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 <u>BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code"</u>

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

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

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)