~~PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390~~
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March 30, 2016

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Topical Report TR-0116-20825, “Applicability of AREVA Fuel Methodology for the NuScale Design,” Revision 0 (NRC Project No. 0769)

REFERENCE: Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, “Final Schedule for Topical Report Submittals,” LO-0116-21371, dated January 28, 2016 (ML16029A315).

In the referenced letter dated July 22, 2015, NuScale Power, LLC (NuScale) provided an updated schedule for topical report submittals. NuScale provided a schedule indicating the intent to submit this topical report by March 31, 2016. Consistent with that schedule, NuScale hereby submits Topical Report TR-0116-20825, “Applicability of AREVA Fuel Methods for NuScale SMR,” Revision 0.

The purpose of this submittal is to request NRC review and approval of the assumptions, codes, and methodologies presented in this report for applying AREVA codes and fuel methodology to the NuScale design. NuScale respectfully requests that the acceptance review be completed in 60 days from the date of transmittal.

Enclosure 1 contains the proprietary version of the report entitled "Applicability of AREVA Fuel Methodology for the NuScale Design." NuScale requests this enclosure be withheld from public disclosure pursuant to 10 CFR § 2.390. The enclosed affidavits (Enclosures 3 and 4) support this request. Enclosure 3 pertains to the AREVA proprietary information to be withheld from the public while Enclosure 4 pertains to the NuScale proprietary information to be withheld from the public. AREVA proprietary is denoted by straight brackets (i.e., “[ ]”) while NuScale proprietary is denoted by double curly brackets (i.e., “{{ }}”.) Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR Part 810.

Enclosure 2 is the nonproprietary version of the report entitled "Applicability of AREVA Fuel Methodology for the NuScale Design."

This letter and its enclosures make no regulatory commitments and no revisions to any existing regulatory commitments.

~~The Enclosure contains Proprietary Information. Upon separation from the Enclosure, this letter is decontrolled.~~

Please contact Jennie Wike at 541-360-0539 or at jwike@nuscalepower.com if you have any questions.

Sincerely,

Thomas A. Bergman
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Attachment:
Enclosure 1: "Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-P, Revision 0, proprietary version
Enclosure 2: "Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-NP, Revision 0, nonproprietary version
Enclosure 3: AREVA Affidavit

Enclosure 4: NuScale Affidavit, AF-0316-48349

Enclosure 1:

"Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-P, Revision 0, proprietary version

Enclosure 2:

“Applicability of AREVA Fuel Methodology for the NuScale Design,” TR-0116-20825-NP, Revision 0, nonproprietary version

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Licensing Topical Report

Applicability of AREVA Fuel Methodology for the NuScale Design

March 2016

Revision 0

Docket: PROJ0769

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Department of Energy Acknowledgement and Disclaimer

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Abstract

This report documents the applicability of the following NRC approved codes and methods used by AREVA to evaluate performance of the NuScale fuel design.

- EMF-92-116(P)(A), Revision 0, Generic Mechanical Design Criteria for PWR Fuel Designs (Reference 6)
- BAW-10227P-A, Revision 1, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel (Reference 2)
- BAW-10231P-A, Revision 1, COPERNIC Fuel Rod Design Computer Code (Reference 4)
- BAW-10084P-A, Revision 3, Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse (Reference 1)
- BAW-10133P-A, Revision 1, and Addenda 1 and 2, Mark-C FA LOCA-Seismic Analyses (Reference 7)
- XN-75-32(P)(A), Supplements 1 through 4, Computational Procedure for Evaluating Fuel Rod Bowing (Reference 5)

The necessary modifications to the AREVA codes and methods to address the differences in the NuScale fuel design are identified and described in this report. This report also discusses the analysis methods used by AREVA to perform design certification calculations for the NuScale fuel design to address Section 4.2 of the NUREG 0800, Standard Review Plan (Reference 8). This report is intended to be used for the AREVA methodology which is applied to the NuScale fuel design, fuel mechanical analysis, and fuel thermal-mechanical analysis work. This methodology will be used to support the NuScale Design Certification Application.

NuScale requests NRC approval to apply the AREVA methodology as described within the above listed documents to the NuScale fuel design and analysis. NuScale also requests NRC approval of the modifications made to the AREVA codes and methods to apply to the NuScale fuel design.

Executive Summary

NuScale intends to use NRC approved AREVA methodology for portions of the fuel analysis completed for the design certification application, which are limited to the fuel design, fuel mechanical analysis, and fuel thermal-mechanical analysis work. This report documents the applicability of the following NRC approved codes and methods used by AREVA to evaluate performance of the NuScale fuel design.

The generic mechanical design criteria for Pressurized Water Reactor (PWR) Fuel Designs (Reference 6) are applicable to NuScale fuel analysis because the NuScale fuel rod design is **{{**

}}.2(a),(c),ECI

The NuScale fuel assembly, however, does not contain a hold-down spring so an axial fuel assembly growth multiplier is used to conservatively model this impact as discussed in Section 7 of this report.

The evaluation of advanced cladding and structural material (M5®) in PWR Reactor Fuel (Reference 2) is applicable to NuScale fuel analysis because the NuScale fuel assembly design uses only M5® fuel rod cladding.

The COPERNIC Fuel Rod Design Computer Code (Reference 4) is applicable to NuScale fuel analysis because both the operating parameters and the materials for the **{{**

}}.2(a),(c),ECI

The program to determine in-reactor performance of fuel cladding creep collapse (Reference 1) is applicable to NuScale fuel analysis because the NuScale fuel rod design is **{{**

}}.2(a),(c),ECI The creep collapse analysis cannot be performed at **{{**

}}.2(a),(c),ECI

The Mark-C Fuel Assembly (FA) LOCA-Seismic Analyses (Reference 7) is applicable to NuScale fuel analysis because the spacer grids are the same as the AREVA PWR 17x17 HTP™ fuel design spacer grid. Two adjustments to this methodology are required for the NuScale fuel analysis. Fuel assembly damping values specific to the NuScale design are derived. Also, a single assembly model for LOCA evaluations is used in place of a core bounce model. Both of these modifications are discussed in Section 8.

The Computational Procedure for evaluating fuel rod bowing (Reference 5) is applicable to NuScale fuel analysis since the NuScale fuel design has a smaller spacer grid span length than the AREVA PWR 17x17 HTP™ fuel design; the fuel design has a lower propensity for rod bowing.

This document demonstrates that these AREVA codes and methods may be extended to the NuScale fuel design for the Design Certification Application. This report also discusses the analysis methods used by AREVA to perform design certification calculations for the NuScale fuel design that address the guidance in Section 4.2 of the NUREG 0800, Standard Review Plan (Reference 8). The AREVA methodology is limited to the NuScale fuel design, fuel mechanical analysis, and fuel thermal-mechanical analysis work. Any modifications made to the NRC-approved AREVA codes and methods to address the differences in the NuScale fuel design are identified and described in this report.

NuScale requests NRC approval to apply existing AREVA methodology to the NuScale fuel design and analysis. The necessary modifications to AREVA methodology are described in detail and require NRC approval. NuScale also requests NRC approval of any necessary modifications that were made to the AREVA codes and methods to apply to the NuScale fuel design.

1.0 INTRODUCTION

This document establishes the applicability of the cited AREVA methodologies for use in NuScale fuel analyses. NuScale requests NRC approval to use existing AREVA methodology in the analysis of NuScale fuel for the design certification application. The necessary modifications to AREVA methodology are described in detail and require NRC approval.

1.1 *Purpose*

This report documents the applicability of the following NRC approved codes and methods used by AREVA to evaluate performance of the NuScale fuel design.

- EMF-92-116(P)(A), Revision 0, Generic Mechanical Design Criteria for PWR Fuel Designs (Reference 6)
- BAW-10227P-A, Revision 1, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel (Reference 2)
- BAW-10231P-A, Revision 1, COPERNIC Fuel Rod Design Computer Code (Reference 4)
- BAW-10084P-A, Revision 3, Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse (Reference 1)
- BAW-10133P-A, Revision 1, and Addenda 1 and 2, Mark-C FA LOCA-Seismic Analyses (Reference 7)
- XN-75-32(P)(A), Supplements 1 through 4, Computational Procedure for Evaluating Fuel Rod Bowing (Reference 5)

This report demonstrates that these AREVA codes and methods are applicable to the NuScale design, and describes the necessary modifications to the AREVA codes and methods to address the differences in the NuScale fuel design.

1.2 *Scope*

This report presents the analysis methods and codes used by AREVA to perform design certification calculations for the NuScale fuel design to address Section 4.2 of the NUREG 0800, Standard Review Plan (Reference 8). The AREVA methodology is applied to the fuel design, fuel mechanical analysis, and fuel thermal-mechanical analysis work. The NuScale specific methodology for performing the neutronics work, safety analysis, and the thermal-hydraulic analyses is not described in this report.

Table 1-1 Abbreviations

Term	Definition
AFA	Advanced Fuel Assembly
BWFC	B&W Fuel Company
DCA	Design Certification Application
DNBR	Departure from Nucleate Boiling Ratio
EGCF	End Gap Correction Factor
FGR	Fission Gas Release
FIV	Flow-induced Vibration
FRGPC	Fuel Rod Gas Pressure Criterion
ID	Inner Diameter
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
OD	Outer Diameter
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCS	Reactor Coolant System
SAFDL	Specified Acceptable Fuel Design Limits
SER	Safety Evaluation Report
SMR	Small Modular Reactor
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake
TD	Theoretical Density
TER	Technical Evaluation Report

2.0 BACKGROUND

AREVA methodology supporting the AREVA fuel design work was approved for use by the NRC for analyzing PWR fuel. This report describes how these approved methods are also applicable to the NuScale fuel design.

To assist in the review of the applicability of AREVA 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. Due to the height of 94 inches, 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 (TD) with a possible enrichment up to 5.0 weight percent ^{235}U consistent with current operating plant licensing requirements.

Zircaloy-4 (Zr-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 Coarse-mesh Filter Plate

The 304 stainless steel bottom nozzle consists of a cast frame of ribs connecting the guide thimble locations. A high strength A-286 alloy coarse 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 which 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. Given the relatively low core flow rate and low assembly pressure drop, the NuScale top nozzle does not require hold-down springs. The NuScale top nozzle design is slightly different with respect to the through hole requirements to accommodate the NuScale top-entry incore detectors.

Table 2-1 provides additional information on the NuScale fuel design and identifies the differences between the NuScale and AREVA 17x17 PWR fuel design.

Table 2-1 NuScale Fuel Design Parameters

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

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)	{{	16
System Pressure (psia)		2280
Core Tave (F)		584
Linear Heat Rate (kW/m)	}}[2(a),(c),ECI]	18.0
RCS Inlet Temperature (F)	503	547
RCS Reynolds Number	76,000	468,000
Maximum Fuel-Assembly-Average burnup (GWd/mtU)	{{ }}[2(a),(c),ECI]	51
Fuel Assemblies in Core	37	193
Fuel Assembly Loading (kgU)	{{	[
Core Loading (kgU)	}}[2(a),(c),ECI]	][ECI]

2.1 *Regulatory Requirements*

Table 2-3 shows the specified acceptable fuel design limits (SAFDLs) that are applicable to the NuScale analysis using AREVA methodology. This table also identifies which of the AREVA topical reports addresses the acceptance criteria in Section 4.2 of NUREG-0800, Standard Review Plan (Reference 8). Table 2-3 focuses only on the sections of the SRP that are being addressed using AREVA methodologies. Criteria not addressed in this table are addressed in other topical reports.

Table 2-3 SRP Criteria Review Summary

Analysis using AREVA Methodology	SRP 4.2 Acceptance Criteria (Reference 8)	AREVA Topical Report
Shipping And Handling Stress Analysis	1.A.i	EMF-92-116(P)(A) (Reference 6)
Fuel Assembly/Component Stress Analysis	1.A.i	
FIV Assessment	1.A.iii	
Axial Growth (Rod and Assembly)	1.A.v	
Fuel Lift Analysis	1.A.vii	
Internal Hydriding	1.B.i	
Clad Stress Analysis	1.A.i	BAW-10227PA (Reference 2)
Fuel Rod Buckling Analysis	1.A.i	
Clad Fatigue Analysis	1.A.ii	
Clad Corrosion Analysis	1.A.iv	BAW-10231PA (Reference 4)
Fuel Rod Internal Pressure	1.A.vi	
Fuel Centerline Melt Analysis	1.B.iv	
Transient Clad Strain Analysis	1.B.vi	
Clad Creep Collapse Analysis	1.B.ii	BAW-10084PA (Reference 1) BAW-10227PA (Reference 2)
Rod Bow Evaluation	1.A.v	XN-75-32PA (Reference 5)
LOCA/Seismic Stress Analysis	Appendix A	BAW-10133PA (Reference 7)

The applicability of each respective AREVA methodology listed in Table 2-3 to the NuScale fuel design is described in Sections 3 through 8 of this report.

3.0 REVIEW OF BAW-10084PA, PROGRAM TO DETERMINE IN-REACTOR PERFORMANCE OF BWFC FUEL CLADDING CREEP COLLAPSE

AREVA topical report BAW-10084PA-03 (Reference 1) documents and summarizes the creep collapse methodology used for AREVA fuel rod designs to ensure that the fuel rods do not collapse during their design lifetimes.

The approved CROV computer code is applicable to AREVA fuel designs but restricted to Zircaloy-4 cladding. AREVA Topical Report BAW-10227PA-01 "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," extended the application of the CROV code to fuel designs with AREVA's advanced cladding M5® (Reference 2).

3.1 *Applications*

The CROV methodology has been used to license various PWR fuel types (i.e., multiple B&W 15x15 and W-type 17x17 fuel designs). The NuScale fuel rod design is **[**

].ECI Below is a chapter by chapter summary of the CROV topical report. Chapter numbers used in this summary refer to the Chapter numbers in the BAW-10084PA-03 topical report.

Chapter 1 *Introduction*

This chapter provides the background and history of the fuel rod creep collapse calculations. The organization and approach for the topical report is also explained. A preliminary discussion of the use of the end gap correction factor (EGCF) is given. **[**

].ECI

Chapter 1 gives no limitations on the use of the topical report for NuScale fuel rod design.

Chapter 2 *Cladding Creep Properties*

This chapter provides the base equation for the cladding creep model and the basis for the use of the EGCF in the creep collapse calculations. The form of the cladding creep model is taken from the **[** **]** (Reference 3). The EGCF is developed based on **[**

].ECI Explanation of how the EGCF will be applied to the infinite tube model is discussed.

The **{{**

}}2(a),(c),ECI and therefore Chapter 2 is applicable for use with the NuScale fuel rod design.

Chapter 3 *Creep Collapse Analytical Method*

Chapter 3 gives the analytical method for the calculation of creep collapse. The section summarizes the CROV code calculations, presents the criteria for considering a rod collapsed, and verification of the calculation. A benchmark of the CROV calculations to the FRGPC topical creep database is presented and good agreement with the measured diameters is shown. The criteria that the code uses to determine if the rod is considered to be collapsed are outlined. The criteria for collapse are broken into four parts: **[**

]. How the code applies the collapse criteria is explained. A benchmark of the CROV code collapse predictions to the Ginna fuel rod collapses is given (fuel rod collapses in the Ginna reactor in 1972 prompted the industry to analyze the mechanism which caused fuel rod cladding creep collapse).

The **{{**

}}.2(a),(c),ECI Additionally, the cladding of the NuScale fuel design will be exposed to pressure differentials which are bounded by that of existing AREVA fuel designs due to the lower system pressure. Therefore, Chapter 3 is applicable to NuScale fuel design.

Chapter 4 *Determination of Creep Collapse Parameters*

Chapter 4 gives the basis for the fuel rod design and operational parameters to be used for the creep collapse calculations. The [ ]ECI to be used are summarized. References are made in this section to the TACO3 fuel rod performance code. The TACO3 code has been replaced by the COPERNIC code. The COPERNIC topical report (Reference 4) discusses the use of COPERNIC for generating [ ] to the creep collapse analysis (CROV calculation).

The [ ]. Additionally, the NuScale design [ ] are bounded by that of conventional PWRs. Therefore, Chapter 4 is applicable to NuScale fuel design.

Chapter 5 *References*

Chapter 5 contains the references for the topical report. This section is applicable to NuScale fuel rod design.

Appendices D and E

Appendices D and E contain the responses to the NRC requests for additional information (RAI). The initial part of this section contains the basis for the update of the CROV code to use the creep equations contained in the FRGPC topical report. After the initial section, a number of questions and responses are included which range from cladding creep databases, formulation of the creep model, to the derivation of the end gap correction factor. Also included within the questions and responses are detailed examples of how creep collapse calculations are [ ].

As stated in the sections above, the NuScale fuel design **[** **]** . Additionally, the relevant NuScale design **[** **]** are bounded by that of conventional PWRs. Therefore, Appendices D and E are applicable for use with the NuScale fuel design.

3.2 *Topical Report Restrictions*

3.2.1 Safety evaluation report

The SER for BAW-10084PA-03 references the TER performed for the CROV code by Pacific Northwest Laboratory. The SER approves the usage of BAW-10084PA-03 for all AREVA fuel reloads. The SER for the CROV topical report makes no restrictions as to fuel design type. The SER for BAW-10084PA-03 is applicable to the NuScale fuel design.

3.2.2 Technical evaluation report

The TER evaluation reviews the updated fuel cladding creep model proposed for usage and the **[** **]**. The TER finds the improvements to the creep collapse modeling to be acceptable within a **[**

] ECI

Thermal-mechanical analyses for the NuScale fuel predict **[** **]** and lower than that of PWRs currently in operation due to the significantly lower rod linear heat generation rates.

The NuScale plant design falls within the ranges evaluated and approved in the SER and TER.

3.3 *NuScale Design Differences and Requirements*

3.3.1 Axial Gap for Creep Collapse

The creep collapse analysis is performed at an **[** **]** which results in the shortest calculated collapse time (page 4-2, Section 4.2 of Reference 1).

The conclusion for the NuScale fuel design creep collapse analysis is that the application of **[** **]** , as is done for current applications of the CROV methodology, is conservative. A revised **[** **]** is consistent with the approved CROV methodology and will not result in cladding collapse over the irradiation lifetime of the fuel based on results of current AREVA fuel.

3.4 *Summary and Conclusions for BAW-10084PA CROV Topical Report*

BAW-10084PA-03 documents and summarizes the creep collapse methodology used for AREVA fuel rod designs to ensure that the fuel rods do not collapse during their design lifetimes. The approved methodology is applicable to the NuScale fuel design with a minor **[** **]** of the NuScale fuel rods.

4.0 REVIEW OF BAW-10227PA M5® MATERIAL TOPICAL REPORT

AREVA topical report BAW-10227PA-01 (Reference 2) documents and summarizes the methodology used for AREVA M5® fuel rod cladding to ensure that the fuel rod design criteria are met. The NuScale fuel assembly design uses only M5® fuel rod cladding. No assembly structural components are made of M5® material and therefore these components are not discussed in this applicability report.

4.1 *Applications*

The BAW-10227PA-01 methodology has been used to license various PWR fuel types (multiple B&W-type 15x15 and W-type 17x17 fuel designs). The NuScale fuel rod design is **[**

].ECI Below is a chapter summary of BAW-10227PA-01.

Chapter 1 *Introduction and Summary*
This chapter provides the background and history of the M5® fuel rod cladding and structural material development and operating experience. The primary benefits of the use of M5® material are presented. Chapter 1 gives no limitations on the use of the topical report for NuScale fuel design.

Chapter 2 *Fuel Assembly Mechanical Design*
This chapter provides information related to the use of M5® material in the structural components of the fuel assembly. The NuScale fuel design does not use M5® as a structural material. Hence, Chapter 2 is not applicable to the NuScale fuel design.

Chapter 3 *Fuel Rod Design Requirements and Analysis Results*
Chapter 3 gives the analytical methods and design requirements for fuel rods with M5® cladding. In particular, Sections 3.3 and 3.6 are relied upon to demonstrate compliance with the fuel rod thermal mechanical design criteria.

The **[**

].ECI

Therefore, Chapter 3 is applicable for use with NuScale fuel rod design.

Chapter 4 *Accident Criteria and Evaluation*

Chapter 4 gives the bases for the fuel rod accident criteria and evaluations. AREVA accident analysis methodology will not be applied to the NuScale fuel design.

Chapter 5 M5® Material Properties

Chapter 5 contains the M5® material properties which are primarily a function of **[**

] experienced in the NuScale design will be bounded by that of conventional PWRs due to the lower linear heat rate. Therefore, this section is applicable to NuScale fuel design.

Chapter 6 References

Chapter 6 contains the references for the topical report. This section is applicable to NuScale fuel design.

Appendix A through D M5® Material Properties and Models

Appendices A through D provide detailed information on M5® material properties and analysis models. Appendix A material properties and models, with respect to **[**

] , are appropriate for use in NuScale fuel rod design certification calculations because the NuScale core conditions are within the range of past and current PWR operating conditions.

Appendix E Post Irradiation Examination (PIE) Plans

Appendix E provides detailed information on M5® PIE plans. This appendix is not relevant for determining NuScale applicability.

Appendix F and G

Appendices F and G provide detailed information on the LOCA analyses and limits of fuel with M5® clad. AREVA LOCA analysis methodology will not be applied to the NuScale fuel design.

Appendix H

Appendix H provides detailed information about the use of stainless steel replacement rods in fuel assemblies with M5® clad and structure. This appendix is not relevant for determining NuScale applicability.

Appendix I, K, and L

Appendices I, K, and L contain the responses to the NRC RAIs. Responses to RAIs I2 and K1 are related to M5® clad yield strength and ultimate strength. New values for these limits were provided. Updated values for material properties are prepared for M5® on a periodic basis. The appropriate values are used in design analysis. Reponses to RAIs K.2.2 and K.11 are related to M5® fatigue. Additional data presentations were performed. The methods described in the main body of the topical report are unchanged.

The fuel rod design for the NuScale design is **[**

] (see Table 2.1). The NuScale design operating conditions (i.e., core power and core temperatures) are also less demanding than those of conventional PWRs. Therefore, the results for NuScale thermal mechanical analyses are less limiting than those for existing PWR fuel. The SRP and the set of specific RAIs for BAW-10227PA-01 are applicable to the NuScale fuel rod thermal mechanical design.

Appendix J

Appendix J provides detailed information about the M5® **[**

].ECI Therefore Appendix J has no specific relevance to the NuScale design applicability.

4.2 *Topical Report Restrictions*

4.2.1 Safety evaluation report

The safety evaluation for BAW-10227PA-01 (Reference 2) was performed by the NRC with Pacific Northwest National Laboratory acting as a consultant. The SER, inclusive of the TER, approves the usage of BAW-10227PA-01 for AREVA PWR fuel reload designs. In general, the topical report is limited in application to fuel rod burnups below **[** **]**ECI for AREVA fuel designs. This limit will continue to be applied to the NuScale fuel design. The SER reviews the M5® fuel cladding design criteria and models proposed for usage. The SER finds Sections 3.3 and 3.6 of the topical report to be acceptable without limitation. The SER for BAW-10227PA-01 is applicable to the NuScale fuel rod thermal mechanical design.

4.3 *NuScale Design Differences and Requirements*

4.3.1 NuScale Design Applicability

The SER for the M5® topical report (Reference 2) makes no restrictions as to fuel design type. The M5® topical report has been applied on a generic basis to AREVA fuel designs.

4.3.2 COPERNIC Input and Output Parameters

The COPERNIC fuel performance code (Reference 4) is used to generate the fuel rod performance parameters for **[** **]**. Inputs to COPERNIC include the fuel rod geometry, power history, back-fill pressure, fast flux and densification kinetics. These inputs are specific to the NuScale fuel design and NuScale design core conditions which are bounded by past and current PWR operating conditions. The COPERNIC fuel performance code outputs the fuel rod internal pressure and cladding

temperatures and dimensions. The applicability of the COPERNIC code to NuScale fuel design is discussed in detail in Section 5 of this report.

4.4 *Summary and Conclusions for BAW-10227PA M5® Topical Report*

BAW-10227PA-01 documents and summarizes the M5® thermal mechanical methodology used for AREVA fuel rod designs. The methodology described and approved is applicable to the NuScale fuel design.

5.0 REVIEW OF BAW-10231PA, COPERNIC FUEL ROD DESIGN TOPICAL REPORT

AREVA COPERNIC topical report BAW-10231PA, Revision 1, (Reference 4) documents AREVA's primary tool to evaluate PWR fuel rod thermal and mechanical performance. This section reviews the COPERNIC topical report and identifies the range of applicability of the COPERNIC fuel performance code and models to the NuScale fuel rod.

This review addresses the major areas of the COPERNIC code including thermal models, fission gas release, pellet and cladding mechanical models, and corrosion.

5.1 *Applications*

5.1.1 Method of review

When evaluating the fuel rod thermal mechanical performance, rod manufacturing characteristics, thermal hydraulic condition, and fuel rod irradiation history are required inputs. These three inputs for the NuScale fuel design are discussed in the following paragraphs.

Based on the information in Section 2 of this report, the materials for the **[**

].[ECI] Therefore, the **[**

] are applicable to the NuScale design. The COPERNIC validity range is listed in Table 5-1.

Comparing the thermal hydraulic information in Table 2-2, the NuScale design system pressure is lower than that of a typical PWR. With the lower system pressure, the coolant saturation temperature decreases approximately **{{ }}**,[2(a),(c),ECI] which lowers the cladding surface temperature in the two phase flow region. Average coolant velocity is approximately **{{ }}**[2(a),(c),ECI] in a typical PWR. The slower coolant velocity will reduce the coolant-cladding outside heat transfer. Coolant average temperature is comparable to that of a typical PWR.

Based on core design information, the fuel enrichment of the NuScale fuel is within the range encountered in a typical PWR. The core average linear heat generation rate (LHGR) is around **{{ }}**,[2(a),(c),ECI] which makes the maximum fuel rod LHGR much less than that of a typical PWR, which is around 18 kW/m. Maximum predicted fuel rod burnup is approximately **{{ }}**[2(a),(c),ECI] and the maximum fuel rod fluence is approximately **{{ }}**.[2(a),(c),ECI] The exposure and fluence are similar to the twice-burned fuel rod in a typical PWR. Table 2-2 provides a comparison of the operating conditions representative of the NuScale design to a typical PWR. Based on these values, the fuel rod in the NuScale design operates in a relatively low power, low exposure, and low fluence range when compared to a typical PWR.

The following sections discuss the major models and databases used in COPERNIC.

5.1.2 Code Validity Range

The overall applicability of COPERNIC code is listed in Table 5-1 and Chapter 1 of the topical report (Reference 4).

[ ] ECI

5.2 *COPERNIC Models Review*

The following sections review the various models within COPERNIC for applicability to the NuScale fuel and operating conditions.

5.2.1 Thermal Models

Thermal models presented in the Reference 4 topical report include coolant-cladding outside surface heat transfer, pellet-cladding heat transfer, fuel thermal conductivity, and pellet radial power profile.

5.2.1.1 Coolant-cladding outside surface heat transfer

In a natural circulation reactor, the flow rate is a strong function of power. The flow conditions at steady-state full power operation are addressed in this document. Two heat transfer models are used in the COPERNIC code. One is the **[ ]** single-phase forced convection heat transfer coefficient and the other one is **[ ]** relationship for two phase forced convection.

Coolant pressure, mass flux and fuel rod surface heat flux will impact the heat transfer coefficient in the **[ ]** correlation. In the NuScale design, the heat source (fuel rod) and the elevated heat sink (riser) act together like a pump to drive the fluid flow through the core. At the coolant-cladding heat transfer side, typical PWR designs and the NuScale design are similar except that the NuScale design will have a lower flow rate and a smaller Reynolds number. For the NuScale design, the Reynolds number is about 20 percent of the Reynolds number in a typical PWR. The Reynolds number for the NuScale design during steady state full power operation is approximately 76,000, which is much higher than 10,000 (a threshold above which forced convection is typically seen). Thus, forced convection is the appropriate mode of heat transfer for the NuScale design and operating conditions. **[**

].

[ ]. The NuScale RCS pressure is expected to fall within that range. The correlation applies to all geometries and for both local and bulk boiling conditions. Hence, the usage of **[ ]** is appropriate.

5.2.1.2 Pellet-cladding heat transfer

The pellet-cladding heat transfer prediction in COPERNIC has three parts: thermal radiation, gaseous conductance, and contact conductance. **[**

]. These models are validated by Garnier and Begej data (Reference 15). [

].[ECI] The NuScale design is expected to have a lower linear heat rate generation, hence lower fission gas release, lower rod internal gas pressure, and lower contact pressure. The fuel rod operation conditions for the NuScale design, such as contact pressure and gas pressure, are expected to be bounded by the conditions in a typical PWR, and therefore, the pellet-cladding heat transfer model is applicable for the NuScale design.

5.2.1.3 Fuel Thermal conductivity

The selection of the fuel thermal conductivity model in COPERNIC is based on the proposed fuel thermal conductivity models in [

] , but it is acceptable due to the data uncertainty and good agreement of the temperature prediction between COPERNIC and the NRC model. Since the NuScale fuel design features, [ ] are identical to typical 17x17 PWR fuel, the fuel thermal conductivity model is acceptable for NuScale design.

5.2.1.4 Pellet radial power profile

The COPERNIC fuel pellet radial power profile table for UO_2 fuel depends on the fuel composition and burnup. The UO_2-Gd_2O_3 radial power profiles are provided by an AREVA neutronics code. The range of application, shown in section 4.5.5 of Reference 4 and in Table 5-2, is sufficient to cover the NuScale application. Since the NuScale fuel is the same type of fuel as used in a typical PWRs, the pellet radial power profiles are acceptable for the NuScale fuel design.

[ ]ECI

5.2.1.5 Global validation on thermal models

COPERNIC has been through a global validation by comparing the COPERNIC centerline fuel temperature prediction with measured data to verify that the thermal models are interacting satisfactorily. Based on the operating conditions for the NuScale design, the fuel rod will have lower burnup and LHGR relative to a typical PWR fuel rod. The maximum NuScale fuel rod burnup is approximately **{{ }}**2(a),(c),ECI and the core average LHGR is **{{ }}**.2(a),(c),ECI The AREVA database supporting the COPERNIC topical report has more measurement data **[ ]**.ECI Figure 4-73 of the topical report shows the trend is well suited and temperature prediction does not have any bias relating to LHGR. Figure 4-70 of the topical report demonstrated that the temperature is **[ ]**.ECI The database shown in Tables 4-4 and 4-5 of the Reference 4 topical report fully covers the NuScale design operating conditions in terms of burnup and LHGR. Note that the Halden data in Table 4-5 of the Reference 4 topical report have a coolant mass flux of **[ ]**,ECI which forms the lower bound of the flow rate in the database. For full power operation in the NuScale design, the mass flux is around **{{ }}**.2(a),(c),ECI Based on this evaluation, the COPERNIC thermal models are acceptable for the NuScale fuel rod design.

5.2.1.6 Fission Gas Release (FGR) Model

The fission gas release model in COPERNIC includes steady state fission gas release and transient fission gas release. The steady state fission gas release (FGR) includes athermal and thermal components. The athermal component is relatively small and dependent on the burnup and fuel microstructure. The thermal release is activated at a burnup dependent temperature threshold. The transient FGR considers the enhanced diffusion and burst release. The experimental database for FGR model calibration is shown in Table 5-3. As shown in Figure 5-9 and 5-10 of the topical report for steady state fission gas release, there are sufficient data to envelope the intended burnup range of the NuScale fuel rod. The majority of measurement data are **[ ]**,ECI where the NuScale fuel rod will operate. Figure 5-11 of the topical report also demonstrates that COPERNIC predicts the transient FGR realistically. In addition, the use of the same manufacturing process for NuScale fuel pellets ensures that the microstructure of the pellet will be the same as standard AREVA fuel. Finally, fission gas release is a primary function of fuel temperatures. Section 5.2.1.5 demonstrates that COPERNIC correctly predicts fuel temperatures for the NuScale fuel design and therefore the FGR is likewise correctly predicted. Therefore, the fission gas release model is acceptable for the NuScale fuel rod.

[ ]ECI

5.2.2 Mechanical models

5.2.2.1 Pellet specific mechanical models

The COPERNIC topical report discusses **[** **]**ECI models. These models are a function of **[** **]**ECI will be lower than that of typical PWRs where these models are already applied, due to the lower linear heat rate. Therefore the COPERNIC pellet mechanical models are applicable to the NuScale design.

5.2.2.2 Cladding specific mechanical models

The cladding strain, either thermal or irradiation strain, is a function of **[** **]**ECI level is from the measurements on fuel rods before pellet-cladding gap closure. The **[** **]**.ECI Combining the calibration database and validation database, Section 7.1.3.4 and 7.2.3.4 of the topical report lists the validation range for both **[** **]** (see Table 5-4). Note that the overall application range of COPERNIC is shown in Chapter 1 of the topical report. Also, the stress values in the following table do not differentiate between the compressive stress and tensile stress. Before the fuel pellet contacts the cladding, the cladding experiences compression. After pellet to cladding contact, the cladding experiences tension.

[ ]ECI

As stated in the previous paragraph, [ ].ECI Due to the low LHGR, the NuScale fuel rod is expected to experience lower stress levels and slower gap closure when compared to a typical PWR. The NuScale fuel rod is expected to have lower burnup and fluence compared to a typical PWR. The calibration database for the [ ]. In conclusion, both temperature and fluence experienced by the NuScale fuel rods are bounded by the AREVA measurement database.

During a fast power increase, there is a strong mechanical interaction between cladding and pellet and the [ ].ECI In conclusion, the application conditions for which COPERNIC has been approved bound the conditions that the NuScale fuel rods are expected to experience.

5.2.3 Cladding Corrosion and Hydriding Models

The M5® waterside corrosion model is formulated in a classical way [

].

The calibration database consists of [

],ECI per RAI Question 19). The highest oxide thickness measurement was obtained with an [ ].ECI The axial location with maximum oxide thickness is typically upstream of the intermediate grids in the last one or two spans. The outlet temperatures in the measurement database bound the NuScale outlet temperature. As shown in Figure 8-16 of the Reference 4 topical report, there are sufficient measurement data [

],ECI which is appropriate to cover the operation range of NuScale fuel rods.

Inlet and outlet temperatures are below the saturation temperature at the NuScale operating pressure. Hence, local boiling does not exist or is limited. Based on the current M5® data base and NuScale-specific analyses, the corrosion level for M5® cladding is expected to be significantly less than the [ ].ECI The hydrogen pickup model is consistent with the M5® database. Overall, the corrosion model and hydrogen pickup model are both acceptable for evaluating NuScale fuel.

5.3 *Material Properties*

The NuScale fuel rod cladding and fuel pellet composition are expected to be the same as those in typical AREVA PWR fuel. Hence, all the approved physical and mechanical properties, such as yield strength, tensile strength, Young's modulus, and Poison's ratio are acceptable for NuScale application.

5.4 Safety Evaluation and Technical Evaluation Report

The safety evaluation for BAW-10231PA-01 (Reference 4) was performed by the NRC with Pacific Northwest National Laboratory acting as a consultant. The SER, inclusive of the TER, approves the usage of BAW-10231PA-01 for fuel licensing applications up to a **[ ]**.[ECI] This limit will continue to be applied to the NuScale fuel design. Therefore, the SER for BAW-10231PA-01 (Reference 4) is applicable to the NuScale fuel rod thermal mechanical design.

5.5 Summary and Conclusions for BAW-10231PA COPERNIC Topical Report

BAW-10231PA-01 documents and summarizes AREVA's fuel performance code for evaluating fuel rod thermal and mechanical performance. The methodology described and approved is applicable to the NuScale design fuel design and therefore COPERNIC may be used to predict NuScale fuel rod performance and attributes (rod internal pressure, cladding and pellet temperatures, etc.).

6.0 REVIEW OF XN-75-32(P)(A) SUPPLEMENTS, COMPUTATIONAL PROCEDURE FOR EVALUATING FUEL ROD BOWING TOPICAL REPORT

AREVA's approved methodology for analyzing the effects of fuel rod bowing and determining the power peaking factor (FQ) and departure from nucleate boiling ratio rod bowing penalties is documented in topical report XN-75-32(P)(A) Supplements 1-4 (Reference 5).

6.1 Applications

Fuel rod bowing is the deviation from straightness of the fuel rods in the fuel assembly. The presence of fuel rod bowing is identified by the change in water channel gap (distance between fuel rods) from nominal conditions. The major concerns addressed in this topical report are the potential effects on local power distribution and on the margin of fuel rods to departure from nucleate boiling (DNB). Since the term critical heat flux (CHF) is more characteristic of the phenomenon that would lead to failure for the operating conditions of the NuScale core, CHF will be used as opposed to DNB in this discussion. The secondary potential effect of fuel rod

bowing involves fuel clad fretting occurring at 100 percent gap closure, though the probability of this phenomenon is low.

Supplement 1

The combination of Supplements 1 through 4 is the NRC-approved fuel rod bowing methodology basis that is generally applied by AREVA to fuel designs equipped with HTP™ spacer grids. Supplement 1 provides the rod bowing (gap spacing) measurement database, the statistical treatment of the gap spacing data, the statistical evaluation of the area change due to rod bowing, the thermal-hydraulic rod bow model, the neutronics effects calculational methods and models, and the consideration of other bowing effects (e.g., fuel assembly bowing). Supplement 1 is applicable to the NuScale fuel design.

Supplement 2

This supplement provides a description of the application of the AREVA Generic Rod Bow Methodology to the fuel designs for Combustion Engineering type plants. Supplement 2 is applicable to the NuScale fuel design.

Supplement 3

This supplement reflects the inclusion of pre- and post-irradiation gap spacing measurements for AREVA fuel prior to 1980 in establishing an empirical relationship describing rod bowing (i.e., gap closure) as a function of fuel assembly exposure as well as comparing the measurements against the basic models. Supplement 3 is applicable to the NuScale fuel design.

Supplement 4

This supplement provides responses to the NRC questions on the AREVA empirically based rod bow methodology defined in Supplements 1, 2, and 3. Supplement 4 is applicable to the NuScale fuel design.

The NRC approval for the use of the above four supplements for the HTP™ spacer grid design is stated on page 7 of the Safety Evaluation Report for ANF-89-060(P)(A), Supplement 1 (Reference 9).

6.1.1 Fuel Design Comparison

Fuel rod bowing is primarily influenced by the following fuel design characteristics:

- the slip loads of the HTP™ intermediate and upper end spacer grids where slip load is the force necessary for the fuel rod to slide through the fuel rod support structure of the spacer grid,
- the fuel rod cladding thickness,
- the cladding material,
- the spacer grid design, and
- the span length between the fuel rod supporting spacer grids.

Fuel rod bowing is also influenced by the following operating environment parameters:

- the coolant cross flow forces,
- the coolant temperature and its impact on the fuel rod cladding material creep rate, and
- the fuel assembly burnup.

Characteristics of the NuScale fuel design are compared in Table 2-1 to those for the AREVA 17x17 PWR fuel designs with HTP™ intermediate structural grids. The noted differences are as follows:

- fuel assembly height,
- active fuel stack length,
- spacer grid span lengths, and
- fuel rod internal pressure.

Of these differences, only the spacer grid span length has a notable influence on fuel rod bow propensity. Because the NuScale fuel design has a smaller spacer grid span length than the AREVA PWR 17x17 HTP™ fuel design, it has a lower propensity for rod bowing. Furthermore, the NuScale fuel design does not introduce any design changes relative to the existing AREVA 17x17 HTP™ 12-foot PWR fuel design which adversely impact the potential for fuel rod bowing.

When considering the operating environment, the NuScale core and cross flow velocities (and the resulting lateral forces) as well as the core outlet temperature and fuel assembly burnup are lower than those for the AREVA Advanced 17x17 HTP™ fuel design currently in operation where no evidence of severe fuel rod bowing exists. As such, the NuScale fuel design does not exhibit a high propensity for fuel rod bowing. In addition, the fuel rod bowing data for current HTP™ fuel designs continue to demonstrate the adequacy of the applicability of the AREVA rod bow correlation in The NuScale fuel design. This design is within the current experience base with regard to fuel rod bending stiffness and core operating outlet temperature and less limiting regarding end grid slip loads and span lengths. Based on a comparison of these attributes, the NuScale fuel design has a relatively high resistance to fuel rod bowing, comparable with AREVA W-type 17x17 PWR fuel.

Therefore, when considering the fuel design and operating condition characteristics of the NuScale fuel design and core, the magnitude of fuel rod bowing is acceptable and comparable to other HTP™ fuel designs. The applicability of the Reference 5 methodology to the HTP™ fuel design has been approved by the NRC in Reference 9.

6.1.2 Justification of CHF Penalties

The XN-75-32(P)(A) rod bowing methodology uses a linear interpolation between 50 percent (threshold value for a CHF penalty) and 100 percent gap closure to determine the rod bowing CHF penalty. The model used in the determination of the CHF penalty includes an estimate of magnitude of gap closure as well as the reduction of critical heat flux in bowed geometry. The magnitude of gap closure is based on rod-to-rod spacing measurements, while the reduction of critical heat flux in bowed geometries is conservatively estimated from the results of critical heat flux testing for bowed geometries.

The reduction of CHF in bowed geometries given in the XN-75-32(P)(A) supplements (Reference 5) is based on an open literature correlation. The CHF test data, obtained from tests conducted at Columbia University, of a bowed-to-contact condition for a range of local mass velocities and pressures were obtained from the work by Nagino, et.al., in Reference 10. The range of tested conditions from the bowed-to-contact CHF tests was:

[ ]ECI

The nominal operating conditions for the NuScale core are approximately a pressure of **{{ }}**,[2(a),(c), ECI] a mass velocity of **{{ }}**,[2(a),(c), ECI] and an inlet temperature of 256 ^{o}C. These NuScale operating conditions are within the range of test conditions examined with the exception of the mass velocity which is approximately 1/3 of the minimum mass velocity tested. Figures 6-1 and 6-2 are provided that show the parameter of δ_{bow}, meas as a function of local mass velocity and pressure, respectively (Reference 10). The δ_{bow}, meas parameter effectively normalizes out any trends in the data except those due to the presence of the fuel rod bowing and to random scatter.

$$\delta_{bow,\,meas} = \frac{\left(\frac{q''_{meas}}{q''_{pred}}\right)_{no\ bow} - \left(\frac{q''_{meas}}{q''_{pred}}\right)_{bow}}{\left(\frac{q''_{meas}}{q''_{pred}}\right)_{no\ bow}}$$

Although the NuScale nominal mass velocity is below the minimum local mass velocity in the tested conditions, the trend in mass velocity at the lower mass velocity region in the figure is small as the NuScale nominal mass velocity is approached.

Based on the above comparison of tested conditions versus NuScale nominal operating conditions, it is concluded that the sensitivity obtained from the CHF tests for a bowed-to-contact condition at various pressure, inlet temperatures and mass velocities would not be significantly different at the NuScale nominal operating conditions. Therefore, the CHF penalty relationship in Reference 5 is considered a reasonable accommodation for the rod bowing based CHF penalty for the NuScale fuel design.

Additionally, in Supplement 4 of Reference 5, the RAI Question 46 response addressed whether a flow dependence should be considered in the CHF penalty. The response noted, **[**

]

As stated in Supplement 1, the CHF penalty determined from **[**

]. Therefore, based upon the comparison of CHF test conditions and NuScale operating conditions, the procedure for the calculation of the CHF penalty for potential rod bowing from the supplements is concluded to be applicable to the NuScale fuel design.

Figure 6-1: Bow Effect Parameter (δ_{bow}, meas) as a Function of Local Mass Velocity

Figure 6-2: Bow Effect Parameter (δ_{bow}, meas) as a Function of Inlet Pressure

6.1.3 Justification of Linear Heat Generation Rate Penalties

The effect of rod bowing (gap closure) on the augmentation of the linear heat generation rate is determined by considering the neutronics effects related to local changes in fuel-to-moderator ratio (non-fueled area change) in the vicinity of the bowed rods. In Supplement 4 (Reference 5), the RAI Question 15 response directly addressed the application of the power peaking augmentation to a wide variety of fuel designs. **[**

] The NuScale HTP™ fuel design characteristics, also shown in Table 6-1 for the pellet diameter, cladding diameter, and pin pitch, result in **[**

]. Since the NuScale fuel design **[**

], it is concluded the power peaking augmentation from Reference 5 is applicable to the NuScale fuel design. Also shown in Table 6-1 is an example of another AREVA 17x17 HTP™ fuel design which is in operation and uses the same fuel rod bowing licensing basis of Reference 5.

Table 6-1 Water-to-Fuel Volume Ratios

Fuel Design	**Pellet OD** (in)	**Clad OD** (in)	**Pin Pitch** (in)	**Volume Ratio H_2O/U**	**References and Notes**
17x17 W	0.303	0.360	0.496	2.00	Supplement 4 (Reference 5)
14x14 W Standard	0.3565	0.426	0.556	1.669	Supplement 4 (Reference 5)
17x17 W (HTP™ fuel design)	0.3215	0.376	0.496	1.663	In operation
NuScale HTP™ (17x17)	0.3195	0.374	0.496	1.70	

6.2 *Topical Report Restrictions*

6.2.1 Technical Evaluation Report

The safety evaluation for the XN-75-32(P)(A) (Reference 5) supplements approves the means for analyzing the effects of fuel rod bowing and determining the power peaking factor and CHF penalties. The SER states that the acceptability is limited to the fuel designs, exposures and conditions covered in the licensing topical report supplements and supporting documentation. The NRC acceptance is not applicable to fuel designs which exhibit a greater propensity for bowing than that given in the data from which the models reviewed by the NRC were developed.

The applicability of the XN-75-32(P)(A) methodology to the HTP™ fuel design has been approved in Reference 9. A comparison of the NuScale fuel design and operating conditions shows that the NuScale fuel design has a relatively high resistance to fuel rod bowing, comparable to the AREVA PWR 17x17 fuel.

Based on the comparison of the NuScale nominal operating conditions to the CHF test conditions that form the basis of the fuel rod bowing CHF penalty, the procedure for the calculation of the CHF penalty for potential rod bowing from Reference 5 is applicable to the NuScale fuel design.

6.3 *Summary and Conclusions for XN-75-32(P)(A) Topical Report*

The XN-75-32(P)(A) Supplements define the methodology for analyzing the effects of fuel rod bowing and determining the power peaking factor (F_Q) and departure from nucleate boiling ratio rod bowing penalties. The fuel rod bowing topical report methods for determining the CHF and LHGR penalties are also applicable for the NuScale design.

7.0 REVIEW OF EMF-92-116(P)(A) GENERIC MECHANICAL DESIGN CRITERIA FOR PWR FUEL DESIGNS TOPICAL REPORT

AREVA topical report EMF-92-116(P)(A) (Reference 6) defines the NRC approved SAFDLs that provide assurance of satisfactory performance for nuclear fuel. The SAFDLs correspond to those of Section 4.2 of the Standard Review Plan (NUREG-0800). EMF-92-116(P)(A) also cites or describes the methodologies to be used to demonstrate acceptable fuel performance.

Only certain sections of EMF-92-116(P)(A) are applicable to the NuScale fuel design. The applicability of Reference 6 is limited to the fuel design, fuel mechanical analysis, and fuel thermal-mechanical analysis. NuScale will not utilize AREVA methodology for neutronics calculations analysis, safety analysis, and a majority of the thermal-hydraulic analyses.

EMF-92-116(P)(A) frequently refers to separately approved methodologies that are used to demonstrate compliance to criteria. In those cases, either the method currently referenced in EMF-92-116(P)(A) will be evaluated separately, as in the case of the rod bowing methodology XN-75-32(P)(A) (see Section 6.0), or a more contemporary code that replaces one of the EMF-92-116(P)(A) referenced methodologies is being evaluated. An example of the latter is the use of COPERNIC rather than RODEX, where COPERNIC replaces the legacy code and was evaluated for applicability to NuScale operating conditions separately (see Section 5).

Table 7-1 lists the sections of EMF-92-116(P)(A) that are being used for the NuScale fuel design.

Table 7-1 Use of EMF-92-116(P)(A) Sections

EMF-92-116(P)(A) Section	Used for NuScale Fuel Design	Comment
3.2.1 Internal Hydriding	Yes	
3.2.2 Cladding Collapse	No	Use alternate method per BAW-10084PA (Reference 1)
3.2.3 Overheating of Cladding	No	NuScale methodology
3.2.4 Overheating of Fuel Pellets	No	NuScale methodology
3.2.5 Fuel Rod Stress and Strain Limits	No	Use alternate method per BAW-10227PA (Reference 2)
3.2.6 Cladding Rupture	No	NuScale methodology
3.2.7 Fuel Rod Mechanical Fracturing	No	Use alternate method per BAW-10133PA (Reference 7)
3.2.8 Fuel Densification and Swelling	No	Use alternate method per BAW-10231PA (Reference 4)
3.3.1 Stress, Strain, or Loading Limits on Assembly Components	Yes	
3.3.2 Fuel Cladding Fatigue**	No	Use alternate method per BAW-10227PA (Reference 2)
3.3.3 Fretting Wear	Yes	
3.3.4 Fuel Cladding Oxidation, Hydriding, and Crud Buildup	No	Use alternate method per BAW-10231PA (Reference 4)
3.3.5 Rod Bow	No	Use alternate method per XN-75-32(P)(A) (Reference 5)
3.3.6 Axial Growth	Yes	
3.3.7 Rod Internal Pressure	No	Use alternate method per BAW-10231PA (Reference 4)
3.3.8 Assembly Liftoff	Yes	
3.3.9 Fuel Assembly Handling	Yes	
3.4 Fuel Coolability	No	NuScale methodology
4.0 Thermal and Hydraulic Design	No	NuScale methodology
5.0 Nuclear Design Analyses	No	NuScale methodology

**Note that guide tube fatigue calculations are performed according to this general method and also in accordance with NUREG-0800 page 4.2-6.

The sections listed below from EMF-92-116(P)(A) describe the methodology that will demonstrate compliance to the SAFDLs for the AREVA items listed in Table 7-1. Each method listed below is evaluated for applicability in Section 7.1.

- Internal hydriding (3.2.1)
- Normal operation stress analysis in the fuel assembly structure – note that rods are evaluated per a separate methodology (3.3.1)
- Fretting wear (3.3.3)
- Axial growth addressing both fuel rod and fuel assembly (3.3.6)
- Fuel lift analysis (3.3.8)
- Shipping and handling stress analysis (3.3.9)

7.1 *Applications*

The SAFDLs presented in EMF-92-116(P)(A) were used to license a number of PWR fuel designs, including W-type 17x17 fuel designs that are similar to the NuScale fuel design. Below is a chapter summary of the topical report. Chapter numbers refer to the Chapter numbers in the EMF-92-116(P)(A) topical report (Reference 6).

Chapter 1 *Introduction*

This chapter provides the background and history of the EMF-92-116(P)(A) topical report, noting that the report contains fuel design criteria which provide assurance for adequacy of the design throughout the design life. Chapter 1 gives no limitations on the use of the topical report for NuScale design fuel rod design.

Chapter 2 *Fuel Assembly Summary and Conclusions*

This chapter summarizes the high level design criteria from 10 CFR 50, Appendix A. In the case of NuScale, for the applicable sections of this report, these criteria are valid.

Chapter 3 *Generic Fuel System Design Criteria*

Chapter 3 provides the individual fuel design criteria. The following paragraphs will address each EMF-92-116(P)(A) criterion that will be applied to the NuScale fuel assembly design as detailed in Table 7.1.

Section 3.2.1 Hydriding

Fuel rod internal hydriding is controlled by fabrication limits for fuel pellet moisture. These controls, typical for AREVA fuel manufacturing, are expected to be maintained for NuScale fuel production. There is no change to the criterion or method as it applies to NuScale fuel, so the EFM-92-116(P)(A) method remains applicable and the mechanical design criteria are unchanged.

Section 3.3.1 Stress, Strain, or Loading Limits on Assembly Components

The structural integrity of the fuel assembly components is calculated under normal operating conditions and handling loads. Fuel rods are addressed by a separate methodology topical report (see Reference 2). Appendix III of the ASME Code (Reference 16) cites the acceptable levels for fuel assembly components. The methods of calculating component stresses use open-literature equations or finite elements methods (such as ANSYS) with no specific range of applicability that relates to the NuScale design.

The NuScale normal operating conditions, listed in Table 2-2, create the same fuel assembly component stress conditions (with lower magnitudes) as a typical PWR. Therefore the stresses may be evaluated to the EMF-92-116(P)(A) mechanical design criteria and the cited methods are applicable to the NuScale design.

Section 3.3.3 Fretting Wear

Fretting wear is predicted by testing prototypical components in applicable flow conditions that represent the reactor coolant system conditions. A 1000-hour fretting test specific to the NuScale fuel design and operating conditions has been performed. Applicability of the testing has been assured by the use of bounding test conditions and use of a prototypic test bundle.

The mechanical design criterion is therefore unchanged and the method of demonstrating compliance is applicable to the NuScale design.

Section 3.3.6 Axial Growth Addressing Both Fuel Rod and Fuel Assembly

Empirical models are used for AREVA fuel rod and fuel assembly growth analyses. The growth analyses consider the dimensional constraints specific to the NuScale fuel design – the core plate to core plate gap constrains the fuel assembly growth and the fuel assembly nozzle to nozzle gap constrains the fuel rod growth. The evaluation also accounts for tolerances specific to the NuScale components. The predicted growth rates are based on fluence values specific to the anticipated NuScale core designs and are within the fluence values that establish the range of applicability for the AREVA growth correlations. The fuel rod growth prediction accounts for the axial growth of M5® cladding material. For the NuScale fuel assembly growth prediction, a growth multiplier will be applied to conservatively account for the impact of having no hold-down spring on total fuel assembly growth.

The EMF-92-116 TER (paragraph two of section 2.6), notes that the fuel assembly (guide tube) axial growth model does not require resubmittal to NRC as long as the upper or lower bound of the base growth model does not change by more than a standard deviation. In the NuScale fuel assembly growth analysis, the predicted Zr-4 irradiation growth is increased by a factor of **[ ]**^{ECI} to account for the absence of a hold-down spring. The **[ ]**^{ECI} factor was derived by comparing data from similar PWR fuel designs but with the key fuel design difference being one set of growth measurements was from a group of fuel assemblies with no hold-down spring system while the other fuel assemblies all had hold-down springs. Applying this factor supports the extension of the EMF-92-116(P)(A) methodology to the NuScale design. See Appendix 1 of this report for additional information concerning the calculation of the growth factor.

Section 3.3.8 Fuel lift analysis

The fuel lift analysis accounts for the fuel assembly mass, hold-down spring loads, and hydraulic forces. The hold-down spring loads are not applicable to the NuScale fuel design. The mass values and predicted hydraulic forces are specific to the NuScale design. The flow rate (approximately **{{ }}**$^{2(a),(c), ECI}$) is within the values used in typical AREVA PWR fuel lift

analyses. The standard analysis method therefore applies and the mechanical design criterion is unchanged.

Section 3.3.9 Fuel Assembly Handling

Methods described in Section 3.3.1 are used to determine stresses from shipping and handling of fuel assemblies. The shipping and handling loading conditions for NuScale fuel are expected to be less limiting than those for the PWR fuel previously analyzed by AREVA with these methods due to the lower NuScale fuel assembly mass. The shipping container is expected to be similar to current AREVA PWR designs and is expected to use the same types of constraints for lateral and axial accelerations. Therefore, the standard method applies and the EMF-92-116(P)(A) mechanical design criterion is unchanged for the NuScale design.

Chapter 4 *Thermal and Hydraulic Design Criteria*

Chapter 4 is not being referenced.

Chapter 5 Nuclear Design Analysis

Chapter 5 is not being referenced.

Chapter 6 Testing, Inspection, and Surveillance

Chapter 6 is not applicable to a discussion of methods. Testing and inspection requirements are specified as part of the fuel design definition. Quality control programs and surveillance programs are beyond the scope of this applicability report.

Chapter 7 Sample Calculation Results

The sample calculations are not applicable to NuScale fuel design.

Chapter 8 References

The references are not relevant for the purposes of this applicability report.

Appendix A Fuel Assembly Component Drawings

Appendix A drawings are not applicable to the NuScale fuel design.

7.2 *Topical Report Restrictions*

7.2.1 Safety evaluation report

The NRC's SER for EMF-92-116(P)(A) approves the usage of EMF-92-116(P)(A) for PWR licensing applications up to **[ ]**ECI rod average burnup. This limit is applicable to the NuScale fuel design. There are no applicable restrictions in the SER. Specific technical limitations are addressed in the associated TER.

Sections 2.1, 2.4, 2.6, 2.8, 2.9, and 2.10 of the TER were reviewed as they correspond to the sections of EMF-92-116(P)(A) being applied to the NuScale design. There are no technical restrictions in any of these paragraphs that would limit the application of the EMF-92-116(P)(A) generic mechanical design criteria to the NuScale design.

7.2.2 Technical Evaluation Report

In the EMF-92-116(P)(A) TER (Reference 6), a note was added in response to the AREVA information that states: "…SPC should pay particular attention to the vibrational characteristics of new fuel assembly designs over their entire flow operating range during their out-of-reactor flow tests to avoid vibrational fretting wear during operation." This observation is not a specific restriction, but it has been reflected in the AREVA life and wear test procedure. The life and wear testing performed on the NuScale fuel design utilized a greater than expected operational flow rate that created a bounding flow condition for testing.

7.3 *Summary and Conclusions for EMF-92-116(P)(A) Topical Report*

AREVA topical report EMF-92-116(P)(A) documents the criteria and methods being applied to specific mechanical analyses that are part of the scope of AREVA methodology for NuScale fuel development. The specified methodologies of EMF-92-116(P)(A) are applicable to the NuScale design.

8.0 REVIEW OF BAW-10133PA SEISMIC TOPICAL REPORT

BAW-10133PA-01, including Addenda 1 and 2 (Reference 7), defines a generic methodology for evaluating 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 Section 4.2, Appendix A of the Standard Review Plan (NUREG-0800).

This section addresses the applicability of BAW-10133PA-01 (Reference 7) to NuScale fuel. Generic concerns raised by the U.S. NRC in Information Notice 2012-09 and other generic advancements made in the BAW-10133PA-01 methodology are addressed in Appendix 2.

8.1 *Applications*

BAW-10133PA-01, including Addenda 1 and 2 (Reference 7), defines a generic methodology to evaluate the structural response of fuel assemblies to externally applied forces. The evaluation methodology starts with the execution of standard, 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. In this way, the described methods are transparent to the design being analyzed and may be used to represent the structural response of any PWR fuel assembly. Plant-specificity for various applications is introduced in the geometry of the model boundary conditions (core dimensions, etc.) and inputs (e.g. core plate motion time histories). The methodology defined in BAW-10133PA-01 and Addenda 1 and 2 was used to evaluate a varied portfolio of PWR fuel types (multiple B&W 15x15 and 17x17, Westinghouse 15x15 and 17x17, and Combustion Engineering 14x14 and 16x16 fuel designs).

The BAW-10133PA-01 topical report, although referencing the Mark-C fuel design in the title, describes the generic methodology applicable to PWR fuel designs. This generic applicability is captured in the NRC's SER for BAW-10133PA, Revision 1 in which it is noted that the methodology is acceptable for the "Mark-C fuel design and similar designs." This methodology was modified in Addendum 1 where it is stated that "the application of this method is for generic use." Addendum 2 introduces the damping values to apply in this analysis and is noted to be "justified for all FCF (AREVA) PWR fuel designs based on the supporting test data." While the sample problem for BAW-10133PA, Revision 1 was executed using the Mark-C fuel design for B&W reactors, Addenda 1 and 2 included a sample problem to demonstrate application to the 17x17 fuel design for Westinghouse reactors. This sample problem demonstrates the generic applicability of the method.

A summary of the content of BAW-10133PA-01 and Addenda 1 and 2 (Reference 7) is provided below.

Chapter 1 *Introduction*

This chapter provides an overview of BAW-10133PA-01 and establishes the expectations that the report will describe "the methods and models used in the B&W 17 by 17, Mark-C fuel assembly (FA) and other FA LOCA and seismic dynamic analyses."

Chapter 1 gives no limitations on the use of BAW-10133PA-01 for the NuScale fuel design.

Chapter 2 *Fuel Assembly Physical Description*

This chapter provides a description of the fuel assembly discussed within BAW-10133PA-01 with the intent to "assist in the visualization of the dynamic modeling of the FA." This chapter recognizes that other FA designs are similar in concept. The description of the fuel provided in this chapter may be extended to the NuScale fuel design (similar to other AREVA 17x17 fuel designs), with the following minor exceptions that have no impact on the defined method:

1) Spacer grids are welded directly to the guide tubes and are not captured by a combination of frictional grips and spacer sleeves.

2) The outer straps of the upper and lower end grid are not extended to be mechanically attached to their respective nozzles.
3) The guide tubes are rigidly connected to the top nozzle through a quick-disconnect mechanism, as opposed to a threaded connection.
4) The instrumentation tube is retained at the lower end grid, as opposed to being retained at the bottom nozzle.
5) The NuScale fuel rod does not use tubular spacers to axially locate the fuel stack.

Chapter 2 does not define any restrictions in fuel assembly design pertaining to the two most notable differences in the NuScale design, length and number of spacer grids, although the impact of these changes is recognized and addressed in Section 8.3 of this report.

Chapter 3 Acceptance Criteria

Chapter 3 defines the acceptance criteria to be applied to the fuel for Safe Shutdown Earthquakes (SSE) and Loss of Coolant Accident (LOCA) or combined SSE and LOCA events. The criteria defined in this chapter are generic and remain applicable to the NuScale design.

Chapter 4 Seismic and LOCA Analysis Model Description

Chapter 4 provides a description of the generic modeling approach that may be 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. Two distinctions may be made from this chapter in applying the methodology to the NuScale design:

1) This chapter defines a core support motion that is provided by a reactor coolant system analysis described in BAW-10131PA (Reference 14). In the application to the NuScale fuel, Reference 14 is replaced by a NuScale specifc analysis to define the basis for the core support motions.

2) Chapter 4 defines two vertical models. A single assembly vertical model is used to evaluate the fuel assembly response to core support motions during a seismic event. In addition, a core bounce model is used to evaluate the system response of both core internals and fuel assemblies to oscillatory hydraulic loads resulting from a LOCA event. For the NuScale design, the LOCA event is not as severe of an event as a standard PWR and there is no significant hydraulic loading to be considered. Therefore, the single assembly vertical model, in lieu of the core bounce model, is to be used to evaluate the NuScale fuel assembly response to structural motions resulting from a LOCA event.

Note that elements of the horizontal model and analysis method are modified in Addenda 1 and 2 to BAW-10133PA-01 as described below.

With the exceptions noted above, Chapter 4 of BAW-10133PA-01 is applicable to the NuScale design.

Chapter 5 Seismic and LOCA Analysis Methods

Chapter 5 defines the process of applying appropriate forcing functions representing seismic or LOCA events to the models described in Chapter 4. Chapter 5 also defines the method of combining the fuel assembly response from the horizontal and vertical models as well as accounting for the combined effect of an SSE and LOCA. 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.

With respect to the LOCA analysis, Chapter 5 describes the pipe breaks that are to be considered in defining the LOCA forcing functions and BAW-10131 PA-01 is referenced for a description of these functions. In place of BAW-10131 PA-01, the NuScale design is analyzed for LOCA events specific to that plant. As noted above, the lack of severity associated with a LOCA event in the NuScale design leads to the adoption of the single assembly vertical model in place of the core bounce model. Chapter 5 notes that the single assembly vertical model is

appropriate for seismic conditions since the fuel assembly excitation is caused only by the motion of the upper and lower core plates. Given that there are no significant hydraulic loads acting on the NuScale fuel during a LOCA, the single assembly vertical model is used to analyze LOCA events for the NuScale design using structural motions as inputs. This approach is addressed in Section 8.3.

Note that elements of the horizontal model and analysis method are modified in Addenda 1 and 2 to BAW-10133PA-01 as described below.

With the exceptions noted above, Chapter 5 is applicable to the NuScale fuel design.

Chapter 6 Fuel Assembly Structural Analysis

Chapter 6 defines the process of performing the structural component stress analysis and the evaluation of spacer grid impact loads using the loads and deflections generated by the SSE and LOCA analyses described in Chapter 5. The process defined in this chapter is generic and remains applicable to the NuScale design.

Chapter 7 Conclusion

Chapter 7 provides a brief summary of the report and does not introduce any new content or restrictions. Therefore, Chapter 7 of BAW-10133PA-01 is applicable to the NuScale design.

Appendix A Fuel Assembly Test Program

Appendix A provides a description of the test program used to obtain fuel assembly data in support of the Mark-C fuel evaluation.

Appendix A of BAW-10133PA-01 is applicable to the NuScale design.

Note that elements of the testing to support the horizontal model development are modified in Addendum 1 to BAW-10133PA-01.

Appendix B STARS Description

Appendix B provides a description of the structural code, STARS, used to carry out the modeling and analysis defined in Chapters 4 and 5. STARS is replaced by a new code,

CASAC, in Addendum 1 to BAW-10133PA-01. CASAC is applied in the analysis of the NuScale design fuel. Therefore, Appendix B is not applicable to the NuScale fuel design.

Appendix C Mark C FA LOCA-Seismic Analysis Results

Appendix C provides the results of the sample problem that accompanies BAW-10133PA-01. The sample problem is executed for the Mark-C fuel design for a B&W “205” plant design. This appendix provides insight on the application of the method to a specific fuel design and reactor. This appendix is not relevant to the NuScale fuel design.

Appendix D References

Appendix D lists the references cited throughout BAW-10133PA-01. This appendix is applicable to the NuScale fuel design.

Addendum 1

Addendum 1 introduces two modifications to the approved methods in BAW-10133PA-01. The first modification is a new method for modeling the stiffness of assembly grid locations that accounts for fuel rod to grid interaction. The second modification is an alternate method of treating the fluid effects on fuel response that adds hydrodynamic coupling between core baffle plates and assemblies. The structural code CASAC is introduced in Addendum 1 as a replacement for STARS. This addendum also describes the mechanical tests used to define the characteristics of the models described in this Addendum. The test program defined in Addendum 1 modifies the test program defined in Appendix A of BAW-10133PA-01.

Section 2.2.1 defines a testing program which provides the first six frequencies and mode shapes of the fuel assembly. 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 8.3 of this report.

Section 2.2.2.4 defines fuel assembly damping values corresponding to minimum operational design flow rates for a typical PWR. These damping values are modified in Addendum 2. However, the reduced flow rate and the shorter fuel assembly length necessitates the establishment of a specific damping value for the NuScale design fuel design. This aspect is addressed in Section 8.3.

It is noted in the introduction to this Addendum that "the application of this method is for generic use (all AREVA fuel assembly designs)." With the exceptions noted above, Addendum 1 is applicable to the NuScale fuel design.

Addendum 2

Addendum 2 defines fuel assembly damping values that are higher than previously approved in BAW-10133PA-01 and Addendum 1. These damping values credit operational flow rates for PWRs. The reduced flow rate and the shorter fuel assembly length necessitate the establishment of a specific damping value for the NuScale fuel design, as addressed in Section 8.3.

Addendum 2 is not applicable to the NuScale design.

8.2 *Topical Report Restrictions*

8.2.1 Safety Evaluation Report

The NRC's SER for BAW-10133PA-01 approves the usage of BAW-10133 for "the current Mark-C design and similar designs." Within the SER, there are no restrictions that would preclude the use of this methodology for the NuScale fuel design.

The SER for Addenda 1 and 2 of BAW-10133PA-01 concludes that both addenda are acceptable for incorporation into the overall methodology described in BAW-10133PA-01. Within the SER, there are no restrictions that would preclude the use of this methodology for the NuScale fuel design.

8.2.2 Requests for Additional Information (RAI) and Response

RAI question number 33 from the review of BAW-10133PA-01 states: "The analysis presented in Appendix C of the topical report is adequate to demonstrate the methodology. However, for future plant analysis, sensitivity calculations will be required." This request is met by performing the sensitivity study consistent with that described in NUREG-0800, Chapter 4.2, Appendix A, Section II.3.

8.3 *NuScale Design Differences and Requirements*

To extend the applicability of BAW-10133PA-01, including Addenda 1 and 2, to the NuScale fuel, the following adjustments are made to the methodology.

8.3.1 Frequency Response of the NuScale Design Fuel

Addendum 1 of BAW-10133PA-01 establishes an expectation that the dynamic characterization testing provides the first six frequencies and mode shapes of the fuel assembly. Similarly, the fuel assembly model is able to match these measured characteristics. The NuScale fuel assembly, however, has an increased lateral stiffness and increased lateral frequencies when compared to a longer PWR fuel assembly. As a consequence, the high mode frequencies of the NuScale fuel assembly are beyond the range of interest for the dynamic events that are analyzed. From the dynamic characterization tests on NuScale fuel, only the first three fuel assembly frequencies were definitively identified.

Following the modeling approach in Addendum 1 of BAW-10133PA-01, the NuScale fuel assembly model will be represented as a single beam with 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. This modeling approach's results are consistent with the frequencies measured from the dynamic characterization tests and reflects adequate mass participation in the dynamic response. Therefore, this modeling approach is applicable to the NuScale design.

8.3.2 Fuel Assembly Damping

Addendum 2 to BAW-10133PA-01 defines fuel assembly damping values that are generically applicable to standard PWR fuel designs. However, relative to a standard PWR, the NuScale design operates with a shorter fuel assembly and reduced flow rates. For these reasons, the basis for the damping values defined in Addendum 2 is not applicable to the NuScale design.

For the NuScale design, the maximum fuel assembly damping ratio values to be used for the analysis of seismic and LOCA events are defined in Table 8-1. Appendix 3 provides additional details regarding the establishment of these values.

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The damping ratio values for the NuScale design reflect the combined presence of structural damping as well as viscous damping in still-water conditions. Conservatively, the damping ratio values in Table 8-1 do not reflect the additional contribution of damping in flowing water. The structural damping component for NuScale is based on direct experimental data for a prototypical NuScale fuel assembly and is the dominant component for the values presented in Table 8-1. The still-water, viscous damping component is defined using the same experimental database presented in Addendum 2 of BAW-10133PA-01. This experiment was performed with a fuel assembly that has similar array and cross-sectional properties as the NuScale fuel, but with a longer length. [

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This method, using the damping values specified in Table 8-1, is applicable to the NuScale design.

8.3.3 Vertical LOCA Model

BAW-10133PA-01 defines a single assembly vertical model for the analysis of seismic events where the excitation of the fuel is caused by the motion of the upper and lower core plates. This approach remains applicable for the NuScale fuel. For the simulation of LOCA events, BAW-10133PA-01 defines a core bounce model that looks at the overall system response of both core internals and fuel assemblies to oscillatory hydraulic loads resulting from the LOCA event. In the case of the NuScale design, the LOCA event does not have a significant oscillatory hydraulic load component. In this case, the core bounce model simplifies to a single assembly model that is only excited by the motion of the upper and lower core plates.

For the analysis of NuScale fuel, a single assembly vertical model with motions applied at the upper and lower core plates is used to analyze the vertical response of the fuel to LOCA events.

8.4 *Summary and Conclusions for BAW-10133PA Topical Report*

BAW-10133PA-01 and Addenda 1 define a generic methodology for performing the evaluation of fuel assembly structural response to externally applied forces (i.e., SSE and LOCA) for PWRs. This methodology is applicable to the NuScale design with two modifications: 1) The use of fuel assembly damping values specific to the NuScale design, and 2) The use of a single assembly model for LOCA evaluations in place of a core bounce model. Both of these modifications are defined and justified within this report.

9.0 OVERALL SUMMARY AND CONCLUSIONS

This report documents the applicability of NRC-approved AREVA codes and methods for evaluating performance of the NuScale fuel design in the design certification application. For application of the six AREVA topical reports to NuScale, the following three minor modifications are required to conservatively predict NuScale fuel behavior:

1. Paragraph 3.3.6 of EMF-92-116(P)(A) addresses the fuel assembly growth methodology. Since the NuScale fuel assembly does not contain a hold-down spring, an axial fuel assembly growth multiplier was developed to conservatively model this impact as discussed in Section 7 of this report. Since this is a change with respect to the effective upper bound of the base growth model, this method modification requires NRC approval.
2. Two adjustments to the BAW-10133PA-01 methodology are required. Fuel assembly damping values specific to the NuScale design were derived. Also, a single assembly model for LOCA evaluations is used in place of a core bounce model. Both of these modifications are discussed in Section 8 and require NRC approval.
3. One adjustment to the BAW-10084PA methodology is required. The creep collapse analysis should not be performed at an **{{**

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These modifications are applied to the NuScale analyses only. With the NRC approval of these modifications, the six AREVA codes and methods cited in this report are applicable to the NuScale fuel design and the analysis results are suitable for the NuScale design certification application.

10.0 REFERENCES

1. BAW-10084PA-03 (BAW-10084PA Rev. 3), Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse, August 1995.

2. BAW-10227PA-01 (BAW-10227PA, Rev. 1), Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, June, 2003.

3. BAW-10183PA-00 (BAW-10183PA, Rev. 00), Fuel Rod Gas Pressure Criterion (FRGPC), February, 1996.

4. BAW-10231PA-01 (BAW-10231PA, Revision 01) COPERNIC Fuel Rod Design Computer Code, January 2004.

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

6. EMF-92-116(P)(A), Generic Mechanical Design Criteria for PWR Fuel Designs.

7. BAW-10133PA-01, Revision 1, and Addenda 1 and 2, Mark-C FA LOCA-Seismic Analyses.

8. NUREG-0800 USNRC Standard Review Plan, Section 4.2, Revision 3 – March 2007.

9. ANF-89-060(P)(A), Supplement 1, Generic Mechanical Design Report High Thermal Performance Spacer and Intermediate Flow Mixer, February 1991.

10. Y. Nagino, et al., “Rod Bowed to Contact Departure from Nucleate Boiling Tests in Coldwall Thimble Cell Geometry”, Journal of Nuclear Science and Technology, 15(8), pp. 568-573, August 1978.

11. Wesley, D. A. and M. Yovanovich, “A New Gaseous Gap Conductance Relationship,” Nuclear Technology, Vol. 72, January 1986, pages 70-74.

12. B.B. Mikic, “Thermal Contact Conductance: Theoretical Considerations,” Int. J. Heat Mass Transfer, Vol. 17, 1973.

13. G. Jacobs et N. Todreas, “Thermal Contact Conductance in Reactor Fuel Elements,” Nuclear Science and Engineering, Vol. 50, 1973.

14. BAW-10131PA, Revision 0, Reactor Coolant System Structural Loading Analysis, November 1977.

15. J. E. Garnier, er S. Begel, “Ex-reactor Determination of Thermal Gap and Contact Conductance Between Uranium Dioxide: Zircaloy-4 Interfaces, Stage I: Low Gas Pressure,” NUREG/CR-0300, 1979, Vol. 2.

16. ASME Boiler and Pressure Vessel Code: Section III, 2015.

APPENDIX 1 - FUEL ASSEMBLY IRRADIATION GROWTH FACTOR

Section 7.1 cited that a **[ ]** multiplier was applied to the fuel assembly axial growth prediction to account for the NuScale fuel assembly being designed with no hold-down spring system. This appendix provides the basis for determining the growth multiplier.

AREVA's W-type fuel 17x17 fuel maximum assembly growth model for Zr-4 material is used for the NuScale 17x17 design. Since this growth model is based on the measured growth of fuel assemblies with hold-down springs, it is adjusted to account for the absence of hold-down springs on the NuScale fuel design. The absence of a hold-down spring system contributes to higher guide tube growth by removing the primary source of the axial, compressive loads that act on the guide tubes during normal operation.

AREVA derived a growth multiplier by comparing the measured growth rates of similar CE-type fuel designs, one having a hold-down spring system and one without. The data set for the CE-type14x14 fuel with a hold-down spring system is shown in Figure A1-1 below.

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APPENDIX 2 – GENERIC AUGMENTATIONS TO THE BAW-10133PA-01 METHODOLOGY

1.0 INTRODUCTION

Section 8 of this document addresses the applicability of BAW-10133PA-01 (Reference 1) to NuScale fuel. That section does not, however, address the generic concerns raised by the NRC in Information Notice 2012-09 (Reference 2), nor does that section discuss the generic advancements to be made in the BAW-10133PA-01 methodology. The purpose of this appendix is to present generic augmentations to the methodology in BAW-10133PA-01 that will be applied to NuScale fuel.

2.0 GENERIC AUGMENTATIONS TO THE BAW-10133PA-01 METHODOLOGY

2.1 Method for Evaluating Fuel in the Irradiated Condition

NRC Information Notice 2012-09, "Irradiation Effects on Spacer Grid Crush Strength," identifies a concern about the impact of the change in behavior of the assembly and assembly components during the operational lifetime. Additional testing and evaluations were included in the NuScale analyses to address this information notice. In particular, characterization tests (free and forced vibration) were performed for two full-scale NuScale test assemblies, one representing the non-irradiated, or beginning-of-life (BOL), condition and the other representing the irradiated, or end-of-life (EOL), condition. Similarly, characterization testing of individual spacer grids was performed on grids in both the non-irradiated and a simulated-irradiated condition. These test data are used to develop separate models that represent the fuel in both non-irradiated and irradiated conditions. These models were applied in the same manner to consider the fuel response in both the non-irradiated and irradiated conditions.

To simulate the effects of irradiation on overall fuel assembly characteristics during dynamic characterization testing, **[**

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The effects of irradiation were simulated in the spacer grid impact testing. The general methodology for the spacer grid dynamic impact testing is established and defined in Addendum 1 of BAW-10133PA-01. The spacer grid dynamic impact testing was performed for both non-irradiated and simulated-irradiated configurations using this testing protocol, but **[**

]ECI It was demonstrated, based on testing

conducted in a hot cell on actual irradiated grids, that **[**

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]ECI This protocol was shown to demonstrate a good, conservative agreement with the results of tests performed on irradiated grids in a hot cell.

2.2 Definition of spacer grid allowable impact in irradiated and non-irradiated condition

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2.3 Updated method of calculating non-grid component stresses

The determination of the stresses in guide tubes and other non-grid components is also modified **[**

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2.4 Clarification of numerical model for vertical

The single assembly vertical model uses a more advanced and detailed architecture compared to that defined in BAW-10133PA-01 in order to allow for the evaluation of specific component loads.

The single assembly model uses three columns representing the guide tubes, fuel rods, and the instrument tube. These columns are modeled as linear translational springs with all nodal degrees of freedom (DOFs) in the lateral and rotational directions constrained. The spring element characteristics are directly derived from geometric and material considerations except for the nozzles whose stiffnesses are derived from design-specific component testing. In addition, the nonlinear capabilities of the model consist of a number of gap-springs, gap dampers, and slider elements. A brief discussion of these elements follows.

- The non-linear gap-spring-damper element representing the lower nozzle: The stiffness of this element is used to account for the stiffness of the load-path through the lower nozzle. The input parameters for this element are benchmarked using results from an axial drop test performed on the NuScale fuel assembly design.
- The non-linear gap-spring-damper element between the fuel rod lower end and the top face of the lower nozzle: In the BOL condition, this gap is open for the NuScale design.

In the EOL condition, the gap is closed due to the subsequent seating of the rods on the upper face of the lower nozzle. The input parameters for this element are benchmarked using results from an axial drop test performed on the NuScale fuel assembly design.

- The non-linear gap-spring-damper element between the top of the fuel rods and the upper nozzle: The characteristics of this spring are similar to the element above. This gap is designed to remain open over the service life of the fuel assembly under normal operating conditions (including irradiation growth), but could close due to impacts produced by seismic or LOCA loading. The input parameters for this element are benchmarked using results from an axial drop test performed on the NuScale fuel assembly design.
- The non-linear slider elements between the fuel rod nodes and the corresponding spacer grid nodes on the guide tube column: These elements are characterized by a stiffness and a saturation force at which the fuel rods begin to slip within the spacer grids. The input parameters for this element are benchmarked using the slip loads measured from tests performed on NuScale HTP™ spacer grids and test results from axial stiffness tests performed on the NuScale fuel assembly.

3.0 CONCLUSIONS / SUMMARY

As described in this appendix, the methodology in BAW-10133PA-01 was augmented with the following enhancements:

- Definition of methodology for evaluating fuel in the irradiated condition (NRC Information Notice 2012-09)
- Definition of the spacer grid allowable impact load in the irradiated and non-irradiated condition
- Definition of an updated method of calculating non-grid component stresses
- Definition of an updated acceptance criteria for guide tubes for SSE and SSE plus LOCA conditions
- Definition of an updated numerical model for vertical load analysis

BAW-10133PA-01, as amended by both Section 8 of this report and by this Appendix, has been applied for the analysis and licensing of NuScale fuel.

4.0 REFERENCES

1. BAW-10133PA, Revision 1, and Addenda 1 and 2, Mark-C FA LOCA-Seismic Analyses.
2. U.S. Nuclear Regulatory Commission Information Notice 2012-09: Irradiation effects on fuel assembly spacer grid crush strength, June 28, 2012.
3. 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.

APPENDIX 3 – NUSCALE DAMPING FOR LATERAL ACCIDENT CONDITION ANALYSIS

1.0 INTRODUCTION

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

1. In-air, structural damping: This loss of energy is related mostly to the energy dissipation at the contact interface between fuel assembly components, including the boundary conditions, and can be measured in air.

2. Quiescent, viscous water damping: This loss is due to irrecoverable pressure losses during the movement of the fuel assembly through water.

3. 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 defines these damping ratio values for the NuScale fuel design in both the non-irradiated and irradiated conditions.

2.0 METHODOLOGY

The different sources of damping may be independently measured by performing free vibration and forced vibration tests on fuel assemblies in-air and in-water conditions. For the NuScale design, these sources of damping are based on in-air tests performed on NuScale fuel assemblies and in-water tests performed on geometrically similar fuel assemblies. 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:

- NuScale tests: The NuScale tests involve two NuScale 17x17 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.

- MASSE tests: These tests were performed using **[**

  ]ECI 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.

- CAMEOL tests: These tests were performed using **[**

]ECI 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.

- MALDIVE tests: These tests were performed using [ ]ECI Both free vibration and forced vibration tests were performed in this campaign in both air and water, including the effects of flowing water.

- MARITIME tests: These tests were performed using [ ].ECI 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. [ ]ECI

- Quiescent, viscous water damping (drag): [ ]ECI

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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 shown to under-represent the damping ratio of the fuel by more than 10 percent of the overall damping ratio. [

]ECI

It is shown that the damping ratio for the NuScale fuel assembly is dependent on the amplitude at which the fuel assembly is oscillating. The methodology employed here results in the valuation of damping at [

]ECI

[

]ECI

3.0 RESULTS

3.1 Structural Damping Ratio

The structural damping ratio measured from experiment for the NuScale fuel assembly is shown in Figure A3-1 (non-irradiated) and Figure A3-2 (simulated-irradiated as described in Appendix 2).
{{

}}2(a),(c),ECI

Figure A3-1: Experimental Damping Ratio Values for Mode 1 for the Non-irradiated Nuscale Fuel Assembly (In-air, Free Vibration Tests)

{{

}}2(a),(c),ECI

Figure A3-2: Experimental Damping Ratio Values for Mode 1 for the Simulated-irradiated NuScale Fuel Assembly (In-air, Free Vibration Tests)

{{

}}2(a),(c),ECI

{{

}}2(a),(c),ECI

{{

}}2(a),(c),ECI

{{

}}2(a),(c),ECI

{{

}}2(a),(c),ECI

3.2 Quiescient, Viscous Water Damping (Zero Flow)

[

]ECI

{{

}}2(a),(c),ECI

{{

}}2(a),(c),ECI

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}}2(a),(c),ECI

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}}2(a),(c),ECI

3.2.1 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 demonstrated through numerous experimental campaigns was quantified for NuScale. Using the MASSE, CAMEOL, and MALDIVE test data, and the same method employed in Section 8.3.2 of this report, a damping ratio was quantified for the NuScale flow conditions. This component is shown to **[**

].ECI However, for conservatism, the effects of coolant flow are not credited in the total damping ratio for the NuScale fuel assembly.

4.0 SUMMARY OF FUEL ASSEMBLY DAMPING RATIOS FOR NUSCALE

A summary of the damping ratio values for modes 1 and 3 for the NuScale fuel assembly is given in Table A3-1. **[**

]ECI

Table A3-1: Summary of NuScale Fuel Assembly Damping Ratios at 0.39 inch

{{

}}2(a),(c),ECI

{{

}}2(a),(c),ECI

Enclosure 3:

AREVA Affidavit

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

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

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

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

3. I am familiar with the AREVA information contained in the following document: "TR-0116-20825-P, Applicability of AREVA Fuel Methodology for the NuScale Design," referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

Martha E. Hoth
____

SUBSCRIBED before me this 11th
day of March, 2016.

____
Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/18
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2018

Enclosure 4:

NuScale Affidavit, AF-0316-48349

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

NuScale Power, LLC

AFFIDAVIT of Thomas A. Bergman

I, Thomas A. Bergman, state as follows:

(1) I am the Vice President of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the process and method by which NuScale develops its Fuel Methodology.

NuScale has performed significant research and evaluation to develop a basis for this process and method and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed report titled "Applicability of AREVA Fuel Methods for NuScale SMR." The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies

upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 30, 2016

Thomas A. Bergman