April 30, 2020 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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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 Additional 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.

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
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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 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

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.

4.2.1.2.3 Fuel Pellet Chemical Properties

Fuel pellet chemical properties are controlled through a rigorous testing and inspection program to demonstrate that each lot of pellets conforms to design requirements and criteria as described in Section 4.2.4.3.

4.2.1.3 Fuel Rod Performance

The basic fuel rod models and the ability to predict fuel rod operating characteristics are described in Reference 4.2-4. The COPERNIC computer code is used to perform the thermal-mechanical analyses to simulate the behavior of the fuel rod during irradiation, and is also used to verify that the fuel rod design meets design and safety criteria. The critical design bases addressed with COPERNIC include fuel rod internal pressure, cladding temperatures, cladding strain, corrosion, and centerline fuel melt under conditions of normal operation, AOOs, and postulated accidents. Reference 4.2-1 provides additional details concerning the design basis for normal operations and AOOs.

Section 4.4 addresses critical heat flux design criteria. Section 15.4 addresses reactivity-initiated accidents, reactivity insertion accidents, and fuel centerline temperatures. Creep collapse is analyzed with the methods and codes described in Reference 4.2-8. The applicability of Reference 4.2-8 to the NuScale fuel design is justified in Reference 4.2-3.

4.2.1.4 Spacer Grids

The spacer grids are designed to maintain the fuel rods in a coolable configuration (PDC 35 and 10 CFR 50.34), and ensure CRA insertion for AOOs and postulated accidents (PDC 27).

Structural evaluations of the grids determine that the grid strength is sufficient to maintain a coolable geometry and ensure control rod insertion for all resulting impact loads. The evaluation methodology (Reference 4.2-5) uses the load limits that are derived from testing, which are provided in Reference 4.2-1 for the fuel assembly mechanical design.

4.2.1.4.1 Mechanical, Chemical, Thermal, and Irradiation Properties of Grids

The strength criteria of the fuel assembly grid components are based on mechanical strength testing of prototypes, including static and dynamic crush testing.

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The design limits are detailed in Reference 4.2-1. The grids are tested to establish a 95 percent confidence level of the mean allowable crushing stress limit for both the unirradiated and a simulated irradiated condition. These limits are sufficient to demonstrate that, under worst-case combined seismic and loss-of-coolant accident (LOCA) events, the fuel assemblies will remain in a coolable geometry (PDC 35 and 10 CFR 50.34) and CRA insertability (PDC 27) is maintained. These

criteria are met by showing that the spacer grids experience no significant plastic deformation exceeding the limit in Reference 4.2-5 as a result of the combined events.

The allowable grid clamping loads during fuel shipment are based on static crush strength testing for static stiffness and elastic load limits. The spacer grids maintain their structural integrity under the maximum lateral shipping loads and the maximum clamping loads. The spacer grid springs are designed to maintain acceptable fuel rod grip forces from the limiting 6 g lateral (transverse) and 4 g axial (longitudinal) shipping loads.

Spacer grid slip load input to the analytical models of the fuel assembly used in the horizontal and vertical faulted analyses are established by mechanical testing.

4.2.1.4.2 Vibration and Fatigue of Grids

The interface between the fuel rods and the spacer grids is maintained throughout the life of the fuel assembly and prevents fuel rod fretting failure. Full-scale fuel assembly testing and a grid-to-rod fretting evaluation detailed in Reference 4.2-1 show that fuel rod cladding wear is expected to be acceptable (see Section 4.2.1.1.3). The grid-to-rod fretting evaluation is performed in accordance with NRC approved methods.

4.2.1.4.3 Chemical Compatibility of Grids with other Core Components

The Zircaloy-4 and Alloy 718 materials of the spacer grids are compatible with the reactor coolant based on extensive operating experience in US PWRs.

4.2.1.5 Fuel Assembly Structural Design

The design bases for evaluating the structural integrity of the fuel assemblies is established by setting design limits on stresses and deformations due to various non-operational, operational, and abnormal loads.

The thermal-hydraulic design basis is presented in Section 4.4.

4.2.1.5.1 Non-Operational Loads

The non-operational load limit is 4 g axial (longitudinal) and 6 g lateral (transverse) with dimensional stability.

4.2.1.5.2 Normal Operating Conditions and Anticipated Operational Occurrences

For AOOs, the fuel assembly component structural design criteria are established for the two primary material categories, austenitic steels and zirconium alloys. The stress categories and strength theory presented in Section III of the ASME BPVC are used as a general guide. The maximum shear theory (Tresca criterion) for combined stresses is used to determine the stress intensities for the austenitic steel components. The stress intensity is defined as the largest numerical difference between the various principal stresses in a three-dimensional field. The design

The results of the analyses in Reference 4.2-1 are applicable to fuel assembly operation in the NuScale Power Module.

4.2.3.5.1 Fuel Assembly Structural Design Evaluation

The design criterion for the structural evaluation of the NuScale fuel assembly design is that stress intensities are less than the stress limits based on Section III of the ASME BPVC. The structural design requirements for the NuScale fuel assembly are common to current AREVA PWR fuel designs and incorporate AREVA's design and incore operating experience with similar PWR fuel designs.

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The requirements are consistent with the acceptance criteria in the Standard Review Plan. Evaluation results show that the calculated stress intensities are less than the applicable stress limits. Fatigue usage is evaluated and found to be acceptable. ASME Code Service Level A criteria are used for normal operating conditions and ~~Code~~Service Level D criteria are generally used for the LOCA and seismic (i.e., faulted) analyses. An exception to this classification is the use of ~~Code~~Service Level C criteria for guide tubes when CRA insertability is required for the faulted analyses.

The fuel assembly component evaluations show that the calculated stresses are lower than the material allowable stresses for both normal operation and faulted conditions for all evaluated components. The evaluation of components for LOCA conditions conservatively considered the square root of the sum of the squares combination of the LOCA and SSE loads.

The fuel assembly components evaluated include:

Guide tubes: The guide tubes do not buckle and remain elastic, thereby ensuring the CRAs can be inserted during normal operation. A positive guide tube buckling safety margin is determined for axial loading for all normal operating conditions. The hot zero power condition was determined as the limiting normal operating case for compressive loading compared with hot full power operation. CRA impact loads due to a scram were considered. The critical buckling load was determined with a classical Timoshenko buckling stress model, with results compared to the material yield strength at operating temperature. Initial lateral deflection (column eccentricity) was imposed on the guide tube model at mid-height in the magnitude of the available assembly and baffle clearance in the most limiting row with added limiting manufacturing variance, to account for potential reduction in the critical load due to fuel assembly bow. A positive buckling stress margin was predicted by implicit solution of the Timoshenko buckling stress formulation. Guide tube corrosion, load maldistribution, and temperature effects on material properties were also considered. Guide tube boiling is not predicted to occur during normal operation. Guide tube stresses were also evaluated for faulted conditions and shown to maintain CRA insertion capability by meeting the applicable criteria.

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Spacer grids: The spacer grids do not ~~plastically~~ deform beyond the approved limits in Reference 4.2-5 during normal operation and faulted conditions. The mechanical design bases of the spacer grids are confirmed through a series of tests on prototype 17x17 HTP™ grids as discussed in Section 4.2.4.

Bottom nozzle: The evaluation for normal operating conditions is performed in accordance with Subsection NG-3228.2 of the ASME BPVC using a design limit of two-thirds of the collapse load limit obtained by testing and adjusted for operating temperature.

The limit based on the maximum test load is further reduced for operating temperature conditions. Axial loading only is considered because the normal operating loads on the bottom nozzle are applied axially by the guide tubes. The maximum normal operating load used in the evaluation accounts for impact loads from a CRA scram, hold-down spring loads, and the fuel assembly mass.

The evaluation of the bottom nozzle for faulted operating conditions is performed in accordance with Appendix F, Paragraph F-1440(a) of the ASME BPVC using a design limit based on the maximum cold test load. The limit based on the maximum cold test load is further reduced for normal operating temperature evaluations. The maximum normal operating loads used in the evaluation included moment loads, plus the assembly weight, plus LOCA, plus SSE axial loads. Moment loads were considered by calculating the axial load equivalent of the moment couples created by the position of the guide tubes in relation to the center of the bottom nozzle. Margin to the design limit was demonstrated in Reference 4.2-1.

Top nozzle: The top nozzle structure is evaluated for normal operating and shipping and handling loads in accordance with Subsection NG-3228.2 of the ASME Code using a design limit of two-thirds of the collapse load limit obtained by testing and adjusted for operating temperature. The limiting case is evaluated for an axial scram load applied to the top nozzle structural framework and showed that positive margin to the design limit was maintained.

Hold-down spring: Stress analysis of the fuel assembly hold-down spring examines stresses, strains, and fatigue usage to confirm that it does not fail. The evaluation confirmed that the ASME BPVC criteria are satisfied. The spring stresses are treated as secondary stresses since the hold-down spring stresses are controlled by the total separation between the lower and upper core plates.

The secondary stress limits are satisfied by performing a plastic analysis in accordance with Subsection NG-3228.1 of the ASME BPVC. The hold-down springs remain within the elastic range. The maximum normal operating loading bounds the faulted condition loading because the fuel assembly lifts off the lower core plate during some severe seismic events. Therefore, satisfying the normal operating conditions criterion also satisfies the faulted condition. The known spring displacements were converted to stresses to demonstrate the criterion is met.

Structural Connections: The guide tube-to-spacer grid weld connections are evaluated for the limiting applied normal operating condition loads which are

The maximum grid impact forces that were obtained for SSE and SSE plus LOCA conditions for a full-core configuration of NuScale fuel assemblies were less than the allowable limits established by testing, as discussed in Section 4.2.3.4. Other fuel assembly components were evaluated for combined loads and stresses under vertical and lateral SSE plus LOCA conditions. The loads and stresses resulting from lateral SSE and LOCA excitations are the result of fuel assembly deflections under those excitations. The component stresses were shown to be less than the allowable limits based on Section III of ASME BPVC criteria. The core coolable geometry and CRA insertability are maintained for all the faulted loads and the component stress intensities are less than the allowable limits.

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The fuel assembly response to seismic excitations during refueling while the core is located in the reactor flange tool (RFT) was also studied. This evaluation was performed using the methodology described in Reference 4.2-5 with adjustments to account for the lower temperatures experienced in the RFT (Reference 4.2-9). While in the RFT, the fuel is already in a safe shutdown condition and therefore the RFT evaluation serves to confirm the structural integrity of the fuel rod in order to protect against the release of fission products. The same fuel rod analysis criteria from Section 4.2.1.5.3 were conservatively applied to the fuel in the RFT and the fuel rod stresses were shown to be less than the allowable limits as defined in Section 4.2.1.5.3.

4.2.3.5.3 Load Applied in Fuel Handling

Both the fuel assembly and individual components are evaluated for structural adequacy for shipping and handling loads in the amount of 6 g in the lateral direction and 4 g in the axial direction. The evaluations result in positive design margins to the stress limits.

4.2.3.5.4 Axial Growth

A fuel assembly top nozzle-to-fuel rod shoulder gap allowance is provided that maintains positive clearance during the assembly lifetime. The evaluation determined that a positive fuel rod shoulder gap occurs at end of life (EOL) hot conditions and considers the upper tolerance limit for fuel rod growth, minimum guide tube growth, and worst-case tolerances on the length of the fuel rods and guide tubes. The evaluated minimum fuel rod shoulder gap is presented in Reference 4.2-1 and is acceptable.

A fuel assembly-to-reactor internals gap allowance is provided that maintains a positive core plate gap clearance throughout the life of the fuel assembly. The core plate gap allowance considers combined worst-case internals-fuel assembly differential thermal expansion and irradiation-induced axial length changes to the guide tubes. The evaluation determined that a positive fuel core plate gap occurs at EOL cold conditions and considered the upper tolerance limit guide tube growth and worst-case tolerances on the length of the fuel rod and core plate separation. The evaluated minimum core plate gap is presented in Reference 4.2-1 and is shown to be acceptable.

4.2.4.5 On-line Fuel System Monitoring

The chemical and volume control system (Section 9.3.4) contains radiation detection instrumentation that continuously monitors for radioactivity and is capable of detecting a fuel leak. In addition, the process sampling system (Section 9.3.2) contains grab sample capability that allows for more detailed assessment of the radionuclides in the primary system water. Detection of a fuel leak may result in more frequent grab sample analysis.

4.2.4.6 Post Irradiation Monitoring

A detailed surveillance program is planned following the irradiation of the fuel assembly and CRAs from the first licensed module. This program includes the schedule and criteria for inspection of selected fuel assemblies and CRAs. The program includes complete visual inspections of selected assemblies and detailed measurements to capture key attributes such as those listed below. The detailed measurements are taken to confirm that the fuel is performing according to the design analyses described in Section 4.2. The key attributes that are assessed as part of the post-irradiation monitoring program include:

- fuel rod growth
- fuel rod bowing
- fuel assembly growth
- fuel assembly bowing
- crud deposition
- fuel rod cladding corrosion (oxide)
- fuel rod cladding diameter

In addition, visual inspection of guide tubes and control rod cladding is performed for indications of wear.

The post-irradiation program makes sure that the above characteristics are within expected values. This surveillance program is expected to span the initial three cycles of operation of the initial licensed NuScale Power Module, with provisions for during-cycle inspections if operation indicates the presence of fuel abnormalities. The surveillance program includes guidance on the disposition of failed fuel.

4.2.5 References

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4.2-1 NuScale Power, LLC, "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," TR-0816-51127-P, Rev. 2~~0~~, ~~December~~July 2019~~6~~.

4.2-2 AREVA Inc., "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," BAW-10227P-A, Rev. 1, June 2003.

4.2-3 NuScale Power, LLC, "Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-P-A, Rev. 1, June 2016.

4.2-4 AREVA Inc., "COPERNIC Fuel Rod Design Computer Code," BAW-10231P-A, Rev. 1, January, 2004.

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4.2-5 AREVA Inc., "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," ANP-10337P-A, Revision 0, April 2018.

4.2-6 AREVA Inc., "Computational Procedure for Evaluating Fuel Rod Bowing," XN-75-32-P-A, Supplements 1-4, February 1983.

4.2-7 AREVA Inc., "Generic Mechanical Design Criteria for PWR Fuel Designs," EMF-92-116(P)(A), Rev. 0, February 1999.

4.2-8 AREVA Inc., "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," BAW-10084P-A, Rev. 3, August 1995.

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4.2-9 NuScale Power, LLC, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," TR-0716-50351-P, Rev. 1~~0~~, ~~September~~July 2019~~6~~.

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

Table 1-1 Abbreviations

Term	Definition
AIC	silver-indium-cadmium
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BOL	beginning of life
CFM	centerline fuel melt
CFR	Code of Federal Regulations
CRA	control rod assembly
CUF	cumulative usage factor
EFPY	effective full power year
EOL	end of life
FIV	flow-induced vibration
FSAR	~~f~~Final ~~s~~Safety ~~a~~Analysis ~~r~~Report
GDC	~~g~~General ~~d~~Design ~~c~~Criteria
LHR	linear heat rate
LOCA	loss-of-coolant accident
NPM	NuScale ~~p~~Power ~~m~~Module
NRC	Nuclear Regulatory Commission
OD	outside diameter
PDC	principal design criterion
PHTF	portable hydraulic test facility (Richland)
PWR	pressurized water reactor
QD	quick disconnect
RCCA	rod cluster control assembly
RCS	reactor coolant system
RFT	reactor flange tool
RMS	root mean square
SRSS	square root of the sum of the squares
SSE	safe shutdown earthquake
TCS	transient cladding strain

Table 4-6 Summary of NuScale fuel assembly damping ratios for when the fuel is located in both the NuScale Power Module and the reactor flange tool

[ ]

The following model parameters are established through design-specific characterization testing:

- [ ]
- [ ]

is measured at key axial locations (e.g., spacer grids and top nozzle). The measurements of location-specific axial stiffness are used to benchmark the stiffness of the grid-to-fuel rod slider elements in the vertical model.

In the event of a fuel assembly drop, two impact mechanisms require characterization. The nozzle-to-core plate gap stiffness and damping is established by a dynamic drop test of the fuel assembly. In this test, full-scale prototypical NuFuel-HTP2™ fuel assemblies in both the BOL and EOL condition were dropped onto a rigid surface from varying heights. **[**

]

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

]

[

]

4.3.5.1.3 Excitation Inputs

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

~~The SSE input motions are the result of an evaluation of multiple, independent sets of soil-structure interaction parameters and module locations. For the certified seismic design response spectrum, considering six soil/rock profiles, cracked and uncracked concrete conditions, and two bounding reactor module configurations, a total of 24 variations are examined. For the generic high frequency hard rock response spectrum, considering two soil/rock profiles, cracked and uncracked concrete conditions, and two bounding reactor module configurations, a total of eight variations are examined.~~The SSE input motions are the result of an evaluation of a single soil-structure profile (S7). For the certified seismic design response spectrum, considering cracked and uncracked concrete conditions and two bounding reactor module configurations, a total of four variations are examined.

To account for the effect of uncertainty in the reactor module dynamic analysis, each time history variation is analyzed with three different scaled time intervals: the reference interval and plus or minus 15%. The frequency shift due to the 15% variation of the time scale is considered to be effectively equivalent to the broadening of spectral peaks that is done when generating in-structure response spectra. Considering the defined variations, a total of ~~96~~12 time histories are considered in the analysis.

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

In addition to the excitation inputs considered for the NPM, seismic core plate motions were also evaluated for the condition where the reactor core is stored in the reactor flange tool (RFT) during refueling. The RFT evaluation is based on the same limiting seismic motion considered for the NPM analysis and considers the same uncertainties, in the form of frequency shifts. There is no LOCA excitation considered for the RFT analysis.

~~The input time histories establish the basis for the fuel design. Subsequent changes to these inputs are evaluated against this base case to ensure the analysis remains bounding. The seismic inputs included in the NuScale Power Module Seismic Analysis technical report (Reference 9.1.14) are evaluated relative to the base case; the evaluation demonstrates that the fuel loads resulting from the spectra in Reference 9.1.14 remain bounded.~~

4.3.5.2 Analysis Results

4.3.5.2.1 Lateral Analysis

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

The peak impact loads for the base case seismic inputs~~, along with~~ and the margin to the grid impact load limit~~,~~ from all cases are summarized in Table 4-7. The peak impact loads for SSE and LOCA in a given direction are combined by the square root of the sum

of the squares (SRSS) method and margin is calculated against this SRSS impact load. The positive margin for these impact loads confirms that the NuScale spacer grid will not experience plastic deformation that exceeds the limit established in the AREVA methodology (Reference 9.1.5). Thus, the requirements for core coolability and control rod insertion are met.

A lateral seismic analysis of the fuel while the core resides in the RFT is also performed. In this configuration, the core is already contained in a safe shutdown condition and post-LOCA coolability is not a factor. Therefore, the only focus of the RFT seismic analysis is to evaluate the structural integrity of the fuel rod cladding in order to ensure that this fission product barrier is not breached. The results of the RFT lateral analysis provide the load inputs to be considered for this fuel rod evaluation.

Table 4-7 Peak grid impact loads and margins

[

]

4.3.5.2.2 Vertical Analysis

The single assembly vertical model is subjected to vertical core plate displacement time histories corresponding to the SSE and LOCA events. The maximum seismic impact load for the base case inputs, **[** **]** The maximum LOCA impact load is **[** **]** Component loads for the guide tubes, fuel rods, hold-down spring, nozzles, and guide tube connections are extracted from the vertical analysis for further load analysis. A bounding vertical seismic load of [ ] is also calculated for the fuel residing in the RFT. This load is considered in calculating the component stresses for the fuel rod in the RFT.

4.3.5.2.3 Stress Analysis

The lateral and vertical analysis results are used as inputs to load and stress evaluations of the non-grid fuel assembly components. Lateral and vertical loads are combined, along with steady-state normal operating loads, for each component for comparison to its respective acceptance criteria.

As defined in Section 8.1.2 of Reference 9.1.5, **[**

]

Table 4-8 Component evaluation margins

4.4 Thermal Hydraulic Evaluation

4.4.1 Core Pressure Drop Evaluation

An evaluation is performed of the fuel assembly pressure drop characteristics. The recoverable and unrecoverable pressure drop is determined for each of the 37 fuel assemblies in the reactor core over a wide range of core power and system flow

9.0 References

9.1 Source Documents

9.1.1 ANP-10231PA-01, “COPERNIC Fuel Rod Design Computer Code,” January 2004.

9.1.2 ANP-10084PA-03, “Program to Determine In-Reactor Performance of B&W Fuels – Cladding Creep Collapse” (CROV computer code), October 1980.

9.1.3 ASME Boiler and Pressure Vessel Code, Section III, Division 1 – Subsection NG, Core Support Structures, 2010 Edition with 2011a Addenda, July 1, 2011.

9.1.4 XN-75-32 (P) (A), Supplements 1-4, Computational Procedure for Evaluating Fuel Rod Bowing, February 1983.

9.1.5 ANP-10337P-A, Rev. 0, “PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations,” April 2018~~August 2015~~.

9.1.6 EMF-92-116(P)(A), Revision 0, “Generic Mechanical Design Criteria for PWR Fuel Designs”, February 2015.

9.1.7 TR-0116-20825-P, Applicability of AREVA Fuel Methodology for the NuScale Design, Revision 1.

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

9.1.9 BAW-10227P-A,Revision 1, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel”, June 2003.

9.1.10 NUREG-0800, U.S. NRC Standard Review Plan Section 4.2 Rev. 3, “Fuel System Design”, March 2007.

9.1.11 Letter, James F. Mallay (Framatome ANP) to Document Control Desk (NRC), EMF-92-116(P), “Generic Mechanical Design Criteria for PWR Fuel Designs,” NRC:99:029, July 9, 1999.

9.1.12 Letter, Stuart A. Richards (NRC) to James F. Mallay (Framatome ANP), “Siemens Power Corporation Re: Request for Concurrence on Safety Evaluation Report Clarifications (MA6160),” November 3, 2000.

9.1.13 O’Donnell, W.J. and B.F. Langer, “Fatigue Design Basis for Zircaloy Components,” Nuclear Science and Engineering, Volume 20, pp. 1-12, September 1964.

9.1.14 TR-0916-51502, "NuScale Power Module Seismic Analysis," ~~January 2017~~April 2019, Revision 2.

Enclosure 3:

Framatome Affidavit of Gayle Elliott

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 in the enclosure to a letter to the Document Control Desk (NRC) from Mr. Zackary W. Rad (NuScale Power, LLC), Docket No. 52-048, with subject, "NuScale Power, LLC Submittal of Corrected Response to NRC Request for Additional Information No. 284 (eRAI No. 9225) on the NuScale Design Certification," 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: April 30, 2020

Gayle Elliott

Gayle Elliott