.

Question 2.2: The NRC requests further information to determine implementation of time phasing in view of L&C #4. If time phasing was used for NuScale, describe how time phasing was used to find the most limiting fuel assembly component stress states. If time phasing was not used, confirm that L&C # 7 was followed.

For convenience L&C #4 is reproduced below:

4. *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.*

Response:

The time phasing process used for the NuScale analysis conforms to Section 8 of ANP-10337PA ([3]). In more detail:

- For each time history, **[** **]** at each specific core location.

- The X and Z deflections at each core location, **[**

  ] due to numerical noise.

- The deformed configurations in the X and Z direction are then used **[**

  ]

- In conclusion, the NuScale maximum **[** **]** deflections are determined via time-phasing at each core location. However, only the overall maximum cases for each simulated event, independent of core location, were retained for detailed analysis with the 3-D model. As such, the margin reporting was performed using Level C limits and therefore, L&C #7 is not a concern. L&C #7 requires that: *"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."*

The rest of the discussion for this question addresses the similarity between the NuScale fuel assembly and the current range of operational experience, which makes the use of time-phased deflection selection acceptable. To this end, the relevant NuScale core and fuel assembly design parameters are compared with the corresponding core and fuel assembly design parameters used in the Sample Problem in Appendix B of ANP-10337PA ([3]). This comparison is summarized in Table 1-1. The parameters listed in this table are relevant for a comparison of the amount of strain the slender components in the two assemblies can be subjected to, given that strain is a geometric scale invariant, by virtue of its definition as deflection divided by length. A mechanical structure under an imposed displacement will experience the same strain as a half-scale of this structure under half the imposed displacement. The direct consequence of this observation is that two fuel assemblies will experience the same strains if they have the same cross-sectional properties, and the length ratio and the deflection ratio in the same mode are the same. Based on this the total geometrically available gap space in the core for a Mode 1 and a Mode 3 deflection has been calculated.

The cross-sectional properties of the two assemblies are the same as shown in reference ([2]). Comparing the length ratio of the NuScale vs. generic ANP-10337PA fuel assemblies and the geometrically feasible displacement ratios, it is apparent that while the NuScale fuel assembly is 56.4% the length of the generic ANP-10337PA fuel assembly, it can experience proportionally lower deflections of only 44.4% in mode 1 and 40.5% in mode 3. This allows the extension, on a scale basis, of all considerations and acceptability measures from the generic fuel assembly of ANP-10337PA to the NuScale assembly. This observation substantiates the conclusion that the NuScale assembly is similar to the generic assembly in the approved methodology of ANP-

10337PA, and the time-phasing method is appropriate in this case. Furthermore, the margin calculations were performed using Level C stress limits and therefore, L&C #7 is fulfilled.

Question 2.3: Confirm that the NuScale 3-D load combination was performed consistent with ANP-10337P limitation and condition number 8.

For convenience L&C #8 is reproduced below:

L&C #8 to ANP-10337PA ([3]) requires that, in the case when **[**

]

8. *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).* ***[***

***]**.*

Response:

The NuScale fuel assembly was analyzed using a load combination in three orthogonal directions and therefore, L&C #8 is not a concern.

Table 1-1 - NuScale vs. ANP-10337PA Sample Problem Fuel Assembly Comparison

Design	NuScale	ANP-10337PA Sample Problem
	Parameter Value	Parameter Value
Spacer Grid	W-17x17 HTP	W-17x17 HTP
Core Gaps	FA-FA: 1.68 mm FA-BP: 1.58 mm	FA-FA: 1.76 mm FA-BP: 1.65 mm
Fuel Assembly Length (between Nozzles)	2196 mm	3892 mm
Maximum Core Row Length	7	15
FA length ratio (NuScale/ANP-10337)	56.4%	N/A
Mode 1 full core gap ratio	44.4%	N/A
Mode 3 half core gap ratio	40.5%	N/A

1.3 RAI 9555 - STRESS ANALYSIS MODEL VALIDATION TO TEST DATA

<u>Question 3.1:</u> Identify the maximum lateral deflection and strain energy calculated over the full set of design basis horizontal analyses

Response:

In this document only the maximum deflection cases for the BOL (Table 1-2) and EOL (Table 1-3) condition are described. The maximum deflections are caused by **[**
]

The deflections in Table 1-2 and Table 1-3 are reported in the absolute coordinate system and must be processed to eliminate the rigid body modes (per RAI 18 of ANP-10337PA ([3]). For the purposes of this discussion it is sufficient to zero the deflections to the Bottom Nozzle displacement. This yields a conservative estimate of the maximum FA deflection of **[ ]** for the BOL case and **[ ]** for the EOL case. **[**
]

Table 1-2 - Maximum Lateral FA Deflection - BOL Seismic

[

]

Note for Table 1-2: **[**
]

Table 1-3 - Maximum FA Lateral Deflection - EOL Seismic

[

]

***Note* for *Table 1-3*: [**

]

<u>Question 3.2:</u> Provide force-deflection data from the lateral stiffness test described in 5.1.1 of TR-0816-51127-P, Revision 1. Provide force-deflection data that encompasses the maximum deflection calculated in the model, or as high of a deflection as was recorded in the lateral stiffness test.

Response:

In this response the test data for the BOL (Figure 1-1) and the EOL (Figure 1-2) condition are provided.

The BOL lateral stiffness test data extends up to **[**

] Similarly, the EOL test extends up to **[**

]

[

]

Figure 1-1 - BOL Lateral Stiffness Test Data and Model Benchmark

[

]

Figure 1-2 - EOL Lateral Stiffness Test Data and Model Benchmark

Question 3.3: Compare the maximum lateral deflection and strain energy calculated in the horizontal dynamic finite element model to the supporting mechanical test data. Confirm that the calculated lateral deflection is within a range where model behavior agrees with test data, or justify the use of the model outside the range where it agrees with test data.

Response:

As shown in Figure 1-1 and Figure 1-2, above, the tested deflection range is representative of the model deflection range and the model agrees reasonably well with the test data, including the non-linear behavior, **[**

]

2.0 IMPACT ON TECHNICAL REPORT:

There are no impacts to the technical report, TR-0816-51127, as a result of this response.

3.0 REFERENCES

1. NuScale, LLC, “NuFuel-HTP2 TMFuel and Control Rod Assembly Designs,” TR-0816-51127-P, Rev. 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, Rev. 0, September 2016

3. AREVA NP, Inc., “PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations,” ANP-10337PA, Rev. 0, May 2018

4. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components,” 2009 Revision with Addenda, New York

Impact on Topical Report:

Topical Report TR-0716-50351, NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces, has been revised as described in the response above and as shown in the markup provided in this response.

1.0 Introduction

1.1 Purpose

This report demonstrates the applicability of *PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations*, ANP-10337P (Reference 1), to the NuScale fuel assembly designed for use in the NuScale Power Module. The generic methodology described in ANP-10337P is used to evaluate the structural response of the NuScale fuel assembly to dynamic loads applied during seismic and loss-of-coolant accident (LOCA) events, consistent with the guidance in Standard Review Plan Section 4.2 Appendix A, *Evaluation of Fuel Assembly Structural Response to Externally Applied Forces*, NUREG-0800 (Reference 2).

NuScale Power, LLC is submitting this topical report for Nuclear Regulatory Commission (NRC) approval to apply the methodology described in ANP-10337P to the NuScale fuel assembly design. ~~Use of this applicability report is contingent upon NRC approval of ANP-10337P, currently under NRC review.~~

1.2 Scope

In order to demonstrate the applicability of ANP-10337P to the NuScale fuel assembly, this report:

- Provides an evaluation of each chapter of ANP-10337P with regard to NuScale applicability.
- Provides additional justification for areas of the methodology that require special consideration due to unique features of the NuScale fuel assembly.
- Summarizes selected elements of the NuScale fuel assembly modeling to enhance the applicability justification.

This report does not provide results of the full suite of characterization tests and the structural response analyses of the NuScale fuel. A summary of the application results are being submitted for review and approval as part of the NuScale Design Certification Application.

2.0 Background

ANP-10337P defines a generic methodology to evaluate the structural response of pressurized water reactor (PWR) fuel assembly designs subjected to dynamic loads under seismic and LOCA events. ~~ANP-10337P has been submitted for NRC review and approval.~~ This report demonstrates that this method is applicable to the NuScale fuel design.

To assist in the review of the applicability of the methodology, a brief summary of the NuScale fuel design is provided below.

Welded Fuel Assembly Structure

The NuScale 17x17 fuel assembly design is a reduced-height version of AREVA's 17x17 PWR fuel designs for Westinghouse-type reactors. The total, nominal height of the fuel assembly is 94 inches (not including hold-down springs). Due to the reduced height and the use of span lengths between spacer grids that are typical for operating PWR plants, the assembly has a total of five spacer grids. The HTP™ grids are welded to the guide tubes, while the HMP™ grid is captured by rings welded to the guide tubes. The design includes features as described in the following subsections.

Fuel Rod with Alloy M5® Fuel Rod Cladding

The fuel rod design features M5® cladding. The seamless M5® cladding encapsulates ceramic UO_2 pellets that are cylindrically shaped with a spherical dish at each end. The fuel rod has an internal spring system that axially restricts the position of the fuel stack within the rod, preventing the formation of gaps during shipping and handling while allowing for the expansion of the fuel stack during operation. The lower end cap has a bullet-nose shape to provide a smooth flow transition in addition to facilitating insertion of the rods into the spacer grids during assembly. The upper end cap has a grippable shape that allows for the removal of the fuel rods from the fuel assembly if necessary, which is typical of AREVA fuel for operating PWR plants.

The nominal density of the pellets is 96 percent theoretical density with a possible enrichment up to 4.95 weight percent ^{235}U.

Zircaloy-4 HTP™ upper and intermediate spacer grids

The four HTP™ spacer grids that occupy the top four grid positions are formed from interlocking strips that are welded at all intersections and welded to the side plates. Each grid strip includes a pair of strips welded back-to-back to produce flow channels. The design creates a flow path that is slanted at its outlet, thus causing a vortex flow pattern under normal PWR operating conditions. The spacer grid design creates line contacts with the fuel rod, which provide resistance to grid-to-rod fretting relative to traditional point-contact spacer grid designs. The HTP™ grids on the NuScale design are identical to those used on AREVA's 17x17 PWR product.

design is identical to the grid cited in the sample problem for ANP-10337P, this potential limitation on the applicability of the methodology is satisfied.

In summary, the NuScale design is consistent with the general conditions defined in Chapter 2 of ANP-10337P. Therefore, the range of applicability defined in this chapter encompasses the NuScale design. The differences in assembly length and number of spacer grids are addressed in Section 3.3 of this report. Compliance to the requirement defined in Section 2.2 of ANP-10337P is demonstrated in Section 4.0 and Appendix A.

Chapter 3 Regulatory Requirements

Chapter 3 reviews the regulatory requirements that are relevant to this methodology. These requirements include Appendix A (GDC 2, 27, and 35) and Appendix S of 10 CFR Part 50 and 10 CFR 50.46. In addition, this chapter reviews the NRC guidance from the Standard Review Plan Section 4.2 that pertains to these requirements (primarily Appendix A).

The requirements of Appendices A (GDC 2, 27, and 35), and S of 10 CFR Part 50, 10 CFR 50.46 are directly applicable to the NuScale design. The guidance of SRP Section 4.2 Appendix A is directly applicable and is implemented in ANP-10337P. ~~The discussion of 10 CFR Part 50 Appendix S is applicable with the following minor clarification:~~

- ~~10 CFR 52.47(a)(2)(iv) specifies the requirements for offsite radiological consequence analyses, including exposure limits, for design certification applicants, as opposed to 10 CFR Part 100, 10 CFR 50.34, and 10 CFR 50.67 as stated in ANP-10337P. However, the conclusion that fuel rod failures are permitted during postulated accidents and must be accounted for in the dose analysis remains applicable. This minor clarification does not affect the applicability of the method.~~

The same regulatory requirements identified in ANP-10337P Chapter 3 are applicable to the NuScale Power Module; therefore, this chapter is applicable to the NuScale design.

Chapter 4 Acceptance Criteria

Chapter 4 establishes the appropriate selection of acceptance criteria in order to satisfy the regulatory requirements specified in Chapter 3. In general, this chapter establishes criteria to evaluate spacer grid impact loads and allowable stresses for non-grid components.

Like the regulatory requirements in Chapter 3, these criteria are generic to PWR fuel. The NuScale fuel design uses the same components and structure as the PWR designs presented in Chapter 2 of ANP-10337P; therefore, the criteria defined in this chapter can be applied to the NuScale design to demonstrate compliance to the regulatory requirements with the following minor clarification:

- Section 4.2.2 of ANP-10337P justifies excluding the evaluation of hold-down springs for accident condition loads based on specific designs of fuel assemblies and reactor core internals. The justification for Westinghouse designs applies to the NuScale

~~Two~~One item~~s~~ from Chapter 6 is~~are~~ potentially affected by the application to NuScale fuel.

- ~~Section 6.1.1.2 states that the objective of the forced vibration tests is to obtain at least the first five natural frequencies. In the case of the NuScale design, due to the shorter length and the presence of only three intermediate spacer grids, it is not necessary, nor practical, to obtain characteristics beyond the first three frequencies and mode shapes. [~~

  ~~] This difference is addressed in Section 3.3.~~
- Section 6.1.3 presents damping values to be used in the horizontal model. As noted in the review of Chapter 5, the contribution of axial coolant flow to the fuel assembly damping is expected to be much less than for other operating PWRs. As a result, the damping definition provided in Section 6.1.3 of ANP-10337P is not applicable to NuScale. This difference is addressed in Section 3.3.3~~3.3~~.

Chapter 6 of ANP-10337P is applicable to the NuScale fuel design with the exception of ~~(1) the request to experimentally characterize fuel assembly up to the first five natural frequencies, and (2)~~ the definition of damping values presented in Section 6.1.3. ~~Both of these~~This item~~s~~ is~~are~~ addressed in Section 3.3.

Chapter 7 Seismic and LOCA Analysis

Chapter 7 defines the process of applying appropriate forcing functions representing seismic or LOCA events to the models described in Chapter 5. Chapter 7 also defines the method of accounting for the combined effect of seismic and LOCA loads. In the horizontal analysis, the model calculates the time-varying displacements and impact forces for assemblies across the core. The results of this analysis are also used for calculating the resulting loads and stresses in the assembly. Similarly, the vertical model calculates a time-varying response from the fuel assembly that is used to evaluate the loading on fuel assembly components.

It was demonstrated above that the development of design-specific models and boundary conditions in accordance with ANP-10337P is applicable to the NuScale design. The process to apply these models to determine the fuel assembly structural response to seismic and LOCA events is design independent. Therefore, this chapter is applicable to the NuScale fuel design.

Chapter 8 Non-Grid Component Strength Evaluation Methodology

Chapter 8 defines the process of performing the structural component stress analysis using the loads and deflections generated by the seismic and LOCA analyses described in Chapter 7. Like the modeling approach addressed in Chapters 5 and 6, the analysis approach in Chapter 8 is applicable to NuScale fuel because the fuel design uses the

~~*Appendix E: Methodology for Evaluating the Effect of Grid Deformation on ECCS Coolability Analyses*~~

~~Appendix E presents the methodology for evaluating the effects of grid deformation on post-LOCA emergency core cooling system coolability analyses. The method presented here is based on a simple convective heat exchange across the fuel rod and it accounts for small reductions in the area of the flow channel.~~

~~This appendix is applicable to the NuScale fuel design.~~

Appendix ~~F~~E: Justification for the Use of Level C Stress Limits to Ensure Guide Tube Functionality

Appendix ~~F~~E provides the basis for the acceptability of using Level C stress limits to ensure guide tube functionality (i.e., control rod insertability) following a seismic or LOCA event. The discussion and data presented in Appendix ~~F~~E are generic to any guide tube geometry. The characterization of guide tube stress states and the definition of the Level C service limit in relation to guide tube geometry are generic. Furthermore, the testing discussed in Appendix ~~F~~E is performed on guide tubes of the same cross-sectional geometry as the NuScale design.

This appendix is applicable to the NuScale fuel design.

3.2 Topical Report Restrictions

~~ANP-10337P is currently under review by the NRC. Thus, no additional restrictions have been imposed on its use.~~This section addresses the Limitations and Conditions (L&Cs) of ANP-10337P in the context of the application of this methodology to the NuScale Fuel Assembly design.

L&C #1 Discussion:

L&C #1 imposes requirements on the tested behavior of grids in order to be compliant with the ANP-10337P (Reference 1) methodology:

1. *Dynamic grid crush tests, must be conducted in accordance with Section 6.1.2.1 of ANP-10337P (as amended by RAI 16), and spacer grid behavior must satisfy the requirements in the TR, the key elements of which are:*
   a. [ ]
   b. [ ]
   c. [ ]

[ ]

The acceptability of the NuScale grids under L&C #1 has been addressed in Reference 3.

L&C #2 Discussion:

L&C #2 imposes requirements on the maximum allowable deformation of spacer grids:

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 TR apply:*
   a. *For all OBE analyses, allowable spacer grid deformation is limited to design tolerances and [ ].*
   b. *For SSE, LOCA, and combined SSE+LOCA analyses, [ ]*

The acceptability of the NuScale grids under L&C #2 has been addressed in Reference 3.

L&C #3 Discussion:

L&C #3 imposes controls and quality requirements on the computer programs implementing the methodology of ANP-10337P (Reference 1):

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 analyses use the same code versions employed in the analytical method demonstration in Appendix B of ANP-10337P. Therefore, L&C #3 is not a concern.

L&C #4 Discussion:

L&C #4 limits the un-restricted use of the ANP-10337P (Reference 1) methodology to fuel designs and applications consistent with the operating fleet. Markedly new designs have to be assessed:

4. *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.*

L&C #4 has been addressed in Reference 3. While in absolute terms the NuScale fuel assembly is different from the operating fleet, this design has been demonstrated to be similar, on a scale basis to the generic fuel assembly of ANP-10337P (Reference 1). The available lateral deflections in core when scaled by fuel assembly lengths are smaller for NuScale than those for the generic assembly in ANP-10337P. This justifies the extension, on a scale basis, of all considerations and acceptability measures from the generic fuel assembly of ANP-10337P to the NuScale assembly. This observation substantiates the conclusion that the NuScale assembly is similar to the generic assembly in the approved methodology of ANP-10337P, and the time-phasing method is appropriate in this case.

L&C #5 Discussion:

L&C #5 limits the applicability of the lateral damping formulation to existing designs, and requires an applicability justification or a new formulation for new designs:

5. *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.

The NuScale fuel assembly is much shorter than the current fleet designs, and the lateral fuel assembly damping has been re-formulated to account for specific test results on short assemblies, the particulars of the axial flow, and the phenomena governing the dynamics of these designs. This formulation is addressed within this document in Appendix 2 and also in Reference 4 (Question 29611).

L&C #6 Discussion:

L&C #6 requests that the fuel rod assessment under faulted conditions be demonstrated.

6. *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.*

The fuel rod analysis is part of the component stress evaluation that was performed for the NuScale fuel design.

L&C #7 Discussion:

L&C #7 requires that when bounding stress analysis of the non-grid components is used, without regard to specific core location, the more stringent limits for control rod locations must be used:

7. *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.*

The margin calculations for the NuScale fuel assembly Guide Tubes were performed using Level C stress limits, which are applicable to control rod locations, therefore L&C #7 is fulfilled.

L&C #8 Discussion:

L&C #8 requires that, in the case when [

]:

8. *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). [*

   ].

The NuScale component stress analysis was performed using a 3-D load combination as discussed in Reference 3. Therefore, L&C #8 is not a concern.

L&C #9 Discussion:

L&C #9 places a restriction over the range of applicability of [

].

9. *[*

   ].

This point has been addressed in Reference 3. The NuScale grid design is the same as the grid in the generic fuel assembly used in the Sample Problem (Appendix B) of ANP-10337P (Reference 1). The limitation of L&C #9 has been met.

3.3 NuScale Design Differences and Requirements

To extend the applicability of ANP-10337P to include NuScale fuel, the following design differences are addressed:

- NuScale fuel assembly is shorter than typical PWR designs presented in Table 2-1 of ANP-10337P.
- ~~The experimental characterization of the frequency response of the NuScale fuel design is limited to the first three natural frequencies, as opposed to the first five natural frequencies as requested in Section 6.1.1.2 of ANP-10337P.~~
- The contribution of axial coolant flow to the NuScale fuel assembly damping is expected to be much less than that for other operating PWRs, and thus, the damping values presented in Section 6.1.3 of ANP-10337P are not applicable to NuScale fuel.

3.3.1 Fuel Assembly Length and Number of Spacer Grids

Although not stated as defining a range of applicability, Table 2-1 of ANP-10337P illustrates typical PWR designs to which ANP-10337P can be expected to be applied. The NuScale fuel assembly is outside the range of parameters in Table 2-1 in terms of fuel assembly length (shorter) and number of grids (fewer).

- The shorter assembly length of the NuScale fuel design will result in unique dynamic properties of the fuel assembly (i.e., higher stiffness and higher natural frequencies). However, this difference in design is captured in the method defined in ANP-10337P because the method requires that fuel assembly models be built to match design-specific experimental dynamic characterization of the fuel design. The expected differences in the dynamic properties of the NuScale fuel assembly due to its shorter length are directly characterized through full-scale prototype testing and the models built to match this tested behavior had negligible error. The application of this method to NuScale with its shorter length is discussed in more detail in Section 4.0 and Appendix A.
- The designs in Table 2-1 of ANP-10337P have between five and nine intermediate spacer grids, whereas the NuScale fuel design has three intermediate spacer grids. The NuScale fuel assembly has a total of five spacer grids, but following the modeling architecture defined in Section 5.2.1 of ANP-10337P, the uppermost and lowermost end grids are not modeled explicitly [

  ] As a result, the NuScale fuel assembly model will be represented as a single beam with three rotational nodes at the intermediate grid locations. With three non-fixed degrees of freedom, the model is only capable of accurately representing the fuel assembly response up to the third mode, consistent with the limitations of the experimental testing of the NuScale fuel assembly, in which it is only practical to characterize assembly frequencies up to the third mode (see Section 3.3.2 below). [

  ] The application of this fuel assembly model, with three rotational nodes, is demonstrated in Section 4.0 and Appendix A of this report and shows negligible error to tested results. Additional studies have demonstrated that results from this modeling approach reflect an appropriate level of mass participation in the dynamic response.

Therefore, with regard to the shorter fuel assembly length and fewer spacer grids, ANP-10337P remains applicable to NuScale fuel without modifications.

3.3.2 ~~Frequency Response of the NuScale Fuel~~Deleted

~~Section 6.1.1.2 of ANP-10337P establishes a requirement that the dynamic characterization testing provides the first five frequencies and mode shapes of the fuel assembly. For the NuScale fuel assembly, because of its shorter length and increased lateral stiffness, it is only practical to characterize the first three natural frequencies. In general, because of the increased lateral stiffness of the NuScale fuel assembly, the higher mode frequencies have shifted beyond the range of interest for the dynamic events that are analyzed. [~~

~~]~~

~~Therefore, with regard to the representation of the NuScale fuel, ANP-10337P remains applicable to NuScale fuel without modifications.~~ This section is no longer needed.

3.3.3 Fuel Assembly Damping

Section 6.1.3 of ANP-10337P defines fuel assembly damping values that are generically applicable to standard PWR fuel designs. However, relative to a standard PWR, the NuScale design will operate with a shorter fuel assembly and reduced flow rates. For these reasons, the damping values defined in Section 6.1.3 of ANP-10337P are not applicable to the NuScale design.

Section 5.0 and Appendix B provide details regarding the establishment of NuScale-specific fuel assembly damping values. For the NuScale design, the maximum fuel assembly damping ratio values to be used for the analysis of seismic and LOCA events in place of those defined in Section 6.1.3 of ANP-10337P are defined in Table 3-1. These damping values do not credit the additional contribution of damping in flowing water.

Table 3-1. NuScale Fuel Assembly Damping Ratio Values

[ ]

4.0 NuScale Fuel Characterization

Two areas of fuel characterization are reviewed to provide an explicit demonstration of the application of ANP-10337P to NuScale fuel. These two items are reviewed in detail in Appendix A, but a summary is provided in this section.

- Section 2.2 of ANP-10337P requires an explicit demonstration of the applicability of grid impact modeling elements.
- Section 3.3 of this document notes that the NuScale fuel assembly is outside the range of typical PWR designs to which ANP-10337P is applied in terms of both overall fuel assembly length and the number of spacer grids. The application of the single fuel assembly model described in Section 5.2.1 of ANP-10337P to the NuScale design, with a shorter overall length and fewer spacer grids, is demonstrated.

4.1 Spacer Grid Behavior

Under lateral impacts over the range of application, Section 2.2 of ANP-10337P specifies that **[**

]

The NuScale fuel design utilizes the same 17x17 HTP™ spacer grid that is currently in use in operating plants. Thus, this behavior is well established for this existing grid design. Figure A.2-1 and Figure A.2-2 in Appendix A demonstrate **[**

]

4.2 Single Fuel Assembly Model

The applicability of the single fuel assembly model, as defined in Section 5.2.1 of ANP-10337P, to the NuScale fuel design is addressed in this section. This section shows the ability of a benchmarked fuel assembly model to replicate a frequency that characterizes test data from free and forced vibration testing.

The free vibration test is performed in order to characterize the primary, or first mode, natural frequency of the fuel assembly. The NuScale fuel assembly was tested over a range of deflections from **[** **]**

7.0 References

1. AREVA Inc., "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," ANP-10337P Rev. 0, August 2015.
2. U.S. Nuclear Regulatory Commission, "Standard Review Plan, Fuel System Design," NUREG-0800, Chapter 4, Section 4.2, Rev. 3, March 2007.
3. ANP-3712P-000, Framatome Responses to NRC RAI No. 9555 regarding NuScale Topical Report TR-0816-51127
4. ANP-3591P, Revision 0, AREVA Responses to NRC RAI 8736 (Questions 29611, 29613-29616) regarding TR-0716-50351, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces"

Appendix A. Review of NuScale Fuel Characterization Test Data Applicability to ANP-10337P

A.1 Introduction

The purpose of this appendix is to provide a review of NuScale fuel characterization test data in order to demonstrate the behavior necessary to confirm the applicability of ANP-10337P. Specifically, Section 2.2 of ANP-10337P requires an explicit demonstration of the applicability of grid impact modeling elements. In addition, Section 3.3 of this report notes that the NuScale fuel assembly is outside the range of typical PWR designs to which ANP-10337P is applied in terms of both overall fuel assembly length and the number of spacer grids. This appendix also demonstrates the application of the single fuel assembly model described in Section 5.2.1 of ANP-10337P to the NuScale design, with a shorter overall length and fewer spacer grids.

A.2 Spacer Grid Behavior

Under lateral impacts over the range of application, Section 2.2 of ANP-10337P specifies that [

]

The NuScale fuel design utilizes the same 17x17 HTP™ spacer grid that is currently in use in operating plants. Thus, this behavior is well established for this existing grid design. The data to be reviewed here are generic and not specific to NuScale.

Figure A.2-1 and Figure A.2-2 demonstrate [

] Figure A.2-1 and Figure A.2-2 present this relationship for single grids, but they are representative of the same behavior seen over the total population of tested spacer grids. [

]

Section D

December 19, 2019

Docket No. PROJ0769

U.S. Nuclear Regulatory Commission
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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351, Revision1

REFERENCES: Letter from NuScale Power LLC, to Nuclear Regulatory Commission "NuScale Power, LLC Submittal of Topical Report TR-0716-50351, 'NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces,' Revision 0 (NRC Project 0769)," dated September 30, 2016

NuScale Power, LLC (NuScale) hereby submits Revision 1 of the "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces" (TR-0716-50351).

Enclosure 1 contains the proprietary version of the report entitled "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces." 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) pertains to Framatome Inc. (formerly AREVA Inc.) proprietary information to be withheld from the public. Framatome proprietary information is denoted by bolded straight brackets (i.e., "**[ ]**"). Enclosure 2 is the nonproprietary version of the report.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please feel free to contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com if you have any questions.

Sincerely,

Michael Melton
Manager, Licensing
NuScale Power, LLC

Distribution: Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Michael Dudek, NRC, OWFN-8H12
Bruce Bavol, NRC, OWFN-8H12

Enclosure 1: "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351-P, Revision 1, proprietary version

Enclosure 2: "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351-NP, Revision 1, nonproprietary version

Enclosure 3: Affidavit of Gayle Elliott, Framatome, Inc.

Enclosure 2:

"NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351-NP, Revision 1, nonproprietary version

Note: This enclosure to NuScale's December 19, 2019 letter to the NRC was the non-redlined version of Revision 1 of the "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response of Externally Applied Forces" Topical Report, and is the same as Revision 1 included in Section B, with the exception that the Section B version includes "-A" in the document identification number. Therefore, this enclosure is not included in the current package.

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
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Enclosure 3:

Framatome Affidavit of Gayle Elliott

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A F F I D A V I T

1. My name is Gayle Elliott. I am Deputy Director, Licensing & Regulatory Affairs for Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the Framatome information contained Licensing Topical Report TR-0716-50351-P-A, Revision 1, entitled, "NuScale Applicability of AREVA Method for Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," dated April 2020 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome 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 Framatome 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 Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome'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 Framatome.

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

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

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

7. In accordance with Framatome'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 Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome 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.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: May 1, 2020

Gayle Elliott

Gayle Elliott