February 9, 2018 Docket: PROJ0769

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of the Approved Version of NuScale Topical Report TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1 (NRC Project No. 0769)

REFERENCE: Letter from Frank Akstulewicz (NRC) to Thomas Bergman (NuScale), "Final Safety Evaluation for NuScale Power, LLC Topical Report-0116-20825, Revision 1, 'Applicability of AREVA Fuel Methodology for the NuScale Design,' dated November 24, 2017 (ML17298A692).

By the referenced letter dated November 24, 2017, the NRC issued a final safety evaluation report documenting the NRC staff conclusion that NuScale Topical Report TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1, is acceptable for referencing in licensing applications for the NuScale design. The referenced letter requested that NuScale publish the approved version of TR-0116-20825, Revision 1.

Accordingly, Enclosure 1 to this letter provides the approved version of the topical report, designated TR-0116-20825-P-A, Revision 1. The approved version includes the November 24, 2017 NRC letter and its final safety evaluation report, the NuScale response to NRC requests for additional information, and the final topical report submittal (Revision 1).

Enclosure 1 contains proprietary information. NuScale requests that the proprietary enclosure be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavits (Enclosure 3 and Enclosure 4) support this request. Enclosure 3 pertains to the NuScale proprietary information, denoted by double braces (i.e., "{{ }}"). Enclosure 4 pertains to the Framatome Inc. (formerly AREVA Inc.) proprietary information, denoted by 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 approved topical report package.

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

Please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com if you have any questions.

Sincerely,

Thomas A. Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC

Distribution: Frank Akstulewicz, NRC, OWFN-8H4A
Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Topical Report TR-0116-20825-P-A, “Applicability of AREVA Fuel Methodology for the NuScale Design”, Revision 1, proprietary version

Enclosure 2: NuScale Topical Report TR-0116-20825-NP-A, “Applicability of AREVA Fuel Methodology for the NuScale Design”, Revision 1, nonproprietary version

Enclosure 3: Affidavit of Thomas A. Bergman, AF-0118-58146

Enclosure 4: Affidavit of Nathan E. Hottle

Enclosure 1:

NuScale Topical Report TR-0116-20825-P-A, “Applicability of AREVA Fuel Methodology for the NuScale Design”, Revision 1, proprietary version

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

Enclosure 2:

NuScale Topical Report TR-0116-20825-NP-A, “Applicability of AREVA Fuel Methodology for the NuScale Design”, Revision 1, nonproprietary version

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

Contents

Section A

November 24, 2017

Mr. Thomas Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC
1100 NE Circle Boulevard, Suite 200
Corvallis, OR 97330

SUBJECT: FINAL SAFETY EVALUATION FOR NUSCALE POWER, LLC TOPICAL REPORT-0116-20825, REVISION 1, “APPLICABILITY OF AREVA FUEL METHODOLOGY FOR THE NUSCALE DESIGN”

Dear Mr. Bergman:

By letter dated March 30, 2016, NuScale Power, LLC (NuScale) submitted Topical Report (TR)-0116-20825, Revision 0, “Applicability of AREVA Fuel Methodology for the NuScale Design” (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16095A244) to the staff of the U.S. Nuclear Regulatory Commission (NRC). NuScale requested the NRC staff to review and approve the assumptions, codes, and methodologies presented in TR-0116-20825 for applying AREVA codes and fuel methodology to the NuScale design. By letter dated June 3, 2016, NuScale requested the NRC to suspend its acceptance review (ADAMS Accession No. ML16155A449). The purpose of this suspension was for NuScale to incorporate comments received from the NRC staff. By letter dated July 1, 2016, NuScale submitted TR-0116-20825, Revision 1, “Applicability of AREVA Fuel Methodology for the NuScale Design” (ADAMS Accession No. ML16187A016) and requested the NRC to review and approve the assumptions, codes, and methodologies presented TR-0116-20825, Revision 1, for applying AREVA codes and fuel methodology to the NuScale design.

The NRC staff has found that the TR-0116-20825, Revision 1, “Applicability of AREVA Fuel Methodology for the NuScale Design,” is acceptable for referencing in licensing applications for the NuScale small modular reactor design to the extent specified and under the conditions and limitations delineated in the enclosed safety evaluation report (SER). The SER defines the basis for acceptance of the TR.

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

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

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

Prior to placing the public version of this document in the publicly available records component of NRC's ADAMS, the staff requests that NuScale perform a final review of the SER for proprietary or security related information not previously identified. If you believe that any additional information meets the criteria, please identify such information line by line and define the basis pursuant to the criteria established in Title 10 of the *Code of Federal Regulations* Part 2, Section 390.

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

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

Sincerely,

/RA Anna Bradford Acting for/

Frank M. Akstulewicz, Director
Division of New Reactor Licensing
Office of New Reactors

Project No. 0769

Enclosures: Enclosure 1 (Non-Proprietary)
Enclosure 2 (Proprietary)

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

SUBJECT: FINAL SAFETY EVALUATION FOR NUSCALE POWER, LLC TOPICAL REPORT-0116-20825, REVISION 1, "APPLICABILITY OF AREVA FUEL METHODOLOGY FOR THE NUSCALE DESIGN" DATED:

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TOPICAL REPORT-0116-20825, REVISION 1, “APPLICABILITY OF AREVA FUEL METHODOLOGY FOR THE NUSCALE DESIGN”

1.0 INTRODUCTION

By letter dated March 30, 2016, NuScale Power, LLC (NuScale) submitted Topical Report (TR)-0116-20825, Revision 0, “Applicability of AREVA Fuel Methodology for the NuScale Design” (Reference 1) to the staff of the U.S. Nuclear Regulatory Commission (NRC). NuScale requested the NRC to review and approve of the assumptions, codes, and methodologies presented in TR-0116-20825 for applying AREVA codes and fuel methodology to the NuScale design.

By letter dated June 3, 2016, NuScale requested the NRC to suspend its acceptance review (Reference 2). The purpose of this suspension was for NuScale to incorporate comments received from the NRC staff. By letter dated July 1, 2016, NuScale submitted TR-0116-20825, Revision 1, “Applicability of AREVA Fuel Methodology for the NuScale Design” (Reference 3) and requested the NRC to review and to approve the assumptions, codes, and methodologies presented TR-0116-20825, Revision 1, for applying AREVA codes and fuel methodology to the NuScale design.

This safety evaluation report (SER) is based on the submitted letter and responses to requests for additional information (RAIs). TR-0116-20825, Revision 1 (Reference 3), is designed to be referenced as part of a Design Certification (DC) licensing approval request. The subject TR provides an applicability analysis of the following AREVA fuel system methodologies and codes for use in NuScale fuel analyses:

1. Babcock and Wilcox (BAW)-10084P-A-03, Revision 3, “Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse”, August 1995 (Reference 4)

2. BAW-10227P-A, Revision 1, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel”, June 2003 (Reference 5)

3. BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code”, January 2004 (Reference 7)

4. XN-75-32(P)(A), Supplements 1-4, “Computational Procedure for Evaluating Fuel Rod Bowing”, February 1983 (Reference 8)

5. EMF-92-116(P)(A), Revision 0, “Generic Mechanical Design Criteria for PWR Fuel Designs”, February 2015 (Reference 9)

This SER is divided into seven sections. Section 1 is the introduction, Section 2 presents a summary of applicable regulatory criteria and guidance, Section 3 contains a summary of the information presented in the TR, and Section 4 contains the technical evaluation of the five major components of TR-0116-20825, Revision 1, as listed above. Section 5 presents the conclusions of this review, Section 6 contains the restrictions and limitations on the use of TR-0116-20825, Revision 1, and Section 7 outlines the utilized references.

2.0 REGULATORY EVALUATION

The applicant submitted TR-0116-20825, Revision 1 (Reference 3) in order to justify and demonstrate applicability of previously approved AREVA codes and methods for use in NuScale safety analyses. These AREVA codes and methodologies are associated with the fuel system design, and generally follow the guidance of SRP Section 4.2.

TR-0116-20825, Revision 1, (Reference 3) by itself does not include any safety analyses and instead would be referenced by a DC application, combined license application, or license amendment request. Therefore, this TR does not independently demonstrate compliance with any rules and regulations but instead would provide the tools to be used by other licensing actions to demonstrate compliance. Based on the intent of this TR, the staff does not make any findings regarding compliance with specific rules or regulations, but instead the staff considers the related rules, regulations, and guidance during the staff's review to determine if the previously approved AREVA codes and methods TRs are applicable to NuScale given the plant specific design differences.

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

2.1 Rules and Regulations Evaluation

Pursuant to Section 52.47 "Contents of applications; technical information" of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," an application for a standard DC must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. Specifically, under 10 CFR 52.47(a)(3), the application must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include, among other things, the design of the facility including (i) the principal design criteria (PDC) for the facility, (ii) the design bases and the relation of the design bases to the PDC; and (iii) information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the design will conform to the design bases with an adequate margin for safety; Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes minimum requirements for the PDC for watercooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing PDC for other types of nuclear power units. In terms of fuel system design, the NuScale design is similar in design and location to plants for which construction permits have previously been issued. This is supported by the NuScale gap analysis report (Reference 16) in which General Design Criterion (GDC) 10 is not listed as containing a gap.

Criterion 10, "Reactor design," requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). The SAFDLs associated with the NuScale plant design are defined by the NuScale standard plant design, which is currently under staff review. The focus of this TR is to demonstrate applicability for the codes and methods which can be used in other licensing actions (e.g. a DC application for NuScale) to analyze the margin to the SAFDL, as required by GDC 10.

2.2 Guidance Evaluation

NUREG–0800, (Reference 14) provides detailed review guidance that the staff finds acceptable in meeting the applicable regulatory requirements. In particular, Section 4.2, "Fuel System Design" of NUREG–0800 contains guidance relevant to this review. It should be noted that this TR does not provide an actual analysis of the NuScale fuel system design and rather provides an applicability report of AREVA codes and methods to the NuScale fuel system design. As such, the staff used the guidance found in NUREG–0800, Section 4.2 to identify the sensitive parameters for each respective analysis topic identified in TR-0116-20825, Revision 1. The staff then compared the NuScale design against the referenced AREVA TR range of applicability for each of these parameters to determine if the referenced AREVA TR is applicable for use in analyzing the NuScale fuel system design.

3.0 SUMMARY OF TECHNICAL INFORMATION

TR-0116-20825, Revision 1 (Reference 3) provides an applicability analysis of AREVA fuel system design analysis codes and methods for the NuScale Small Modular Reactor (SMR) design. The purpose of the TR is to provide a regulatory basis supporting the use of these codes and methods to support the NuScale DC submittal and specifically the fuel system design analysis.

3.1 BAW-10084P-A-03, Revision 3, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse"

Section 3 of TR-0116-20825, Revision 1 states that the limits and methodologies described in the cladding creep collapse methodology with the CROV code (which is used to define fuel rod parameters such that cladding creep collapse will not occur during the life of the fuel) will be used for NuScale fuel for the clad creep collapse analysis. It is further stated that BAW-10084P-A, Revision 3 (Reference 4) only contains creep correlations for Zircaloy-4. Therefore, consistent with the AREVA approach for M5 rods in pressurized-water reactors (PWRs), the creep correlation from BAW-10227P-A, Revision 1 (Reference 5) will be used in the CROV code.

Additionally, Section 3 of TR-0116-20825, Revision 1 (Reference 3) provides an applicability analysis of each chapter of the referenced methodology. This applicability analysis extends to the SER associated with the referenced approved methodology.

3.2 BAW-10227P-A, Revision 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel"

Section 4 of TR-0116-20825, Revision 1 states that the limits and methodologies described in the M5 license topical report (LTR) will be used for NuScale fuel in the following areas:

- Clad Stress Analysis

- Fuel Rod Buckling Analysis
- Clad Fatigue Analysis

The TR clarifies that only the portions of BAW-10227P-A, Revision 1 related to M5 fuel rods are applicable. Therefore, the portions related to assembly structural components are not applicable and not discussed in Section 4 of this SER.

NuScale notes that the SER for this LTR makes no restrictions as to fuel type and should therefore be applicable to NuScale fuel. It is also stated that this LTR has been approved for fuel with M5 cladding up to 62 GWd/MTU which bounds the anticipated operation of NuScale fuel.

3.3 BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code"

Section 5 of TR-0116-20825, Revision 1 (Reference 3) states that the limits and methodologies described in the COPERNIC TR will be used for NuScale fuel in the following areas:

- Clad Corrosion Analysis
- Fuel Rod Internal Pressure
- Fuel Centerline Melt Analysis
- Transient Clad Strain Analysis

The applicability review addresses thermal models, fission gas release, pellet and cladding mechanical models, and corrosion. NuScale provides the applicability ranges for the COPERNIC code (reproduced from the referenced topical report) and corresponding anticipated NuScale values, thereby supporting the applicability analysis.

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

Section 6 of TR-0116-20825, Revision 1 (Reference 3) states that the NuScale fuel rod bow evaluation is based on limits and methodologies described in the fuel rod bowing methodology, XN-75-32(P)(A) (Reference 8). The TR provides a comparison of the similarities and differences between the NuScale fuel assembly and the fuel assemblies which formed the basis for the referenced rod bowing methodology. The NuScale fuel assembly characteristics which differ from standard AREVA fuel assemblies are identified and the effect of these differences are analyzed.

3.5 EMF-92-116(P)(A), Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs"

Section 7 of TR-0116-20825, Revision 1 (Reference 3) states that the Generic Mechanical Design Criteria for PWR fuel designs (EMF-92-116(P)(A) Reference 9) is used. This generic mechanical design TR defines the SAFDLs that provide assurance of satisfactory performance for nuclear fuel and the methodologies used to demonstrate acceptable fuel performance. NuScale states that only parts of EMF-92-116(P)(A) (Reference 9) are applicable to the NuScale fuel design. The following analysis methodologies from EMF-92-116(P)(A) are stated to be applicable for NuScale fuel:

- Internal Hydriding
- Stress, Strain, or Loading Limits on Assembly Components
- Fretting Wear
- Axial Growth (Rod and Assembly)
- Assembly Liftoff
- Fuel Assembly Handling

4.0 TECHNICAL EVALUATION

4.1 BAW-10084P-A-03, Revision 3, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse"

TR-0116-20825, Revision 1 (Reference 3) states that the limits and methodologies described in the cladding creep collapse methodology with the CROV code will be used for the NuScale fuel clad creep collapse analysis. The staff notes that BAW-10084P-A, Revision 3 (Reference 4) only contains creep correlations for Zircaloy-4. Therefore, consistent with the AREVA approach for M5 rods in PWRs, the creep correlation from BAW-10227P-A, Revision 1 (Reference 5) will be used in the CROV code.

NuScale provides an assessment in Section 3.3 of TR-0116-20825, Revision 1 (Reference 3) to demonstrate that the CROV code and the associated methodology to evaluate creep collapse would be acceptable for NuScale. The referenced methodology (Reference 4) was modified in the portion which states that the largest potential for creep collapse is at 90 inches from the bottom of the fuel column. NuScale determined that it is not appropriate for NuScale fuel, which is less than 90 inches long. Therefore, NuScale performs a revised calculation to determine the location of the limiting axial node.

TR-0116-20825, Revision 1 (Reference 3) does not describe how this calculation is performed. In order to understand the revised methodology, the staff requested additional information in RAI-8727, Question 04.02-29594b (Reference 11). NuScale responded that the maximum fast flux and cladding temperature are determined at each time step for the NuScale fuel design using COPERNIC. This was the same methodology used in Reference 4 to determine that 90 inches was the appropriate axial location for AREVA large light-water designs investigated. Additionally, [ ]. The staff finds this approach conservatively over estimates the conditions.

Based on the staff's review of the differences between the NuScale and AREVA plant designs for the parameters important to clad collapse, the staff concludes that BAW-10084P-A, Revision 3 (Reference 4) is acceptable to evaluate the creep collapse of NuScale fuel with the following modifications:

- The creep correlation from BAW-10227P-A, Revision 1 (Reference 5) will be used in the CROV code.

- The COPERNIC code and methodology described in BAW-10231P-A, Revision 1 (Reference 7) will be used for creep collapse initialization.

- [

] as opposed to the existing methodology (Reference 4) that uses these values calculated at 90 inches from the bottom of the fuel stack that is not applicable to the NuScale fuel.

4.2 BAW-10227P-A, Revision 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel"

TR-0116-20825, Revision 1 (Reference 3) states that the limits and methodologies described in the M5 LTR will be used for NuScale fuel in the following areas:

- Clad Stress Analysis
- Fuel Rod Buckling Analysis
- Clad Fatigue Analysis

NuScale notes that the SER for BAW-10227P-A, Revision 1 (Reference 5) makes no restrictions as to fuel type and should therefore be applicable to NuScale fuel. The staff also notes that this TR has been approved for fuel with M5 cladding up to 62 GWd/MTU which bounds the anticipated operation of NuScale fuel. The staff confirmed that BAW-10227P-A, Revision 1 does not contain any conditions or limitations which would prevent its use in the evaluation methodology for the NuScale fuel assembly design. Additionally, the staff compared the NuScale fuel design against the parameters important to the clad stress, fuel rod buckling, and clad fatigue analyses (e.g. clad material, clad dimensions, etc.). The staff confirmed that the NuScale and standard AREVA fuel designs are identical in these parameters.

Based on the above discussion, the staff concludes that BAW-10227P-A, Revision 1, (Reference 5) is acceptable to evaluate the following design criteria for NuScale fuel:

- Clad Stress Analysis
- Fuel Rod Buckling Analysis
- Clad Fatigue Analysis

4.3 BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code"

BAW-10231P-A Revision 1 presents a fuel design tool developed by FRAMATOME to evaluate fuel rod thermal-mechanical performance. TR-0116-20825, Revision 1 (Reference 3) states that the limits and methodologies described in the COPERNIC LTR will be used for NuScale fuel in the following areas:

- Clad Corrosion Analysis
- Fuel Rod Internal Pressure
- Fuel Centerline Melt Analysis
- Transient Clad Strain Analysis

Additionally, although not specifically discussed in Section 5 of TR-0116-20825, Revision 1 (Reference 3), it is stated in Section 3 that COPERNIC and its associated methodology will be used for creep collapse initialization. This use of COPERNIC is evaluated in Section 4.1 of this SER.

The methodology for Loss-Of-Coolant Accident (LOCA) initialization in BAW-10231P-A, Revision 1 (Reference 7) will not be used for LOCA initialization of NuScale fuel and is therefore not part of the applicability analysis, nor part of the staff's safety evaluation (SE).

Table 5-1 of TR-0116-20825, Revision 1 (Reference 3) presents the range of applicability for COPERNIC. The staff reviewed the applicability range and confirmed that the planned operation of NuScale fuel is bounded by the COPERNIC range of application for the parameters presented in Table 5-1.

COPERNIC has been designed to model light-water reactor (LWR) fuel rods in PWR conditions. The core of the NuScale integral PWR is very similar to that of a large commercial PWR with fuel rods grouped in assemblies and cooled by flowing water. One notable difference however, is that the NuScale reactor core will be cooled by water flowing under natural circulation, where a typical PWR is cooled via pumped water. Other differences include the expected power level, coolant pressure, coolant inlet temperature, and core height.

The following sub-sections examine these differences and evaluates the applicability of COPERNIC in the range that the NuScale SMR will operate.

4.3.1 Fuel Rod Geometry

The NuScale fuel design parameters are very similar to those of an AREVA 17x17 PWR fuel assembly. The differences between the NuScale fuel and an AREVA 17x17 PWR fuel assembly are summarized in Table 2-1 of TR-0116-20825, (Reference 3). It can be seen from this table that the primary differences are in the stack and rod length, the spacer grid span length, and the initial fill pressure. COPERNIC has been used to model short fuel rods that have been irradiated in various test reactors as part of the code validation and has no limitations related to fuel stack or rod length. COPERNIC does not model any effects of spacer grids and therefore a slight change in spacer design will have no impact on the ability of the code to model this fuel. Finally, commercial and test reactor fuel has been irradiated with a wide variety of initial fill gas conditions down to 1 atm (14.7 psig) of air.

Based on the staff's evaluation in the paragraph above, the differences between the NuScale fuel assembly design and the COPERNIC fuel assembly range of applicability, the staff finds that there are no limitations in COPERNIC that would invalidate its ability to model the geometry of the NuScale fuel.

4.3.2 Reactor Coolant Conditions

The largest difference with regard to the fuel operation in the NuScale reactor is the core and coolant operating conditions. Table 2-2 of TR-0116-20825, Revision 1 (Reference 3) summarizes the differences between the NuScale operating conditions and a typical 17x17 PWR. The greatest differences from this table are the system pressure and coolant temperature which are both lower than a typical PWR. The staff confirmed that the COPERNIC steam tables can calculate the saturation properties for water at the expected inlet and outlet conditions.

Section 5.2.1.1 of TR-0116-20825, Revision 1, (Reference 3) discusses coolant-cladding outside surface heat transfer. It is stated that two different heat transfer models are used in the

COPERNIC code, and justification is provided for the applicability of these models to NuScale. In the response to RAI-8722, Question 04.02.29594c (Reference 11), NuScale notes that the two-phase correlation was fitted for a pressure range which bounds the NuScale coolant pressure. The staff confirmed that the NuScale coolant pressure is within the range given in Section 5.2.1.1 of TR-0116-20825, Revision 1 (Reference 3) and that the two-phase correlation was appropriate for the general NuScale design. Based on this evaluation and a comparison with the two-phase flow correlation used in the staff's fuel-rod thermal-mechanical performance confirmatory tool, FRAPCON (Reference 15), the staff finds the two-phase correlation used in COPERNIC to be acceptable for use to analyze NuScale.

The staff notes that the single-phase heat transfer correlation used is based on forced flow, but the NuScale reactor relies on natural circulation. Based on a review of the general reactor design, the staff agrees that a gravity head will cause the NuScale convection in the core to behave similarly to that of a standard reactor design which relies on pumps to maintain flow. The NuScale flow rates are less than what is typically found in traditional PWRs, and this results in a reduction in the Reynolds number. NuScale justified the use of their single-phase flow correlation based on a comparison of the Reynolds number with that of the threshold above which forced convection is typically seen. NuScale provided additional support in RAI-8727, Question 04.02-29594d (Reference 11) to support the use of their single-phase flow correlation. The staff reviewed the information provided and confirmed that existing tests have been conducted for Reynolds number ranges which bound NuScale and support the use of NuScale's single-phase flow correlation. This correlation compares well with the Dittus-Boelter correlation which is used by the staff's confirmatory tool, FRAPCON. Based on the justification provided and the comparisons with a similar correlation, the staff therefore concludes that the NuScale single phase flow correlation is acceptable for use as described.

Additionally, the staff reviewed the SER for BAW-10231P-A, Revision 1 (Reference 7) and determined that there are no limitations that would invalidate its ability to model the cladding coolant heat transfer of the NuScale fuel.

4.3.3 Model Applicability

The NuScale reactor will use UO_2 and UO_2-Gd_2O_2 fuel which is the same as PWR fuel. The material property models for the fuel include;

- Melting temperature
- Specific heat and enthalpy
- Thermal conductivity
- Emissivity
- Thermal expansion
- Densification and swelling

All of these properties other than densification and swelling are validated down to room temperature and therefore would not be invalidated by fuel running at lower power levels and lower pellet surface temperature. The staff recognizes that in a traditional LWR core there are always fuel rods running at low power such as those expected in the NuScale reactor. Likewise with densification and swelling, no temperature effect has been observed within a large range of pellet temperatures that bounds those expected for the NuScale fuel. Therefore, the staff finds that BAW-10231P-A, Revision 1 (Reference 7) is applicable for modeling NuScale fuel pellets.

The NuScale fuel will use M5™ cladding. The properties of this alloy are known and models have been placed into COPERNIC. The material property models for the cladding include;

- Specific heat and enthalpy
- Thermal conductivity
- ZrO_2 conductivity
- Emissivity
- Thermal expansion
- Elastic modulus
- Yield strength
- Ultimate tensile strength
- Uniform elongation
- Axial irradiation growth
- Thermal and irradiation creep

All of these properties other than axial irradiation growth and irradiation creep are validated down to room temperature and are not expected to be invalidated by fuel running at lower power levels and lower cladding surface temperature. For irradiation growth, no temperature effect has been observed within a significant large range of coolant temperatures from commercial and test reactors that bounds those expected for the NuScale cladding. The irradiation creep model has been developed and validated within a fairly narrow range of temperature conditions. This range and the applicability to NuScale fuel will be discussed in a following subsection. Therefore, the staff finds that BAW-10231P-A, Revision 1 (Reference 7) is applicable for modeling M5 cladding in the NuScale fuel assembly design.

COPERNIC contains some models that have been developed to describe fuel performance. The fuel performance models include;

- Fission gas release
- Fuel cracking and relocation
- Cladding corrosion and hydrogen pickup
- Fuel/cladding gap conductance
- Radial power profile
- Gaseous swelling
- High burnup rim formation

The fission gas release, fuel cracking and relocation, fuel/cladding gap conductance, radial power profile and high burnup rim formation models have all been validated for low power LWR rods with fuel temperatures that are well within the fuel temperatures for NuScale. The gaseous swelling model is applicable at high temperature such as those seen during large power ramps. The gaseous swelling model therefore applies to any similar ramps included in NuScale AOOs of interest.

The cladding surface temperature is of prime importance for the cladding corrosion and hydrogen pickup models and this temperature is somewhat lower for the NuScale cladding than for PWR 17x17 cladding. The following section gives estimates of these temperatures and the staff's assessment of the applicability of the creep and corrosion models to the expected temperature ranges.

Coolant and Cladding Temperature

In order to determine if the temperatures anticipated for the cladding in the NuScale reactor fall within the range of validation for the cladding irradiation creep and cladding corrosion and hydrogen pickup models, FRAPCON was used to create a sample run of NuScale fuel. The staff independently confirmed that the thermal solution is adequate for the NuScale fuel assembly.

A sample calculation was performed for NuScale fuel irradiated at the core average linear heat generation rate (LHGR) of 2.5 kW/ft (8.2 kW/m) for 40 GWd/MTU. NuScale has not provided axial power profiles so typical PWR axial power profiles were assumed for beginning, middle, and end of cycle. The results of the coolant and cladding temperature calculations are shown in Table 1.

Table 1. Staff Confirmatory Coolant and Cladding Temperatures for NuScale SMR and AREVA 17x17 PWR Fuel

Output Value	NuScale	17x17 PWR
Coolant Temperature	Inlet=503°F Outlet=597°F Average=546°F	Inlet=547°F Outlet=616°F Average=580°F
Cladding Surface Temperature	528°F-630°F	562°F-653°F
Cladding Midwall Temperature	532°F-643°F	572°F-678°F

The staff's confirmatory run is consistent with NuScale's analysis which states that the anticipated cladding temperatures for NuScale fall within the range of validation for the cladding irradiation creep and cladding corrosion, and hydrogen pickup models. Based on the staff's review of NuScale's analysis as supported by the staff's confirmatory analysis, the staff finds that COPERNIC is applicable for use in analyzing cladding temperatures in the NuScale fuel system design.

Cladding Irradiation Creep

The COPERNIC cladding irradiation creep model has been validated over the ranges given in Table 5-4 of TR-0116-20825, Revision 1 (Reference 3).

Typically cladding midwall temperature is used to calculate cladding creep. The NuScale midwall temperature is expected to be within the COPERNIC validation range of the irradiation creep database (see Table 5). Additionally, the staff compared the COPERNIC flux range with the calculated NuScale value provided in TR-0116-20781-P, Revision 0 (Reference 12) and confirmed that the NuScale value was within the COPERNIC validation range.

Based on the NuScale conditions being within the COPERNIC validation ranges as noted in the above staff evaluation, the staff finds that the cladding irradiation creep model in COPERNIC is valid over the expected range of temperature, fast neutron flux and stress for the NuScale fuel.

Cladding Corrosion and Hydrogen Pickup

TR-0116-20825, Revision 1 (Reference 3) discusses the corrosion and hydrogen pickup models in COPERNIC and demonstrates that the calibration database bounds the expected temperature and heat flux for NuScale. The staff concurs with this assessment based on the calculated cladding temperatures shown in Table 1.

The cladding corrosion and hydrogen pickup models in COPERNIC are expected to provide good predictions of corrosion and hydrogen pickup for the NuScale reactor design.

4.3.4 Summary of BAW-10231P-A Revision 1 Code and Methodology Applicability to NuScale Fuel

Based on the staff's review of the SE for BAW-10231P-A, Revision 1 in comparison with the NuScale fuel assembly design, the staff concludes the following regarding the applicability of COPERNIC for the analysis of NuScale:

- There are no limitations in BAW-10231P-A, Revision 1 that would prevent its use to model the geometry of the NuScale fuel assembly.

- There are no limitations in BAW-10231P-A, Revision 1 that would prevent its use to model the cladding coolant heat transfer of the NuScale fuel.

- The material property and fuel performance models in BAW-10231P-A, Revision 1 are applicable to the fuel and cladding materials and those conditions that they will be exposed to during irradiation in the NuScale reactor.

Based on the staff's evaluation presented in Section 4.3.3 of this SE, the staff concludes that BAW-10231P-A, Revision 1 (Reference 3) is acceptable to evaluate the following design criteria for NuScale Fuel.

- Clad Corrosion Analysis
- Fuel Rod Internal Pressure
- Fuel Centerline Melt Analysis
- Transient Clad Strain Analysis
- Creep Collapse Initialization

The applicability analysis of COPERNIC provided in TR-0116-20825, Revision 1 (Reference 3) did not address LOCA initialization. The methodology for LOCA initialization related to fuel will therefore be covered in the evaluation of the NuScale LOCA TR.

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

TR-0116-2-0825-P, Revision 1 (Reference 3) states that the limits and methodologies described in the fuel rod bowing methodology will be used for NuScale fuel for the fuel rod bow evaluation.

Section 6.1.1 of TR-0116-20825, (Reference 3) lists the primary fuel assembly design contributors to fuel rod bowing and Table 2-1 presents a comparison of the NuScale fuel design parameters with those of a reference AREVA 17x17 PWR fuel assembly. Of these, only spacer grid span length differs between the NuScale fuel assembly design and the reference AREVA

17x17 fuel assembly design. The span lengths are similar, but the NuScale fuel assembly spacer grid span length is shorter and therefore more conservative in terms of fuel rod bowing. Section 6.1.1 of TR-0116-20825, Revision 1 (Reference 3) also presents some environment parameters which have a secondary effect. The NuScale reactor design results in less limiting environmental parameters. The staff reviewed the NuScale fuel assembly design parameters and confirmed that the values are similar to the referenced 17x17 AREVA PWR fuel assembly and that the parameters important to fuel rod bowing are less limiting for the NuScale fuel assembly design. Therefore, the staff expects that the NuScale fuel assembly will have less propensity for rod bowing than an AREVA 17x17 fuel assembly.

In Section 6.1.2 of TR-0116-20825, Revision 1 (Reference 3), NuScale discusses the critical heat flux (CHF) penalties methodology for bowed fuel and provided justification for the use of these penalties based on no measurable trends in departure from nucleate boiling ratio penalty with mass velocity. The staff reviewed the justification provided and determined that the applicant sufficiently demonstrated that the NuScale fuel assembly design parameters are bounded by those used to develop the CHF penalty methodology. Therefore, the staff finds that the CHF penalty is acceptable for use in the NuScale fuel assembly bowing analysis.

In Section 6.1.3 of TR-0116-20825, Revision 1 (Reference 3), NuScale discusses the LHGR penalties methodology for fuel assembly rod bowing. NuScale states that the NuScale fuel assembly water-to-fuel volume ratio is bounded by values presented in Table 15.1 of Supplement 4 of XN-75-32(P)(A), (Reference 8) and therefore the power peaking augmentation is applicable. The staff reviewed the references cited by NuScale and finds that LHGR penalties are appropriate for the NuScale fuel assembly design.

Based on the review and findings listed above, the staff concludes that XN-75-32(P)(A), Supplements 1 through 4, Computational Procedure for Evaluating Fuel Rod Bowing is acceptable to perform the fuel rod bow evaluation of NuScale fuel.

4.5 EMF-92-116(P)A, Revision 0, Generic Mechanical Design Criteria for PWR Fuel Designs

TR-0116-20825, Revision 1 (Reference 3) states that the Generic Mechanical Design Criteria for PWR fuel designs will be used for NuScale fuel in the following areas:

- Shipping and Handling Stress Analysis
- Fuel Assembly/Component Stress Analysis
- Flow Induced Vibration Assessment
- Axial Growth (Rod and Assembly)
- Fuel Lift Analysis
- Internal Hydriding

The staff compared the NuScale fuel assembly design with the fuel assemblies used in EMF-92-116(P)A (Reference 9) and agrees that the physical NuScale fuel assembly design lies within the range of applicability that has already been approved in EMF-92-116(P)A (Reference 9).

Although the staff agrees that the NuScale fuel assembly design is not significantly different than the assembly designs considered in the referenced TR, the staff is concerned that the empirical growth models could potentially be impacted by hold-down force, hydraulic lifting force, and temperatures. AREVA made an assessment of assembly growth in Reference 13 stating:

> In recent times much attention has been given to the growth of AREVA fuel assemblies with M5 guide tubes, principally those in the US. Unlike fuel rod growth, whose predictable growth with increasing burnup is largely insensitive to fuel assembly design, fuel assemblies with M5 guide tubes displayed a variation according to specific design features. This trend is consistent with historic performance of other alloys such as Zr-4.

Given this stated variation in assembly growth in assemblies with both M5 and Zircaloy-4 guide tubes according to specific design features, the staff recognizes the importance of a detailed surveillance program to confirm that the empirical growth models perform as expected. The staff recognizes that the scope of TR-0116-20825, Revision 1 (Reference 3) does not include a detailed surveillance plan and that any applicant referencing this TR would need to cover the surveillance plan separately.

Based on the staff's review of the basis for TR EMF-92-116(P)A (Reference 9) and comparison with the NuScale fuel assembly design, the staff finds that EMF-92-116(P)A, Revision 0 is acceptable to evaluate the following design criteria for NuScale fuel:

- Shipping and Handling Stress Analysis
- Fuel Assembly/Component Stress Analysis
- Flow Induced Vibration Assessment
- Axial Growth (Rod and Assembly)
- Fuel Lift Analysis
- Internal Hydriding

5.0 STAFF CONCLUSIONS

The staff has completed its review of TR-0116-20825, (Reference 3) and concludes that the applicant has demonstrated that the AREVA fuel system design codes and methods cited in the TR, with the stated modifications, are applicable for use in NuScale fuel system analyses. The staff reached its conclusions by (1) reviewing conditions/limitations of the referenced approved TRs, (2) independent verification that the expected NuScale parameters fall within the validation limits of the respective referenced approved TRs, and (3) evaluation of the justification provided in TR-0116-20825, Revision 1 (Reference 3).

The staff, therefore, approves the use of AREVA fuel codes and methodologies as described in TR-0116-2082, Revision 1 (Reference 3) to analyze the NuScale fuel system design.

6.0 CONDITIONS AND LIMITATIONS

The staff's evaluation of TR-0116-20825P, Revision 1 (Reference 3) was limited to the analyses and technical areas presented in the TR. In particular, the staff notes that no information was provided which would support fuel operation beyond that associated with plant baseload operation. Any applicant or licensee referencing this TR who wishes to operate in modes other than baseload would need to address this in their application or license amendment request.

7.0 REFERENCES

1. "NuScale Power, LLC Submittal of TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 0 (NRC Project No. 0769)", dated March 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16095A244).

2. "NuScale Power, LLC Request for Suspension of Acceptance Review of TR-0116-20825, "Applicability of AREVA Fuel Methodology for NuScale Design," Revision 0 (NRC Project No. 0769)", dated June 2016 (ADAMS Accession No. ML16155A449).

3. "NuScale Power, LLC Submittal of Topical Report TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design", Revision 1 (NRC Project No. 0769)", dated July 2016 (ADAMS Accession No. ML16187A016).

4. BAW-10084P-A-03, Revision 3, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse", August 1995 (ADAMS Accession No. 9507260025).

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

6. BAW-10183P-A, Revision 0, "Fuel Rod Gas Pressure Criterion" (FRGPC), February 1996 (ADAMS Accession No. 9507270402).

7. BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code", January 2004 (ADAMS Accession No. ML042930236).

8. XN-75-32(P)(A), Supplements 1-4, "Computational Procedure for Evaluating Fuel Rod Bowing", February 1983 (ADAMS Accession No. ML081710709).

9. EMF-92-116(P)(A), Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs", February 2015 (ADAMS Accession No. ML003681168).

10. ANF-89-060(P)(A), Supplement 1, "Generic Mechanical Design Report High Thermal Performance Spacer and Intermediate Flow Mixer", February 1991 (ADAMS Accession No. ML9104090206).

11. LO-0317-53210, Revision 0, "NuScale Response to NRC Request for Additional Information Letter No. 12 for TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1" (ADAMS Accession No. ML17068A190).

12. TR-0116-20781-P, Revision 0, "Fluence Calculation Methodology and Results", December 2016 (ADAMS Accession No. ML17005A146).

13. G. L. Garner and J. P. Mardon 2011, "Alloy M5 cladding performance update" Nuclear Engineering International.

14. NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” dated March 2007 (ADAMS Accession No. ML070810350).

15. FRAPCON-4.0: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup September 2015 (ADAMS Accession No. ML16118A434).

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

Section B

Licensing Topical Report

Applicability of AREVA Fuel Methodology for the NuScale Design

June 2016

Revision 1

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

The applicability of the AREVA seismic methodology is addressed in a separate NuScale submittal.

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 7). Only one modification to the cladding creep method is required for application to the NuScale design. All other AREVA methods addressed in this report are directly applicable to the NuScale design.

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 modification made to BAW-10084P-A for application to the NuScale fuel design.

Executive Summary

NuScale intends to use NRC approved AREVA methodologies to perform 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. Seismic analysis methodology is addressed in a separate NuScale submittal. This report provides justification for using AREVA's cited NRC approved codes and methods 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 identical to fuel supplied by AREVA to W 17x17 PWR units with respect to cladding material, cladding thickness, fuel pellet dimensions, and cladding and pellet material properties.

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 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 fuel pellet and cladding in the NuScale fuel design are similar to those in a typical 17x17 PWR, for which COPERNIC is approved.

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 identical to fuel supplied by AREVA to W 17x17 PWR units with respect to cladding material, cladding thickness, fuel pellet dimensions, and cladding and pellet material properties. The creep collapse analysis cannot be performed at an axial elevation of 90 inches, as specified in Reference 1, due to the shorter active fuel length of the NuScale fuel design. Rather, the creep collapse calculation is performed at a composite axial location that combines the most limiting neutron flux at any position along the rod with the peak cladding temperature to ensure a conservative prediction of cladding creep.

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 are acceptable for 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 7). 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 use existing, NRC-approved AREVA methodologies to perform associated analyses of the NuScale fuel design. One modification to BAW-10084P-A, required to support extension of its applicability to the NuScale fuel design, is described in detail and requires NRC approval. Therefore, NuScale also requests NRC approval of this modification.

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, and one modification, for the analysis of NuScale fuel in the Design Certification Application.

1.1 Purpose

This report documents the applicability of the following NRC approved codes and methods used 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)
- XN-75-32(P)(A), Supplements 1 through 4, Computational Procedure for Evaluating Fuel Rod Bowing (Reference 5)

This report concludes that these AREVA codes and methods, in addition to one modification, are applicable to the NuScale 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 7). The cited AREVA methodologies are applied to the fuel design, fuel mechanical analysis, and fuel thermal-mechanical analysis work. AREVA's seismic methodology is not included in the scope of this topical report. Also, 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

The cited AREVA methodologies supporting the AREVA fuel design work were 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. The applicability of AREVA's seismic methodology to the NuScale fuel design is being provided in a separate NuScale submittal.

To assist in the review of the applicability of AREVA methodologies, 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 95.89 inches. Due to the reduced height, and the use of span lengths between spacer grids that are typical for operating PWR plants, the assembly has a total of five spacer grids. The HTP™ grids are welded to the guide tubes, while the HMP™ grid is captured by rings welded to the guide tubes. The design includes features as described in the following subsections.

Fuel rod with alloy M5® fuel rod cladding

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

The nominal density of the pellets is 96 percent theoretical density (TD) with a possible enrichment up to 5.0 weight percent ^{235}U.

Zircaloy-4 HTP™ upper and intermediate spacer grids

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

Alloy 718 HMP™ lower spacer grid

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

Bottom Nozzle with Mesh Filter Plate

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

Zircaloy-4 MONOBLOC™ guide tubes

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

Reconstitutable Top Nozzle

The top nozzle consists of a 304-stainless steel frame that is attached to the fuel assembly with quick disconnect features at each of the 24 guide tube locations. The NuScale top nozzle design is different from AREVA's standard 17X17 PWR product with respect to the through hole requirements in the grillage to accommodate the NuScale top-entry incore detectors. The top nozzle design incorporates four sets of two-leaf hold-down springs made of Alloy 718.

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

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)	95.89	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)	3.1	16
System Pressure (psia)	1850	2280
Core Tave (F)	547	584
Linear Heat Rate (kW/m)	8.2	18.0
RCS Inlet Temperature (F)	503	547
RCS Reynolds Number	76,000	468,000
Maximum Fuel-Assembly-Average burnup (GWD/mtU)	{{ }}2(a),(c),ECI	51
Fuel Assemblies in Core	37	193
Fuel Assembly Loading (kgU)	249	455
Core Loading (kgU)	9,213	87,815

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 7). Table 2-3 focuses only on the sections of the SRP that are being addressed using AREVA methodologies referenced herein. 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 7)	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-10227P-A (Reference 2)
Fuel Rod Buckling Analysis	1.A.i	
Clad Fatigue Analysis	1.A.ii	
Clad Corrosion Analysis	1.A.iv	BAW-10231P-A (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-10084P-A (Reference 1) BAW-10227P-A (Reference 2)
Rod Bow Evaluation	1.A.v	XN-75-32(P)(A) (Reference 5)

The applicability of each respective AREVA methodology listed in Table 2-3 to the NuScale fuel design is described in Sections 3 through 7 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-10084P-A, Revision 3 (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-10227P-A, Revision 1 (Reference 2) extended the application of the CROV code to fuel designs with AREVA's advanced cladding M5®.

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 identical to fuel supplied by AREVA to W-type 17x17 PWR units with respect to cladding material, cladding thickness, and fuel pellet dimensions and properties. Below is a chapter by chapter summary of the CROV topical report. Chapter numbers used in this summary refer to the Chapter numbers in Reference 1.

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 provides 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 AREVA Topical Report BAW-10138P-A, Revision 0 (Reference 3). The EGCF is developed based on **[**

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

The pellet and cladding radial dimensions of the NuScale fuel design are identical to current AREVA fuel and therefore Chapter 2 is applicable for use with the NuScale fuel rod design.

Chapter 3 *Creep Collapse Analytical Method*

Chapter 3 provides 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 cladding material and radial dimensions of the NuScale fuel design are identical to current AREVA fuel designs. 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 provides the basis for the fuel rod design and operational parameters to be used for the creep collapse calculations. The fuel rod geometry, fuel stack axial gap, power histories, delta pressures, flux, and cladding temperature inputs 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. AREVA Topical Report BAW 10231P-A, Revision 0, (Reference 4) discusses the use of COPERNIC for generating pressures and temperatures for input to the creep collapse analysis (CROV calculation).

The cladding material and radial dimensions of the NuScale fuel design are identical to current AREVA fuel designs. Additionally, the NuScale design operating parameters (i.e., cladding temperatures and system pressure) 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 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 AREVA Topical Report BAW-10183P-A, Revision 0 (Reference 3). After the initial section, a number of RAIs 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 RAIs and responses are detailed examples of how creep collapse calculations are **[**

] .

As stated in the sections above, the NuScale fuel design radial dimensions and materials are identical to current AREVA fuel designs. Additionally, the relevant NuScale design operating parameters (i.e., system pressure and cladding temperatures) 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.

3.2.2 Technical evaluation report

The TER evaluation reviews the updated fuel cladding creep model proposed for usage and the improved creep collapse calculation methodology which uses the EGCF. The TER finds the improvements to the creep collapse modeling to be acceptable within a **[**

]ECI

Thermal-mechanical analyses for the NuScale fuel predict cladding operating temperatures well below 700 degrees-F 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 **[** **]** The axial location of 90 inches is from the bottom of the fuel column for the 12 foot active length. Studies have shown that this location, 90 inches, experiences the worst combination of fast neutron flux and cladding temperature that 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*

AREVA Topical Report BAW-10084P-A, Revison 3 (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 methodology is applicable to the NuScale fuel design with a minor adjustment to the location of the limiting axial elevation, as discussed in Section 3.3.1, to account for the shorter length of the NuScale fuel rods.

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

AREVA Topical Report BAW-10227P-A, Revision 1 (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 M5® fuel rod cladding only. Assembly structural components are not made of M5® material and therefore these components are not discussed in this applicability report.

4.1 *Applications*

The Reference 2 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 identical to fuel supplied by AREVA to W-type 17x17 PWR units with respect to cladding material, cladding thickness, fuel pellet dimensions, and cladding and pellet material properties. Below is a chapter summary of Reference 2.

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 identifies 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 provides 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 stress, buckling, and fatigue analyses rely mainly on radial fuel dimensions and plant operating conditions (namely operating pressures and temperatures). The radial dimensions of the NuScale fuel design are identical to that of current AREVA fuel. The NuScale design operating pressures and temperatures are bounded by that of conventional PWRs. Therefore, Chapter 3 is applicable for use with the NuScale fuel design.

Chapter 4 *Accident Criteria and Evaluation*

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

Chapter 5 M5® Material Properties

Chapter 5 contains the M5® material properties which are primarily a function of cladding temperatures. The cladding temperatures 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 the NuScale fuel design.

Appendices 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 the cladding fatigue, stress, and buckling analyses, 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.

Appendices 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 is not 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.

Appendices 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 are 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 are performed. The methods described in the main body of the topical report are unchanged.

The fuel rod design for the NuScale design is radially identical to the AREVA W-type 17x17 fuel rod design but with a shortened fuel stack (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 Reference 2 are applicable to the NuScale fuel rod thermal mechanical design.

Appendix J

Appendix J provides detailed information about the M5® cladding creep in the calculation of fuel-clad liftoff. Rod internal pressures at NuScale are limited to system pressure, thereby precluding any risk of fuel-clad lift-off. 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 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 Reference 2 for AREVA PWR fuel reload designs. In general, the topical report is limited in application to fuel rod burnups below 62 GWD/mtU for AREVA fuel designs. This limit is 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.

4.3 *NuScale Design Differences and Requirements*

4.3.1 NuScale Design Applicability

The SER for 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 the cladding fatigue, stress, and buckling analysis. 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*

Reference 2 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-10231P-A, 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 design.

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 fuel pellet and cladding in the NuScale fuel design are expected to be the same as those in a typical 17x17 PWR, for which COPERNIC is approved. Therefore, the material properties approved for use in COPERNIC 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 to the NuScale design, system pressure is lower than that of a typical PWR. With the lower system pressure, the coolant saturation temperature decreases approximately 30 degrees-F, which lowers the cladding surface temperature in the two phase flow region. Average coolant velocity for the NuScale design is approximately 20 percent of the coolant velocity in a typical PWR. The slower coolant velocity reduces 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 8.2 kW/m, 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. 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 AREVA Topical Report BAW-10231P-A, Revision 1 (Reference 4).

Table 5-1 COPERNIC Validity Range

Parameter	**Range of Validity**
Fuel	UO_2, UO_2-Gd_2O_3
Cladding	M5®
[	
	]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 Reference 4 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 impacts 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 has 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. Because of Dittus-Boelter's wide acceptability (used in FRAPCON) and application range (Re > 10,000), the heat transfer coefficient predicted from the **[ ]** correlation was compared to the Dittus-Boelter equation and was found to be applicable to 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 Begel data (Reference 13). **[**

].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, pellet dimensions and materials, 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 has a 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 8.2 kW/m. The AREVA database supporting Reference 4 has more measurement data **[ ]**.ECI Reference 4, Figure 4-73 shows the trend is well suited and temperature prediction does not have any bias relating to LHGR. Reference 4, Figure 4-70 demonstrated that the temperature is **[ ]**.ECI The database shown in Reference 4, Tables 4-4 and 4-5 fully addresses the NuScale design operating conditions in terms of burnup and LHGR. Note that the Halden data in Reference 4, Table 4-5 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 600 kg/(m^2s). 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 Reference 4, Figures 5-9 and 5-10 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. Reference 4, Figure 5-11 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

Reference 4 discusses pellet relocation, densification and solid swelling, and gaseous swelling models. These models are a function of **[ ]**. ECI NuScale fuel temperatures and burnups are 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 **[**

] level is from the measurements on fuel rods before pellet-cladding gap closure. The **[**

]. Combining the calibration database and validation database, Reference 4, Sections 7.1.3.4 and 7.2.3.4 list the validation range for both **[**

] (see Table 5-4). Note that the overall application range of COPERNIC is shown in Reference 4, Chapter 1. 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 will 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 [ ], 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 Reference 4, there are sufficient measurement data [ ], 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 Poisson's ratio are acceptable for NuScale application.

5.4 *Safety Evaluation and Technical Evaluation Report*

The safety evaluation for AREVA Topical Report BAW-10231P-A, Revision 1 (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 Reference 4 for fuel licensing applications up to a rod average burnup of 62 GWD/mtU. This limit is applied to the NuScale fuel design.

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

Reference 4 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 is 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 (F_Q) and departure from nucleate boiling ratio rod bowing penalties is documented in AREVA 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 Reference 5 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 is 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 AREVA Topical Report XN-75-32(P)(A), 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 RAIs 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 8).

6.1.1 Fuel Design Comparison

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

- 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,
- fuel rod cladding thickness,
- cladding material,
- spacer grid design, and
- span length between the fuel rod supporting spacer grids.

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

- coolant cross flow forces,
- coolant temperature and its impact on the fuel rod cladding material creep rate, and
- 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 that 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 AREVA rod bow correlation and the applicability to 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 8.

6.1.2 Justification of CHF Penalties

The Reference 5 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 CHF in bowed geometry. The magnitude of gap closure is based on rod-to-rod spacing measurements, while the reduction of CHF in bowed geometries is conservatively estimated from the results of CHF testing for bowed geometries.

The reduction of CHF in bowed geometries given in Reference 5 and 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 9. The range of tested conditions from the bowed-to-contact CHF tests was:

- Pressure: 105 ~ 170 ata**,**
- Mass Velocity: 7.3 x 10^6 ~ 17.0 x 10^6 kg/hr-m^2
- Inlet Temperature: 200 ~ 320^{o}C

The nominal operating conditions for the NuScale core are approximately a pressure of 126 ata, a mass velocity of 2.2 x 10^6 kg/hr-m^2, and an inlet temperature of 256 degrees-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 9). 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.

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

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

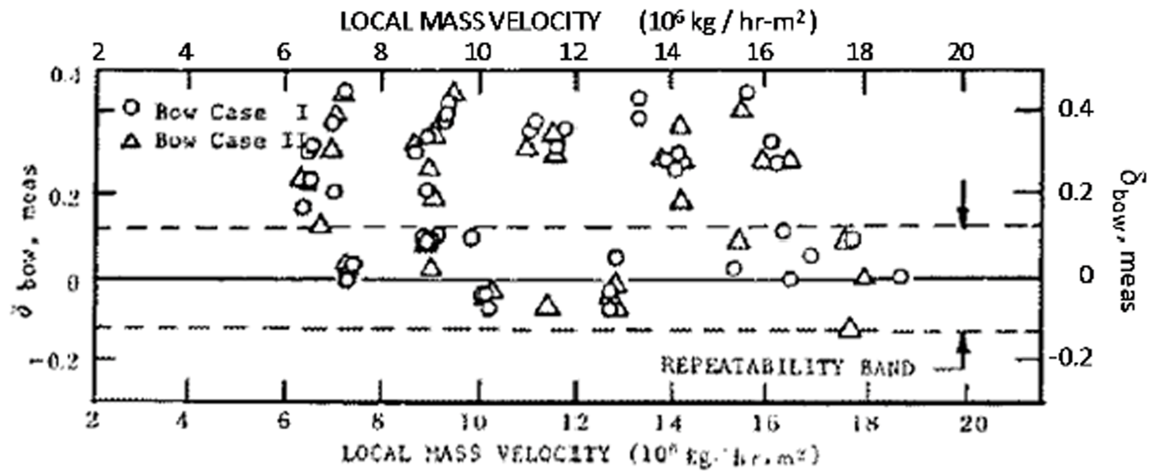
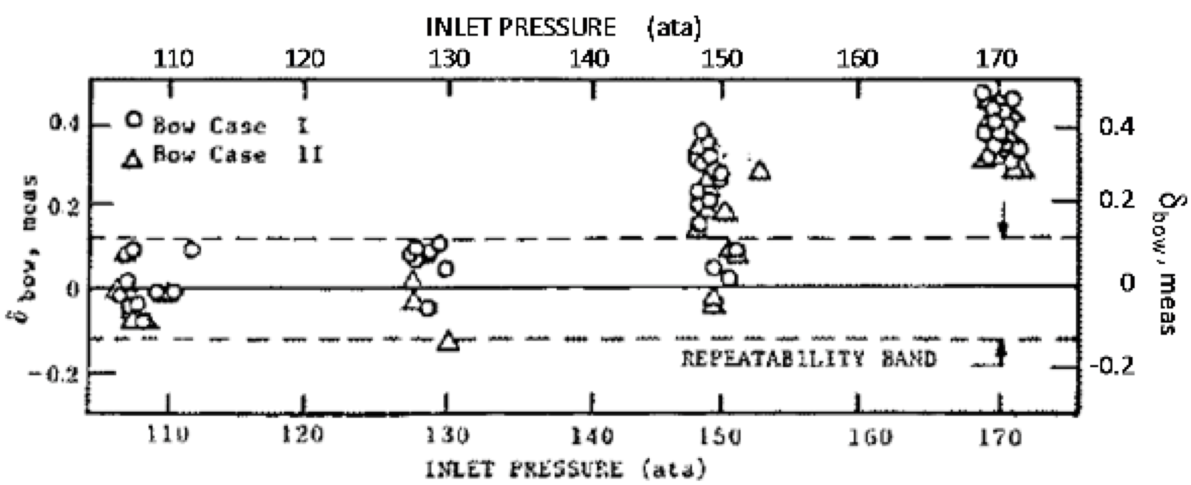


Figure 6-2: Bow Effect Parameter (δ_{bow}, meas) as a Function of Inlet Pressure



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 AREVA Topical Report XN-75-32(P)(A), Supplements 1-4 (Reference 5) 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 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 Reference 5 methodology to the HTP™ fuel design has been approved in Reference 8. 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*

AREVA Topical Report XN-75-32(P)(A), Supplements 1-4 defines 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). Reference 6 also cites or describes the methodologies to be used to demonstrate acceptable fuel performance.

Only certain sections of Reference 6 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 does not utilize AREVA methodology for neutronics analysis, safety analysis, and a majority of the thermal-hydraulic analyses.

Reference 6 frequently refers to separately approved methodologies that are used to demonstrate compliance to criteria. Where this is the case for a methodology being applied to NuSclale, the referenced methodology is discussed in other sections of this report. In some cases, the applicability of the method currently referenced in Reference 6 is evaluated, as in the case of the rod bowing methodology (see Section 6.0). In other cases, a more contemporary code that replaces one of the EMF-92-116(P)(A) referenced methodologies is 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 5.0).

Table 7-1 lists all the sections of Reference 6 and shows which 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-10084P-A (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-10227P-A (Reference 2)
3.2.6 Cladding Rupture	No	NuScale methodology
3.2.7 Fuel Rod Mechanical Fracturing	No	To be addressed in a separate NuScale submittal.
3.2.8 Fuel Densification and Swelling	No	Use alternate method per BAW-10231P-A (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-10227P-A (Reference 2)
3.3.3 Fretting Wear	Yes	
3.3.4 Fuel Cladding Oxidation, Hydriding, and Crud Buildup	No	Use alternate method per BAW-10231P-A (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-10231P-A (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, Section 4.2, page 4.2-6.

The sections listed below from Reference 6 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 criteria and methods presented in Reference 6 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 Reference 6. Chapter numbers refer to the Chapter numbers in Reference 6.

Chapter 1 *Introduction*

This chapter provides the background and history of Reference 6, noting that the report contains fuel design criteria which provide assurance for adequacy of the design throughout the design life. Chapter 1 identifies no limitations on the use of Reference 6 for NuScale fuel 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 address each EMF-92-116(P)(A) criterion that is applied to the NuScale fuel assembly design as detailed in Table 7.1.

Section 3.2.1 Internal 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 Reference 6 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 14) cites the acceptable stress levels for fuel assembly components. Component stresses are calculated using 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 Reference 6 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 flow conditions that represent the reactor coolant system. 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 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. These growth analysis elements are consistent with the Reference 6 methodology. Therefore, the mechanical design critiria remain unchanged and the Reference 6 methodology applies to the NuScale fuel design.

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 mass values and predicted hydraulic forces are specific to the NuScale design. The flow rate (approximately 3.1 ft./sec) 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 mechanical constraints to resist lateral and axial accelerations. Therefore, the standard method applies and the Reference 6 mechanical design criterion is unchanged for the NuScale design.

Chapter 4 *Thermal and Hydraulic Design Criteria*

Chapter 4 is not applicable to the NuScale design.

Chapter 5 Nuclear Design Analysis

Chapter 5 is not applicable to the NuScale design.

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 the 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 Reference 6 for PWR licensing applications up to 62 GWD/mtU 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 Reference 6 being applied to the NuScale fuel design. There are no technical restrictions in any of these paragraphs that would limit the application of the Reference 6 generic mechanical design criteria to the NuScale design.

7.2.2 Technical Evaluation Report

In the Reference 6 TER, 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 an 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 Reference 6 are applicable to the NuScale design.

8.0 OVERALL SUMMARY AND CONCLUSIONS

This report provides justification for of the use of five NRC-approved AREVA codes and methods for evaluating performance of the NuScale fuel design in the design certification application. For application of the five cited AREVA topical reports to NuScale, one modification is required to conservatively predict NuScale fuel behavior. The creep collapse analysis should not be performed at an axial elevation of 90 inches due to the shorter active fuel length. Rather, the creep collapse calculation is performed at a composite axial location that combines the worst neutron flux anywhere along the rod with the peak cladding temperature to ensure a conservative prediction of cladding creep.

This modification is applied to the NuScale analysis only. With the NRC approval of this modification, the five 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.

9.0 REFERENCES

1. BAW-10084P-A-03, Revision 3, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse", August 1995.

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

3. BAW-10183P-A, Revision 0, "Fuel Rod Gas Pressure Criterion" (FRGPC), February, 1996.

4. BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code", January 2004.

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

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

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

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

9. 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.

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

11. B.B. Mikic, "Thermal Contact Conductance: Theoretical Considerations," Int. J. Heat Mass Transfer, Vol. 17, 1973.

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

13. 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.

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

Section C

March 8, 2017 Docket: PROJ0769

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Response to NRC Request for Additional Information Letter No. 12 for the Review of Topical Report TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1 (PROJ0769).

REFERENCES:

1. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, “NuScale Power, LLC Submittal of Topical Report TR-0116-20825, “Applicability of AREVA Fuel Methodology for the NuScale Design,” Revision 1 (NRC Project No. 0769),” dated July 1, 2016 (ML16187A017)
2. Letter from U.S. Nuclear Regulatory Commission to NuScale Power, LLC, “Request for Additional Information Letter No. 12 for the Review of Topical Report TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1 (PROJ0769),” dated February 10, 2017 (ML17044A021)

In a letter dated July 1, 2016 (Reference 1), NuScale Power, LLC (NuScale) submitted the topical report entitled “Applicability of AREVA Fuel Methodology for the NuScale Design,” Revision 1. In a letter dated February 10, 2017 (Reference 2), the NRC Staff provided a Request for Additional Information (RAI) regarding the subject of the topical report.

The purpose of this letter is to provide NuScale responses to the NRC RAIs.

Enclosure 1 is the proprietary version of the NuScale responses to RAI Letter No. 12. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale responses to RAI Letter No. 12.

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

Please feel free to contact Stephanie Seely at 980-349-4897 or at sseely@nuscalepower.com if you have any questions.

Sincerely,

Thomas A. Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC

Distribution: Frank Akstulewicz, NRC, TWFN-6C20
Greg Cranston, NRC, TWFN-6E55
Omid Tabatabai, NRC, TWFN- 6E55
Samuel Lee, NRC, TWFN-6C20
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Enclosure 1: NuScale Response to NRC Request for Additional Information Letter No. 12 for TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1, proprietary version

Enclosure 2: NuScale Response to NRC Request for Additional Information Letter No. 12 for TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1, nonproprietary version

Enclosure 3: Affidavit of Thomas A. Bergman, AF-0317-53213

Enclosure 2:

NuScale Response to NRC Request for Additional Information Letter No. 12 for TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1, nonproprietary version

NRC RAI Number: 8727 NRC RAI Date: February 10, 2017

NRC Review of: Applicability of AREVA Fuel Methodology for the NuScale Design, TR-0116-20825-P Revision 1

NRC RAI Question Number: 04.02-29594a

NRC RAI Question

Page 8 of TR-0116-20825-P Revision 1 provides design parameters for the NuScale fuel design. The staff notes that there are a few relevant parameters missing. Please provide the pellet dish and chamfer dimensions, plenum length, and plenum spring dimensions to allow staff to evaluate the void volume calculations.

NuScale RAI Question Response

Table 1, Table 2, and Table 3 provide the dimensions necessary to calculate the void volumes. Table 1 in concert with Figure 1 provides the pellet dimensions; the **[**
]ECI . Table 3 in concert with Figure 2 and Figure 3 provide the Fuel Plenum Spring dimensions. Table 2 provides additional useful information. These dimensions should be sufficient to calculate an approximate void volume.

Table 1. Pellet dimensions

Description	**Value**
Pellet diameter (A) (inch)	0.3195
Pellet height (B) (inch)	0.400
[	
	]ECI

Table 2. Additional fuel dimensions

Description	Value
[	
	]ECI
Fuel rod plenum length (inch)	5.311
Fuel rod inner diameter (inch)	0.326

Table 3. Fuel plenum spring dimensions

[

]ECI

[

]

Figure 1. Pellet dimensions

[

]ECI

Figure 2. Fuel rod plenum spring

[

]

Figure 3. Fuel rod plenum spring end detail

Impact of NRC RAI Question Response on TR-0116-20825:

This RAI response does not require revision to the report.

Attachments:

None.

NRC RAI Number: 8727 NRC RAI Date: February 10, 2017

NRC Review of: Applicability of AREVA Fuel Methodology for the NuScale Design, TR-0116-20825-P Revision 1

NRC RAI Question Number: 04.02-29594b

NRC RAI Question

Page 15 of TR-0116-20825-P Revision 1 discusses a clad creep analysis and states that "a revised **[**

] ". It is unclear to the staff if this is a new methodology or is part of a previously approved methodology.

Provide a more detailed description of the revised approach used, whether it is a revised approach to a previously approved methodology or a new methodology, and the intent of its use. If this revised approach is being used for more than additional justification for the CROV methodology, then update the topical report to include this clarification.

NuScale RAI Question Response

The description on Page 15 of TR-0116-20825-P Revision 1 is not a new methodology. TR-0116-20825-P Revision 1, Section 3.3.1 describes a NuScale-specific adjustment to the approved methodology based on the unique characteristics of the NuScale design. The key characteristic of the NuScale design that differs from the design described in the approved methodology is the fuel column length.

As required by the approved methodology, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," BAW-10084P-A, Revision 3 (Reference 1), the creep collapse analysis is performed **[**

] that results in the shortest calculated collapse time. For a 12 foot fuel column, the axial location has been determined to be at the 90 inch elevation . The NuScale fuel column is only 78.74 inches long so the 90 inch location noted in the approved methodology cannot be used. For NuScale, the **[**

] used for the creep collapse analysis are conservative.

Due to the high initial fuel pellet density (96 percent theoretical density) of the NuScale design, the fuel pellets will experience low densification. The **[** **]** gap specified in the approved method was based in part on the **[**

] . Given the low densification and shorter fuel column length, the resulting calculated axial gap for the NuScale design is **[** **]** noted in the approved methodology. As opposed to changing the approved methodology by justifying a different axial gap, the existing **[ ]** is conservatively applied.

<u>Impact of NRC RAI Question Response on TR-0116-20825:</u>

This RAI response does not require revision to the report.

<u>References:</u>

1. BAW-10084P-A-03, Revision 3, “Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse,” August 1995.

<u>Attachments:</u>

None.

NRC RAI Number: 8727 NRC RAI Date: February 10, 2017

NRC Review of: Applicability of AREVA Fuel Methodology for the NuScale Design, TR-0116-20825-P Revision 1

NRC RAI Question Number: 04.02-29594c

NRC RAI Question

RAI Section 5.2.1.1 of TR 0116-20825 Revision 1 discusses the heat transfer model used by COPERNIC for the coolant-cladding interface and the applicability of this model to the NuScale design. It states that the **[**

].

Is two-phase flow anticipated in the NuScale reactor during normal operation or AOOs.

NuScale RAI Question Response

The NuScale Power plant is designed to operate with single-phase flow in the primary coolant system. The control system operates to maintain the core average outlet temperature at a maximum of 590 degrees F. The saturation temperature at the operating pressure of 1850 psia is 625 degrees F. NuScale employs a hot leg temperature analytical limit of 610 degrees F to ensure that subcooled margin is maintained.

Under full power steady state operation, the maximum expected equilibrium quality in the limiting core subchannel is {{ }}2(a),(c) (subchannel void fraction is 4 percent). With conservative assumptions for flow, flow distribution, and instrumentation error, and application of conservative hot channel factors to the limiting subchannel, the maximum predicted equilibrium quality is less than {{ }}2(a),(c) . The most limiting anticipated operational occurrence (AOO) in regards to thermal margin is Uncontrolled Control Rod Assembly Withdrawal from Power (FSAR Section 15.4.2); the limiting subchannel for this event is predicted to have a quality of approximately {{ }}2(a),(c) .

Just as in large pressurized water reactors (PWRs), some two-phase flow is anticipated in the limiting subchannel of the NuScale core. Thus, the application of the **[** **]** for calculation of cladding temperatures under subcooled boiling conditions is appropriate for the NuScale operating conditions and consistent with PWR experience.

Impact of NRC RAI Question Response on TR-0116-20825:

This RAI response does not require revision to the report.

Attachments:

None.

NRC RAI Number: 8727 NRC RAI Date: February 10, 2017

NRC Review of: Applicability of AREVA Fuel Methodology for the NuScale Design, TR-0116-20825-P Revision 1

NRC RAI Question Number: 04.02-29594d

NRC RAI Question

Page 23 of TR-0116-20825 Revision 1 discusses the applicability of the **[ ]** and Dittus-Boelter correlations to the NuScale reactor with natural circulation based on the Reynold's number being greater than that where forced convection has been seen.

Have any tests or analyses been performed to demonstrate that either of these correlations would be valid under typical conditions for the NuScale reactor (i.e. power, coolant temperature, and flow rate)? If so, provide a reference to, or summary of, the tests or analyses to minimize the need for future RAIs during the topical report review.

NuScale RAI Question Response

Pressure drop and critical heat flux (CHF) tests have been performed for the fuel design under typical NuScale reactor operating conditions. No tests have been performed specifically for single-phase heat transfer. However, pressure drop tests were performed for NuScale applicable Reynold's number ranges {{ }}$^{2(a),(c)}$ in the derivation of grid spacer and rod friction loss coefficients. The purpose of the pressure drop tests was to determine form loss coefficients for typical flow conditions for the NuScale reactor. Critical heat flux test data utilized in the NSP2 correlation (TR-0116-21012, Revision 0) development for the NuScale fuel design covers Reynold's number ranges of approximately {{ }}$^{2(a),(c)}$.

The Dittus-Boelter correlation is used extensively throughout the nuclear industry for turbulent forced convection flow in the single-phase liquid heat transfer regime. The use of the **[ ]** correlation, with its comparison to the Dittus-Boelter correlation in this topical report, is consistent with the correlations used in the NuScale Subchannel Methodology, TR-0915-17564. TR-0915-17564, Section 5.5 discusses the thermal margin to CHF calculations that employ the "EPRI" single-phase heat transfer correlation. This selection is a code requirement to remain consistent with two-phase flow models used within the code; however, the coefficients used in the EPRI correlation are those of the Dittus-Boelter correlation.

The Dittus-Boelter correlation is valid for ranges of Reynolds numbers greater than 10,000. With conservative assumptions consistent with the methodology in TR-0915-17564, the minimum Reynold's number at 5 percent rated power is {{ }}$^{2(a),(c)}$ with a maximum of {{ }}$^{2(a),(c)}$ at 100 percent rated power. Thus, the ranges in Reynold's numbers exhibited by typical NuScale conditions indicate that the Dittus-Boelter correlation is applicable.

Impact of NRC RAI Question Response on TR-0116-20825:

This RAI Response does not require report revisions.

Attachments:

None.

Section D

July 01, 2016

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Topical Report TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1 (NRC Project No. 0769)

REFERENCES: Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "NuScale Power, LLC Request for Suspension of Acceptance Review of TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 0," LO-0616-49551, dated June 3, 2016 (ML16155A449).

In the referenced letter dated June 3, 2016, NuScale Power, LLC (NuScale) requested the NRC to suspend its acceptance review of TR-0116-20285, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 0. NuScale provided a schedule indicating the intent to submit this Revision 1 of this topical report incorporating comments received during the NRC's acceptance review by June 30, 2016. Consistent with that schedule, NuScale hereby submits Topical Report TR-0116-20825, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1.

The following changes are incorporated into TR-0016-20285, Revision 1:

- Deletion of discussions concerning seismic methodology applicability, along with associated Appendices 1 through 3;
- Deletion of discussions concerning damping factors;
- Deletion of discussions of adjustments made to fuel growth analyses to account for the absence of a hold-down spring

In addition to the above, TR-0016-20285, Revision 1 includes changes to designations of the NuScale Proprietary and Export Controlled Information, along with minor changes.

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.

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

Sincerely,

Thomas A. Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC

Distribution: Frank Akstulewicz, NRC, TWFN-6C20
Greg Cranston, NRC, TWFN-6E7
Omid Tabatabai, NRC, TWFN-6E7
Mark Tonacci, NRC, TWFN-6E7

Enclosure 1: "Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-P, Revision 1, proprietary version
Enclosure 2: "Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-NP, Revision 1, nonproprietary version
Enclosure 3: AREVA Affidavit
Enclosure 4: NuScale Affidavit, AF-0616-50037

Enclosure 2:

"Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-NP, Revision 1, nonproprietary version

NOTE: This enclosure to NuScale's July 1, 2016 letter to the NRC is identical to the topical report included in Section B of the current NuScale letter with two exceptions: the Section B version includes "-A" in the document identification number and brackets were removed from Pages 2, 15, and 46 because information previously marked proprietary was subsequently determined to be nonproprietary. Therefore, this enclosure is not included in the current package.

Enclosure 3:

Affidavit of Thomas A. Bergman, AF-0118-58146

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 NuScale fuel design and associated analytical methods.

NuScale has performed significant research and evaluation to develop a basis for this design 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 topical report TR-0116-20825-P-A entitled "Applicability of AREVA Fuel Methodology for the NuScale Design." 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 be 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 February 9, 2018.

Thomas A. Bergman

Enclosure 4:

Affidavit of Nathan E. Hottle

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 Framatome Inc. (Framatome) and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the Framatome information contained in the following document: TR-0116-20825-P-A Revision 1, "Applicability of AREVA Fuel Methodology for the NuScale Design," referred to herein as "Document." Information contained in Sections A, B, and C of this document has been classified by Framatome as proprietary in accordance with the policies established by Framatome 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 Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

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

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

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

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

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

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

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

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

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

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

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

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

SUBSCRIBED before me this 12th

day of January, 2018.

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