NuScale Power

January 7, 2017

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of NuScale Technical Report "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," TR-0816-51127 (NRC Project No. 0769).

REFERENCES: 1. NuScale Letter to the NRC, "NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application," dated December 31, 2016.

NuScale Power, LLC (NuScale) has submitted a Design Certification Application for its Integral Small Modular Reactor design (Reference 1). The purpose of this letter is to provide NuScale Technical Report "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," TR-0816-51127. This technical report is referenced in the NuScale Final Safety Analysis Report and provides supplementary information, data and analyses.

Enclosure 1 contains the proprietary version of the NuScale Technical Report "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," TR-0816-51127. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from public disclosure per the requirements of 10 CFR § 810. Enclosure 4 pertains to the AREVA proprietary information to be withheld from the public while Enclosure 3 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., "{{ }}") in the report. Enclosure 2 contains the non-proprietary version of NuScale Technical Report "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," TR-0816-51127.

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

Please feel free to contact Zackary Rad, Director, Regulatory Affairs at 980.349.4831 or at zrad@nuscalepower.com if you have any questions.

Sincerely,

Thomas A. Bergman
Vice President, Regulatory Affairs

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Enclosure 1: NuFuel-HTP2™ Fuel and Control Rod Assembly Designs, TR-0816-51127-P, Revision 1, proprietary version
Enclosure 2: NuFuel-HTP2™ Fuel and Control Rod Assembly Designs, TR-0816-51127-NP, Revision 1, nonproprietary version
Enclosure 3: Affidavit of Thomas A. Bergman, AF-0816-51127
Enclosure 4: Affidavit of Nathan E. Hottle, AREVA, Inc.

Enclosure 1:

NuFuel-HTP2™ Fuel and Control Rod Assembly Designs, TR-0816-51127-P, Revision 1, proprietary version

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Enclosure 2:

NuFuel-HTP2™ Fuel and Control Rod Assembly Designs, TR-0816-51127-NP, Revision 1, nonproprietary version

NuFuel-HTP2™ Fuel and Control Rod Assembly Designs

January 2017

Revision 1

Docket: PROJ0769

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Abstract

This technical report provides a description of the NuScale fuel assembly (NuFuel-HTP2™) and control rod assembly (CRA) designs. The report identifies the applicable regulatory requirements and summarizes the analyses and tests performed to demonstrate that the designs meet the regulatory requirements. The evaluations are focused on the mechanical aspects of the designs, consistent with Section 4.2 of NUREG-0800. Neutronic and thermal-hydraulic performance of the designs are addressed in other parts of the NuScale design certification application.

Executive Summary

The purpose of this report is to describe the NuFuel-HTP2™ fuel and control rod assembly (CRA) design, summarize the key analysis results and evaluate the fuel and CRA design performance against regulatory requirements.

The fuel is designed and analyzed in accordance with regulatory requirements in NUREG-0800 Section 4.2 and the regulatory requirements from GDC 2, GDC 10, PDC 27, PDC 35, and 10 CFR 50.46.

Fuel Assembly

The NuScale fuel design is a reduced height 17x17 pressurized water reactor design. The assembly contains five spacer grids, 24 MONOBLOC guide tubes and a top and bottom nozzle. The bottom nozzle contains a mesh debris filter. The top nozzle is removable to allow for reconstitution of fuel rods if needed. The top four grids provide structural support for the fuel assembly and enhance mixing of the coolant. The bottom grid is primarily for structural support. The fuel rod cladding is M5®, an advanced zirconium alloy. The fuel is UO_2 enriched up to 4.95 weight percent ^{235}U, with gadolinia homogeneously mixed in some fuel pellets.

The fuel assembly is analyzed using established design criteria and methods to demonstrate that the fuel assembly is not damaged during normal operations, anticipated operational occurrences, and postulated accidents. Potential failure mechanisms include mechanical loading, fatigue, fretting, oxidation, growth, and distortion. The NuScale criteria and methods have been applied to fuel assembly designs currently in power operation and have been demonstrated to be applicable to the NuScale fuel assembly design. The evaluations presented in this report demonstrate that the fuel assembly design meets all design criteria.

The fuel rod is analyzed using established design criteria and methods to demonstrate that the fuel rod is not damaged during normal operation, anticipated operational occurrences, and postulated accidents. Potential failure mechanisms include internal pressure, internal hydriding, creep collapse, centerline melting, pellet/cladding interaction, and mechanical loading. Certain design criteria (e.g., those associated with loss-of-coolant accidents) are addressed in other portions of the NuScale design certification application and not in this report. The NuScale criteria and methods have been applied to fuel rod designs currently in power operation and have been demonstrated to be applicable to the NuScale design. The evaluations presented in this report demonstrate that the fuel rod design meets all design criteria.

Mechanical testing is performed to characterize the mechanical performance of the fuel assembly. The test results support the design evaluations presented in this report. Thermal-hydraulic evaluations and tests are performed to characterize the hydraulic performance of the fuel assembly and demonstrate that the fuel design meets all applicable design criteria.

The evaluations and tests performed are comprehensive and demonstrate the acceptability of the fuel assembly design for use in the NuScale Power Module.

Control Rod Assembly

The NuScale control rod assembly (CRA) contains 24 individual control rods fastened to a stainless steel spider hub. The control rod tubes are stainless steel and contain silver indium cadmium (Ag-In-Cd) and boron carbide (B_4C) neutron absorbers.

The CRA design is analyzed using established design criteria and methods to demonstrate acceptable performance over the design lifetime. Potential failure mechanisms include stresses and loads, strain, creep collapse, fatigue, wear, internal pressure, and component melting. The evaluations summarized in this report demonstrate that the CRA design meets all criteria and is acceptable for use in the NuScale Power Module.

A CRA testing program is defined to confirm CRA drop times and CRA drop velocity, to assess the propensity for vibration wear, and to ensure that CRA insertion is not adversely affected by the maximum expected fuel assembly distortion and the misalignment of the lead screw and guides.

1.0 Introduction

NuScale has developed a 17X17 fuel assembly (NuFuel-HTP2™) and control rod assembly (CRA) design for use in the NuScale Power Module based on existing AREVA methods and technology. This design incorporates the full range of AREVA 17X17 operating experience for application to the NuScale operating environment.

1.1 Purpose

The purpose of this report is to describe the NuFuel-HTP2™ fuel assembly and CRA designs, summarize the key analysis results and demonstrate that the designs meet regulatory requirements.

1.2 Scope

NuScale used NRC approved AREVA methodologies to perform the analysis of the NuFuel-HTP2™ fuel design. This report evaluates the NuFuel-HTP2™ fuel design against the requirements of NUREG-0800 Section 4.2 (Reference 9.1.10). The evaluations are focused on the mechanical aspects of the design. Neutronic and thermal hydraulic performance of the fuel assembly are addressed in other parts of the NuScale design certification application.

This report also summarizes the CRA design and analyses.

Chapter 2 provides the regulatory framework that identifies the requirements against which the fuel design is evaluated.

Chapter 3 describes the mechanical design and function of the fuel assembly and provides a brief description of applicable operating experience.

Chapter 4 summarizes the results of the fuel assembly design evalutions using NRC approved methods.

- The NuFuel-HTP2™ fuel mechanical design is evaluated using generic mechanical design criteria for pressurized water reactor (PWR) fuel designs (Reference 9.1.6 and Reference 9.1.9).

- The COPERNIC Fuel Rod Design Computer Code (Reference 9.1.1) is used in the fuel performance analysis of the NuFuel-HTP2™ fuel design.

- The CROV code (Reference 9.1.2) is used to evaluate the cladding creep performance.

- The computational procedure for evaluating fuel rod bowing (Reference 9.1.4) is used to evaluate the impacts of fuel rod bow for the NuFuel-HTP2™ design.

- The PWR fuel assembly structural response to externally applied dynamic excitations (Reference 9.1.5) is used to perform seismic structural analysis on the fuel design.

The applicability of the fuel analysis methods to the NuScale design is demonstrated in *Applicability of AREVA Fuel Methodology for the NuScale Design*, TR-0116-20825-P Revision 1 (Reference 9.1.7) and *NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces*, TR-0716-50351-P (Reference 9.1.8).

Chapter 5 describes the comprehensive testing performed to support the fuel assembly design and analysis.

Chapter 6 describes the CRA design and the analyses and testing that support the design.

Chapter 7 provides guidance for managing design changes to the fuel and CRA design. Criteria are provided to distinguish changes that require NRC approval.

Chapter 8 provides a summary and concludes that the NuFuel-HTP2™ fuel and NuScale CRA design satisfies the applicable design requirements.

Table 1-1 Abbreviations

Term	Definition
AIC	silver-indium-cadmium
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BOL	beginning of life
CFM	centerline fuel melt
CFR	Code of Federal Regulations
CRA	control rod assembly
CUF	cumulative usage factor
EFPY	effective full power year
EOL	end of life
FIV	flow-induced vibration
FSAR	final safety analysis report
GDC	general design criteria
LHR	linear heat rate
LOCA	loss-of-coolant accident
NPM	NuScale power module
NRC	Nuclear Regulatory Commission
OD	outside diameter
PDC	principal design criterion
PHTF	portable hydraulic test facility (Richland)
PWR	pressurized water reactor
QD	quick disconnect
RCCA	rod cluster control assembly
RCS	reactor coolant system
RMS	root mean square
SRSS	square root of the sum of the squares
SSE	safe shutdown earthquake
TCS	transient cladding strain

2.0 Background

This report supports final safety analysis report (FSAR) Section 4.2 of the NuScale design certification application. This section identifies the regulatory requirements that form the basis of the fuel design evaluation.

2.1 Regulatory Requirements

The mechanical design of the NuFuel-HTP2™ fuel assembly is evaluated against criteria established to be consistent with NUREG-0800 Section 4.2 (Reference 9.1.10). These criteria are specified in Section 4.0 of this report. Some of the NUREG-0800 acceptance criteria are addressed in plant transient analyses and are not addressed by this report, as noted in Section 4.0.

The following regulatory requirements apply to the fuel mechanical evaluations summarized in this report:

- GDC 2, as it relates to designing fuel assemblies to withstand the effects of earthquakes.
- GDC 10, as it relates to assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOO).
- 10 CFR 50.46, as it relates to ensuring that fuel assembly and fuel rod damage will not interfere with effective emergency core cooling.

Additionally, the following principal design criteria specified in FSAR Section 3.1 of the NuScale design certification application apply to the fuel mechanical evaluations summarized in this report:

- PDC 27, as it relates to ensuring that fuel system damage is never so severe as to prevent control rod insertion when it is required.
- PDC 35, as it relates to ensuring fuel damage does not interfere with effective emergency core cooling.

Section 4.0 identifies the approved methodologies used for evaluation of each criterion. These methodologies are demonstrated to be applicable to the NuScale design in *Applicability of AREVA Fuel Methodology for the NuScale Design*, TR-0116-20825-P Revision 1 (Reference 9.1.7) and *NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces*, TR-0716-50351-P (Reference 9.1.8).

3.0 NuFuel-HTP2™ Fuel Assembly Description

The NuFuel-HTP2™ fuel assembly, shown in Figure 3-1, is a 17x17 pressurized water reactor (PWR) design that is approximately one-half the length of typical PWR nuclear plant fuel. Other than the shortened length, the assembly contains design features similar to those of proven HTP™ fuel designs. All components of the fuel assembly have relevant operating experience that demonstrates their suitability for use in reactor cores. The assembly is supported by five spacer grids, 24 guide tubes, and a top and bottom nozzle that together provide the structural skeleton for the 264 fuel rods. The fuel rod consists of M5® alloy cladding and uranium dioxide (UO_2) pellets with gadolinium oxide (Gd_2O_3) as a burnable absorber homogeneously mixed within the fuel pellets in select rod locations. Table 3-1 and Table 3-2 list key fuel design parameters.

Table 3-3 provides representative operating conditions for the NuScale Power Module. The NuScale core design is comprised of 37 fuel assemblies. Sixteen of the fuel assembly locations contain CRAs. Figure 3-2 provides a representative core loading pattern showing the arrangement of the fuel assemblies in the reactor core.

The total, nominal height of the fuel assembly is 94 inches (excluding the hold-down spring height). Due to the assembly 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 that provide lateral support for the fuel rods. Four HTP™ grids at the intermediate and top spacer locations are welded to the guide tubes, while the HMP™ lower grid at the bottom location of the fuel assembly is captured by rings welded to the guide tubes.

The fuel assembly materials (Table 3-4) are chosen for their low cobalt content to reduce plant dose, for corrosion resistance, and for desirable structural properties.

A summary of the NuFuel-HTP2™ components is provided below.

3.1 Top Nozzle

The top nozzle, shown in Figure 3-3, consists of a 304L stainless steel frame that interfaces with the reactor upper internals and the core components while providing for reactor coolant flow. The top nozzle flow hole pattern provides low pressure drop while satisfying strength requirements. A through-hole feature allows for insertion of the in-core instrumentation from above the fuel assembly.

The top nozzle is attached to the fuel assembly with quick disconnect (QD) features at each of the 24 guide tube locations. The QD features allow for removal of the top nozzle for fuel assembly reconstitution.

Two diagonally opposed corners of the top nozzle contain holes for accommodating the upper core plate alignment pins. Mounted on the other two corners are four two-leaf hold-down springs. The spring leaves are made of Alloy 718 and are designed to maintain positive hold-down margin. The leaf spring sets are fastened to the top nozzle with Alloy 718 clamp screws. The upper leaf has an extended tang that engages a cutout in the top plate of the nozzle to retain the spring leaves in the unlikely event of a leaf or clamp screw failure.

On one of the corners with leaf springs is a through-hole that allows for identification of rotational orientation of the assembly from above. Additionally, on the top left side of each face of the nozzle are marks that allow for identification of the orientation of the assembly from the side.

3.2 Bottom Nozzle with Mesh Filter Plate

The 304L stainless steel bottom nozzle, shown in Figure 3-4, consists of a cast frame of ribs. Twenty-four holes allow for connecting the guide tubes to the nozzle using cap screws and a center hole is provided for the instrument tube. A high-strength A-286 alloy mesh filter plate is pinned to the top of the frame to provide debris resistance and is captured by the guide tubes and cap screws. The four corners have concave feet with a radius designed to interface with the alignment pins of the lower core plate. As with the top nozzle, the top left side of each face of the bottom nozzle contains marks to allow identification of assembly orientation from the side.

3.3 Zircaloy-4 MONOBLOC™ Guide Tubes

The MONOBLOC™ guide tubes, shown in Figure 3-5, have a constant outer diameter. The upper portion of the guide tube has a large internal diameter that allows for rapid insertion of the CRA during a reactor trip. The lower portion of the guide tube has a reduced inner diameter that acts as a dashpot that decelerates the CRA to limit the impact forces on the fuel assembly during a reactor trip. The guide tube has four holes located just above the top of the dashpot to allow for cooling flow for inserted CRAs and outflow during a reactor trip. The outside diameter of the guide tube is constant. The added thickness in the dashpot of the MONOBLOC™ guide tube increases the lateral stiffness of the fuel assembly and inhibits fuel assembly distortion and bow.

The guide tube lower end plug, shown in Figure 3-6, is threaded to accept a stainless steel cap screw to secure the guide tube to the bottom nozzle. The cap screw has a through-hole that allows for cooling flow to the guide tube as well as outflow during a reactor trip. The design of the dashpot and cap screw hole are consistent with existing PWR designs.

The guide tube is connected to the top nozzle with a QD assembly, Figure 3-7. The QD consists of a double-spline sleeve made of Zircaloy-4 attached to the guide tube with multiple spot welds. Machined keyway-type features within the guide tube attachment holes in the top nozzle provide either clearance for removal or restraint for securing the nozzle, based on the radial orientation of the QD features.

3.4 Zircaloy-4 Instrument Tube

The Zircaloy-4 instrument tube has constant inner and outer diameters and is located at the center of the 17x17 array. It provides guidance for the in-core instrumentation, which is inserted from the top of the fuel assembly. The instrument tube is not attached to either of the fuel assembly nozzles, but has its axial position fixed by sleeves welded above and below the bottom HMP™ grid.

3.5 Zircaloy-4 HTP™ Upper and Intermediate Spacer Grids

The four HTP™ spacer grids, shown in Figure 3-8 and Figure 3-9, that occupy the top four grid positions are formed from interlocking Zircaloy-4 strips that are welded at all intersections to form a 17x17 matrix of square cells. Each grid strip includes a pair of strips welded back-to-back to create a doublet. The doublet is formed with flow channels that are angled at the outlets to create a swirling flow pattern. The flow channels are arranged so that there is no net hydraulic torque on the fuel assembly. The shape of the flow channel creates line contacts with the fuel rod that provide increased resistance to grid-to-rod fretting relative to traditional point-contact spacer grid designs. Sideplates are welded to the grid of doublets to complete the spacer grid design. The sideplates include lead-in tabs to eliminate hang-up during fuel movement.

The HTP™ spacer grids are spot welded to the guide tubes to limit axial movement and maintain alignment with adjacent fuel assemblies.

The HTP™ grids on the NuScale design are identical to those used on AREVA's 17x17 PWR product that has extensive operating experience in the United States.

3.6 Alloy 718 HMP™ Lower Spacer Grid

The HMP™ spacer grid, shown in Figure 3-10, resembles the HTP™ spacer grid with respect to spring design, rod-to-grid surface contact and manufacturing. The HMP™ spacer grid is made from low cobalt, precipitation-hardened Alloy 718 strip material, which provides enhanced strength and relaxation characteristics. The higher strength alloy allows thinner grid strips, resulting in reduced hydraulic resistance. The doublet flow channels are straight (non-mixing) flow channels that provide added reduction in hydraulic resistance. The grids are captured above and below by Zircaloy-4 sleeves spot welded to the guide tubes to maintain axial alignment.

The HMP™ grid on the NuFuel-HTP2™ design is identical to AREVA's 17x17 PWR product that has extensive operating experience in the United States.

3.7 Fuel Rod with Alloy M5® Fuel Rod Cladding

The fuel rod design, shown in Figure 3-11, consists of ceramic UO_2 pellets contained in seamless M5® zirconium alloy tubing with end caps welded at each end. The M5® cladding material significantly improves the resistance to corrosion compared to other cladding materials.

M5® cladding material was first inserted in a U.S. reactor core in 1995. Twenty-two U.S. reactors have used M5® alloy in more than 7500 fuel assemblies. Globally, over 5 million M5® fuel rods have operated in more than 21,000 fuel assemblies in 84 reactors. The operational experience of M5® cladding covers PWR fuel arrays from 14x14 up to 18x18.

The fuel stack height is 78.74 inches and rests on top of the lower end cap. The fuel rod has an internal stainless steel spring in the upper plenum 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 void volume of the fuel rod is designed to accommodate fission gas generation during operation to maintain rod internal pressure less than system pressure.

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.

The fuel pellet has chamfered edges and is dished on the top and bottom. The chamfers allow for ease of loading and reduce pellet chipping. The dishing and chamfers accommodate pellet swelling during operation and reduce the tendency to produce an hour-glass shape, reducing pellet-to-cladding stress concentrations and the potential for pellet stack gaps.

The design density of the UO_2 pellets is 96 percent theoretical density with a possible enrichment up to 4.95 weight percent ^{235}U consistent with current operating plant licensing requirements. The fuel rod design can utilize axial blanket and Gd_2O_3 fuel configurations.

Table 3-1 NuScale fuel assembly parameters

Parameter	Value
Fuel rod array	17 x 17
Fuel rods per assembly	264
Guide tubes per assembly	24
Instrument tubes per assembly	1
Spacer grids per assembly	5
Fuel assembly height without holddown spring (inch)	94.0
Fuel rod pitch (inch)	0.496
Guide tube outside diameter (inch)	0.482
Guide tube inside diameter – upper region (inch)	0.450
Guide tube inside diameter – dashpot region (inch)	0.397
Instrument tube outside diameter (inch)	0.482
Instrument tube inside diameter (inch)	0.450
HTP™ outer/inner strip height (inch)	1.950 / 1.750
HTP™ outer/inner strip thickness (inch)	**[ ]ECI**
HMP™ outer/inner strip height (inch)	1.950 / 1.750
HMP™ outer/inner strip thickness (inch)	**[ ]ECI**

* Single strip thickness of a welded doublet

Table 3-2 NuScale fuel rod parameters

Parameter	Value
Cladding material	M5®
Fuel rod length (inch)	85.0
Length of total active fuel stack (inch)	78.74
Cladding outer diameter (inch)	0.374
Cladding inner diameter (inch)	0.326
Cladding inner surface roughness (μin)	45
Fuel rod internal pressure (psig)	215
Fuel rod fill gas	Helium
Fuel rod plenum height (inch)	5.31
Fuel pellet outer diameter (inch)	0.3195
Fuel pellet length (inch)	0.400
Fuel pellet surface roughness (μin)	**[ ]**^{ECI}
Fuel pellet density (% TD)	96
Resinter densification limits (24-hour test)	**[ ]**^{ECI} **[ ]**^{ECI}
Fuel pellet grain size (μm)	**[ ]**^{ECI}
Fuel pellet open porosity fraction	**[ ]**^{ECI}
Sorbed gas	**[ ]**^{ECI} **[ ]**^{ECI}
Fuel pellet dish volume, nominal/percent (mm^3)	**[ ]**^{ECI}
Fuel pellet void volume, nominal/percent (cm^3)	**[ ]**^{ECI}
Plenum spring free length (inch)	**[ ]**^{ECI}
Plenum spring outer diameter (inch)	**[ ]**^{ECI}
Plenum spring wire diameter (inch)	**[ ]**^{ECI}
Plenum spring active coils	**[ ]**^{ECI}
Plenum spring volume (in^3)	**[ ]**^{ECI}
Lower end cap height (inch)	0.575

TD = theoretical density
UTL = upper tolerance limit
LCL = Lower confidence limit
UCL = Upper confidence limit

Table 3-3 NuScale operating conditions

Parameter	NuScale Design Value	AREVA 17x17 PWR Value
Rated thermal power (MWt)	160	3455
Average coolant velocity (ft/s)	2.7	16
System pressure (psia)	1850	2280
Core tave (°F)	543	584
Core avg linear heat rate, approx. (kW/m)	8.2	18.0
Reactor coolant system (RCS) inlet temperature (°F)	497	547
RCS Reynolds number	76,000	468,000
Fuel assemblies in core	37	193
Fuel assembly loading (kgU)	249	455
Core loading (kgU)	9,213	87,815

Table 3-4 Fuel assembly materials

Component	Material
Top nozzle	AISI 304L stainless steel
Bottom nozzle frame	AISI 304L stainless steel
Mesh filter plate	Alloy 286
Guide tube and QD sleeves	Zr-4
Holddown leaf springs	Alloy 718
Holddown spring clamp screw	Alloy 718
Top connection (quick disconnect)	Zr-4 and Alloy 718
Bottom cap screw	AISI 316L stainless steel
HMP™ grid	Alloy 718
HTP™ grid	Zr-4
Fuel rod cladding	M5® - cold worked and recrystallized zirconium alloy
Fuel rod plenum springs	AISI 302 stainless steel
Fuel pellets	UO_2 and UO_2 plus Gd_2O_3

Note: Stainless steels are low cobalt.

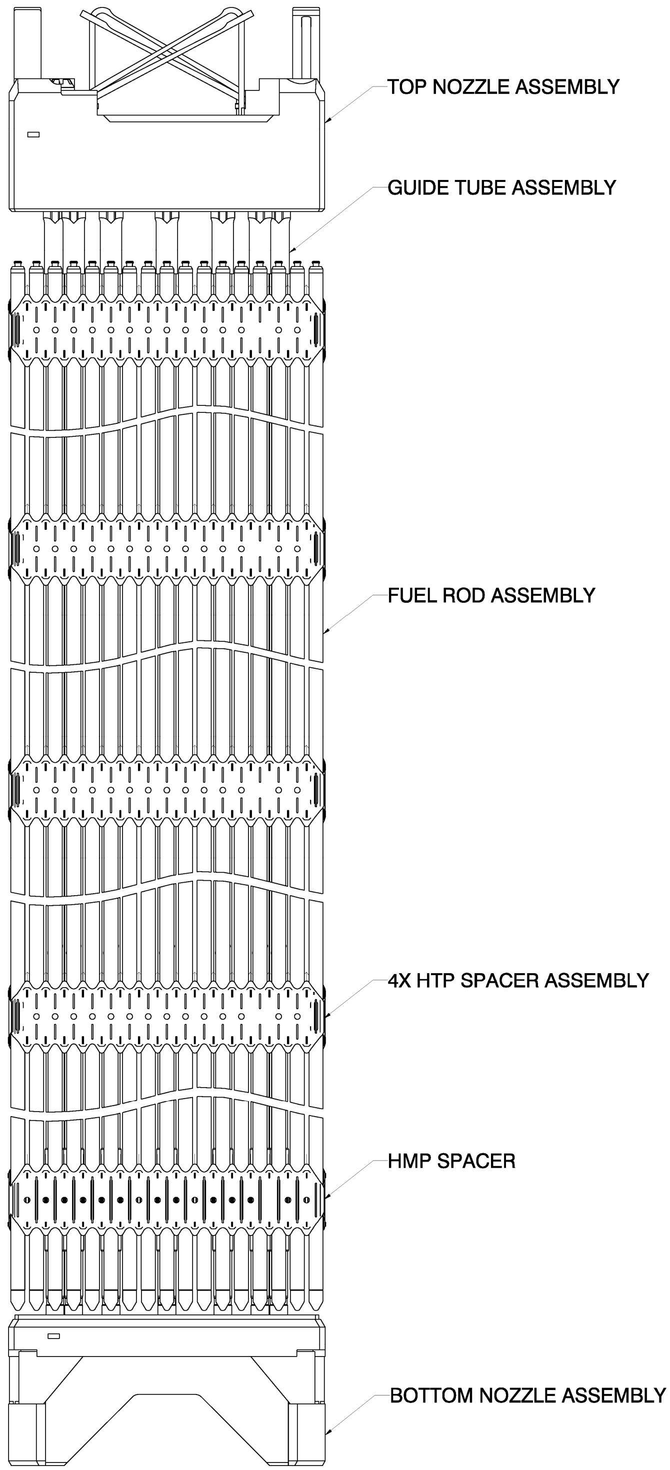

Figure 3-1   Fuel assembly general arrangement

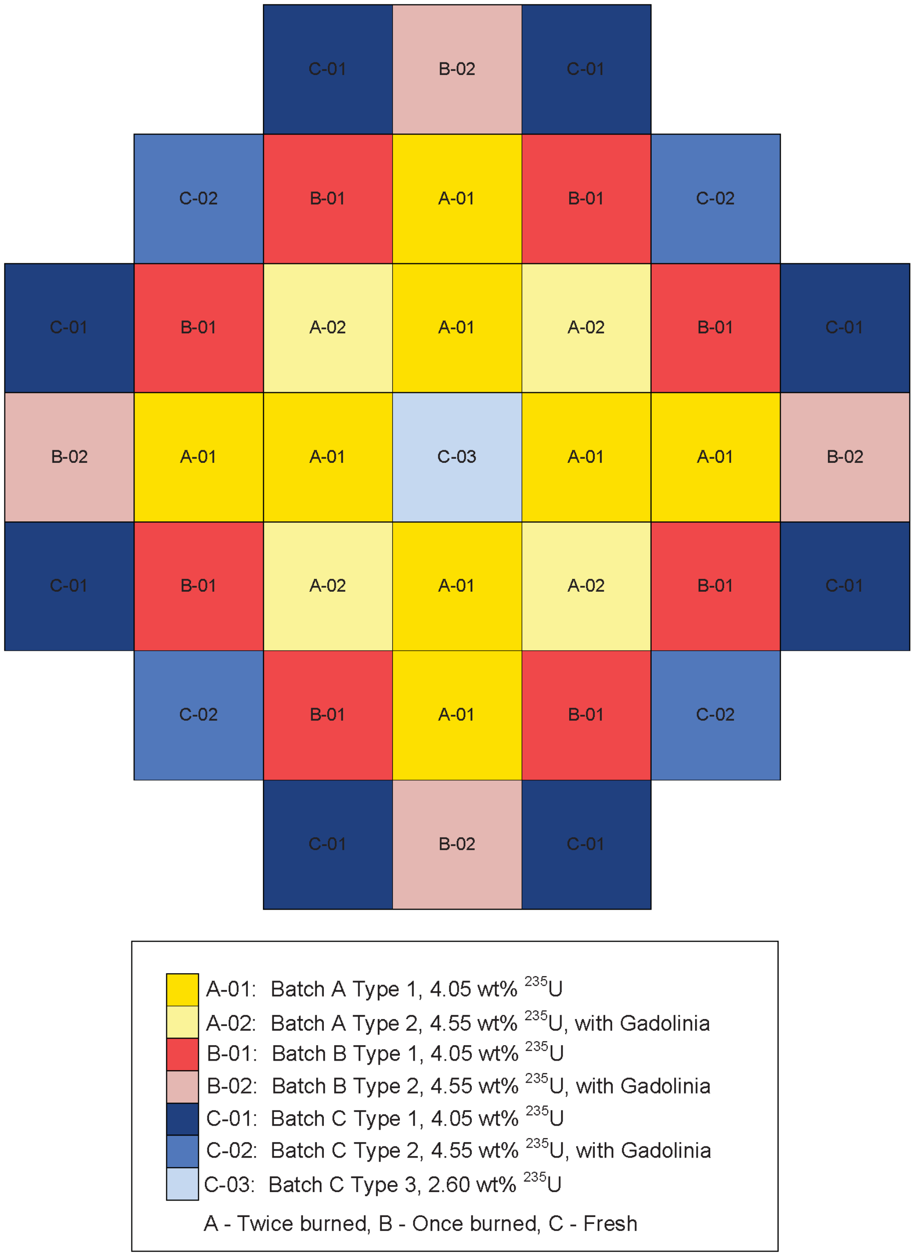

Figure 3-2 Representative core loading pattern

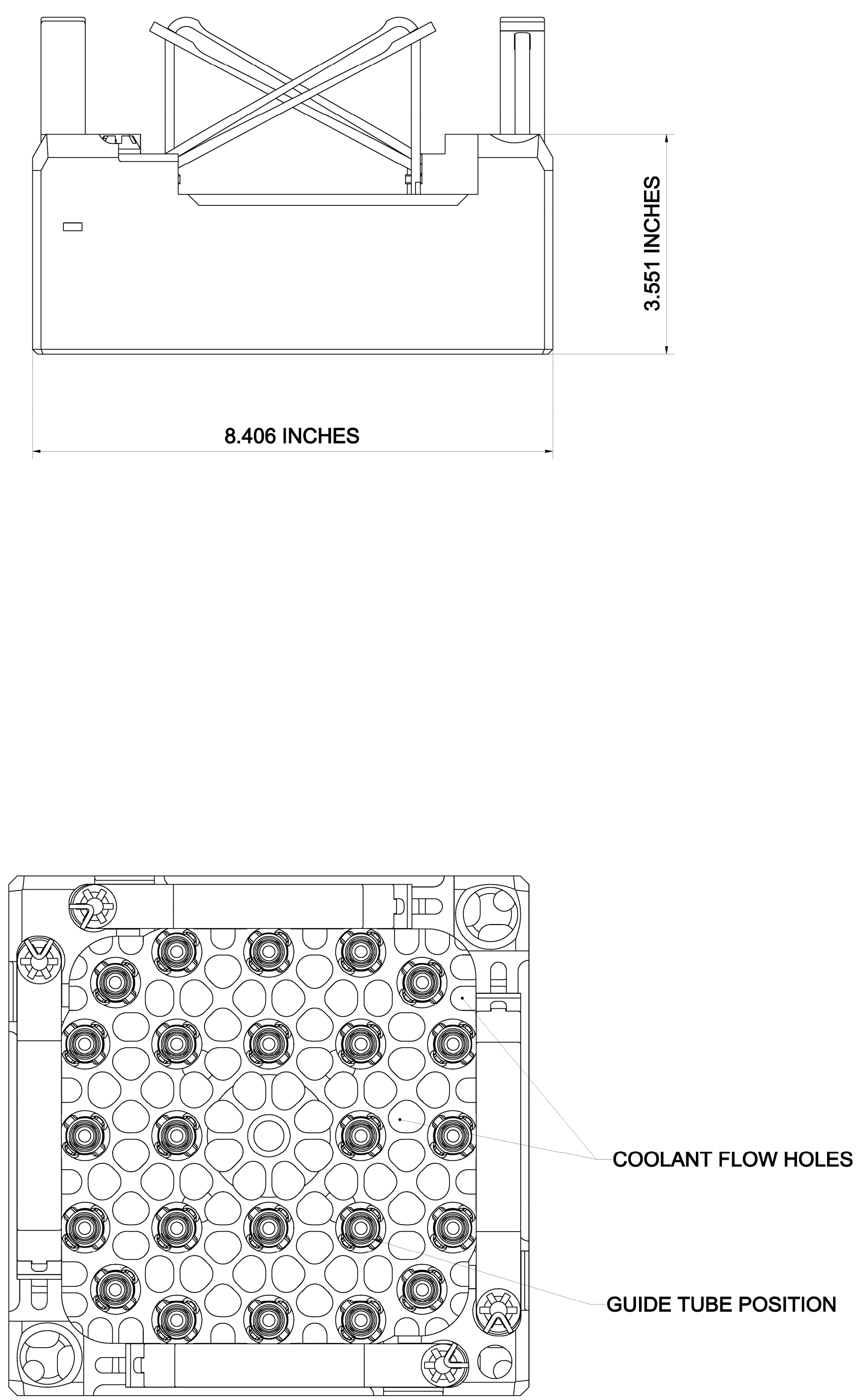

Figure 3-3 Top nozzle

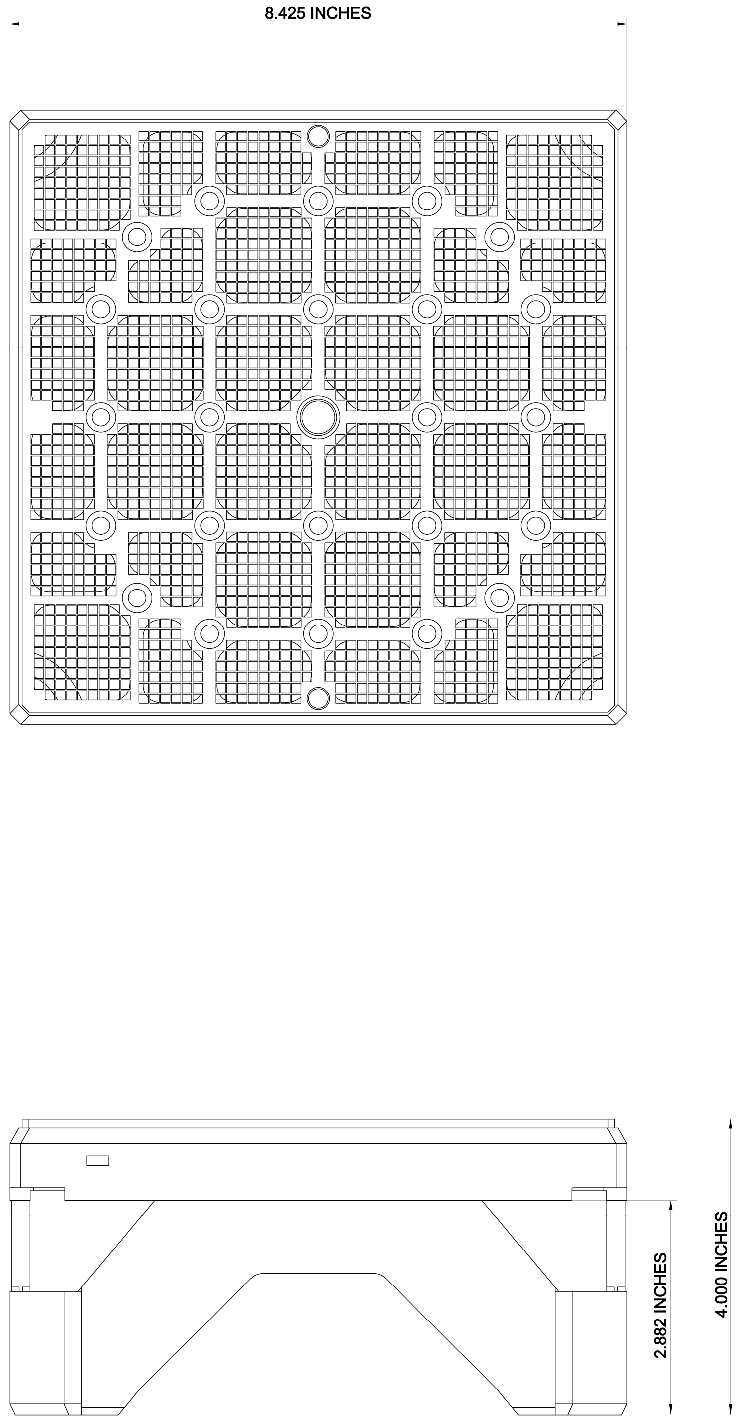

Figure 3-4   Bottom nozzle

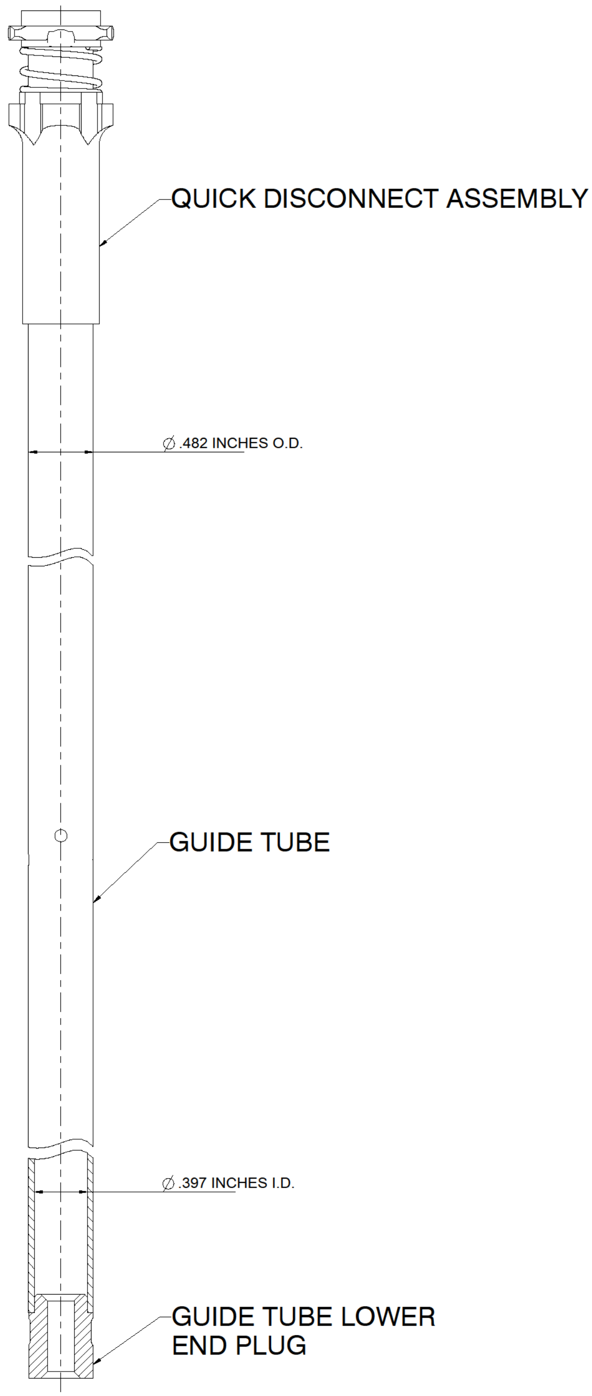

Figure 3-5 Guide tube assembly

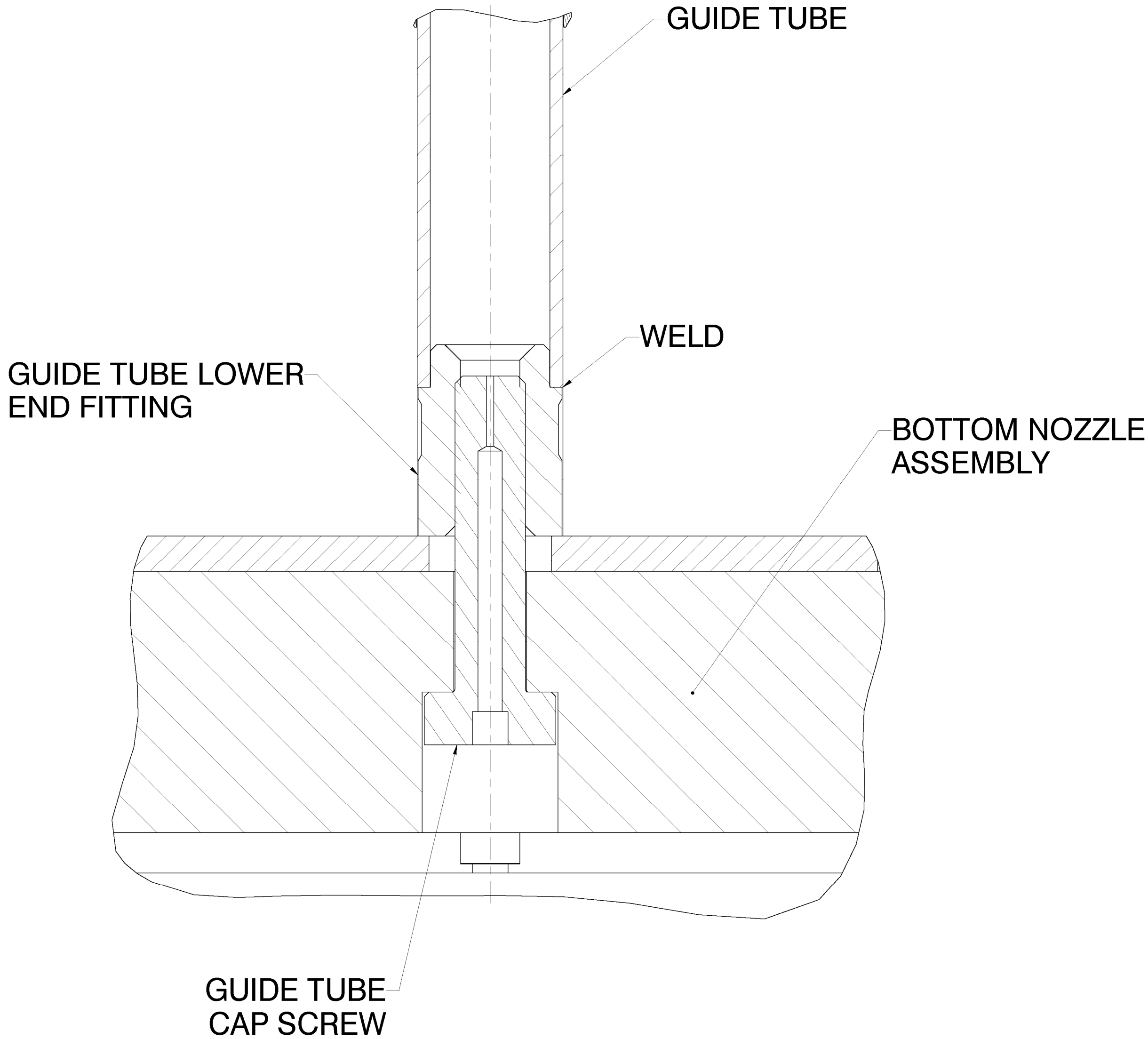

Figure 3-6   Cap screw bottom nozzle connection

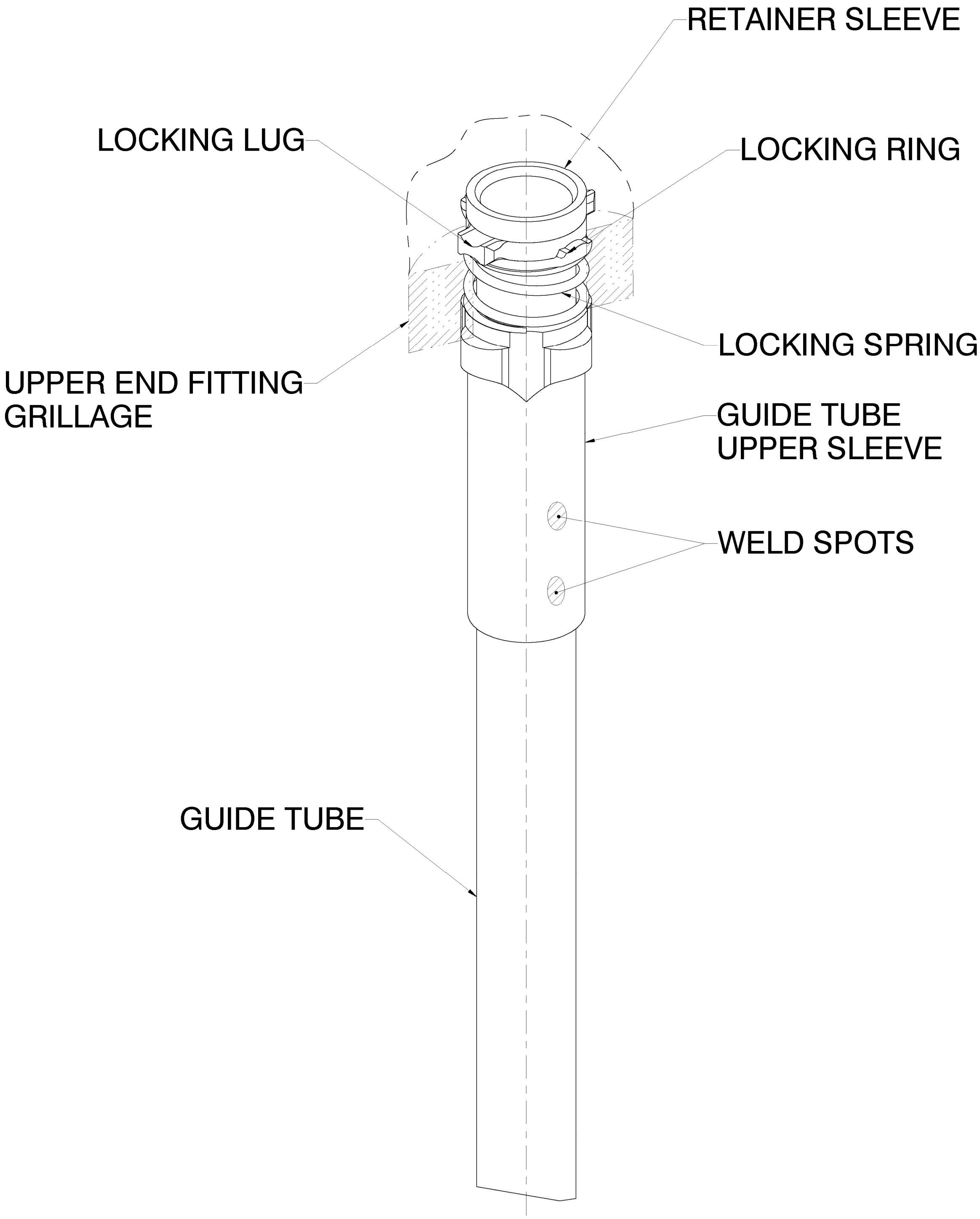

Figure 3-7 Guide tube quick disconnect top nozzle connection

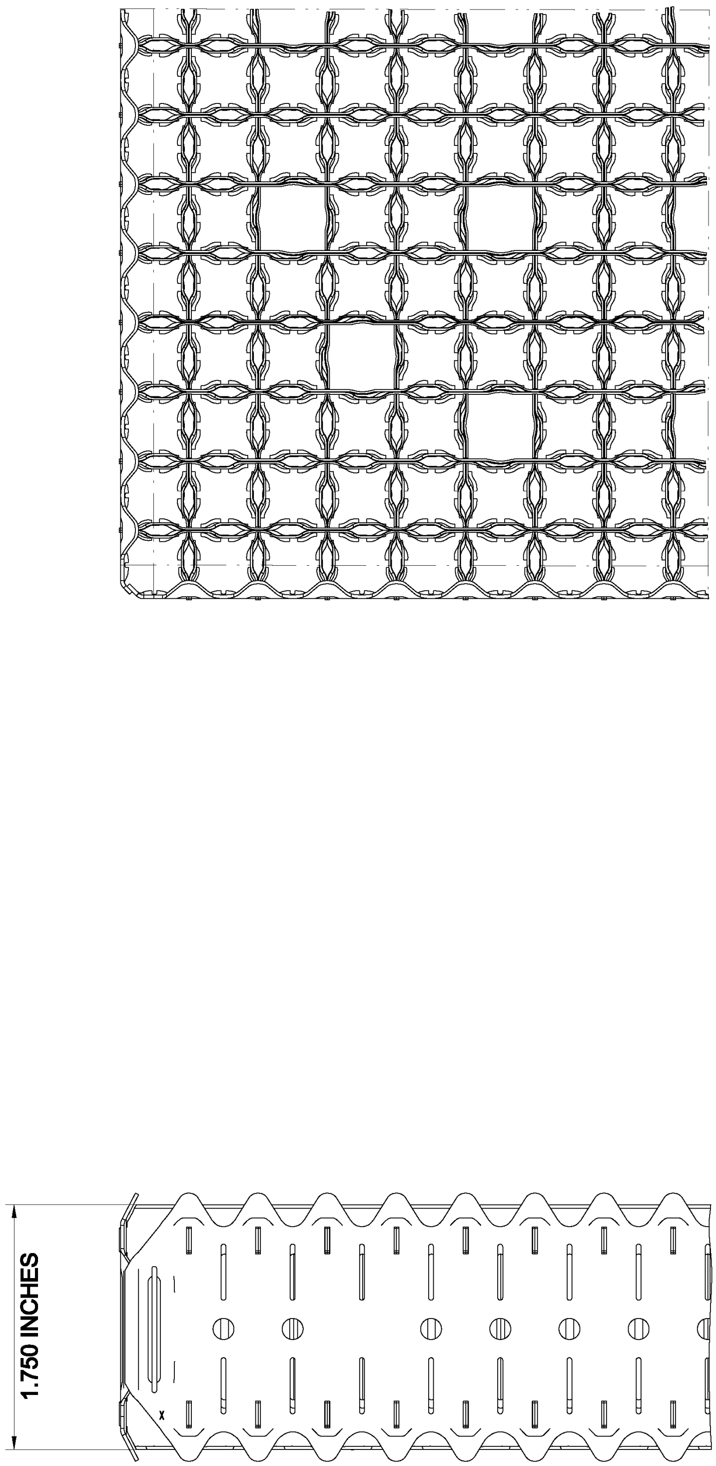

Figure 3-8 HTP™ grid

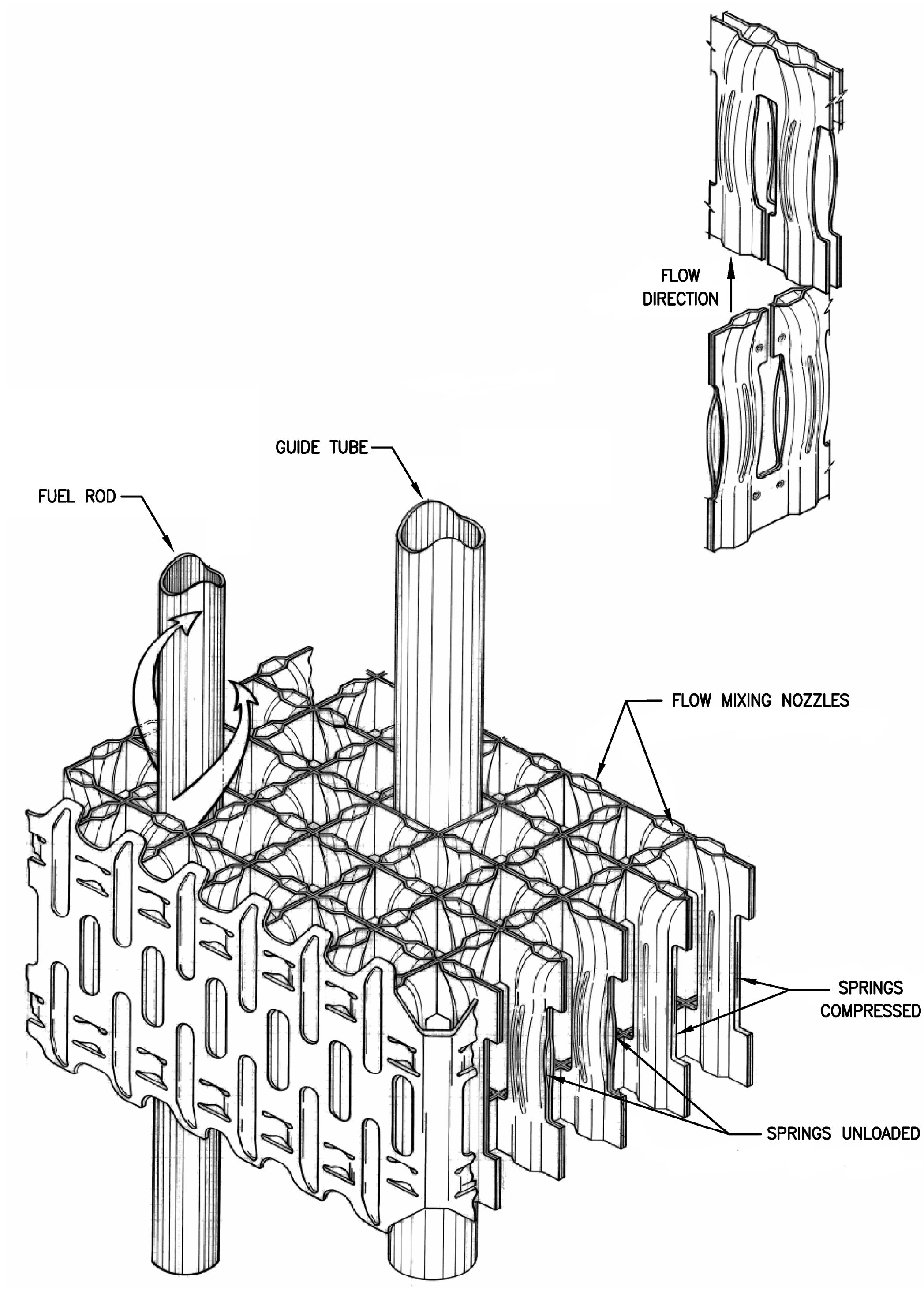

Figure 3-9 HTP™ spacer grid characteristics

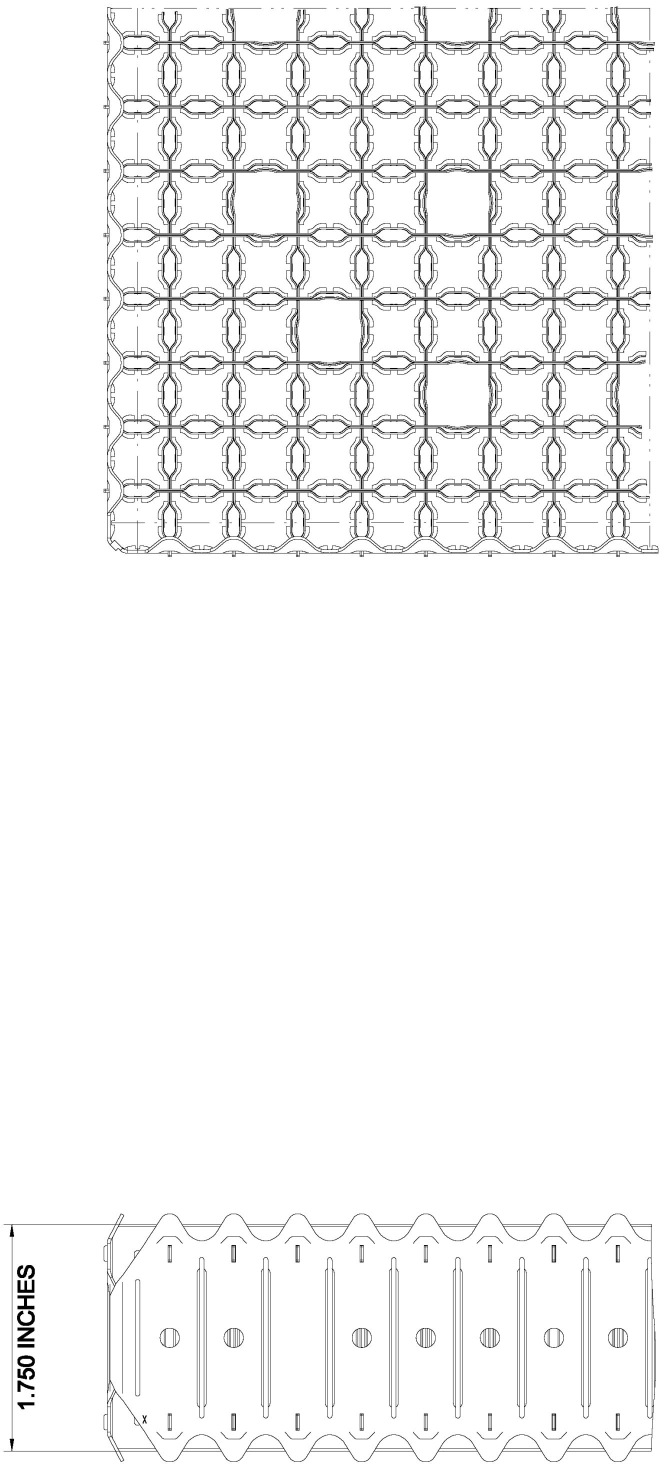

Figure 3-10 HMP™ spacer grid

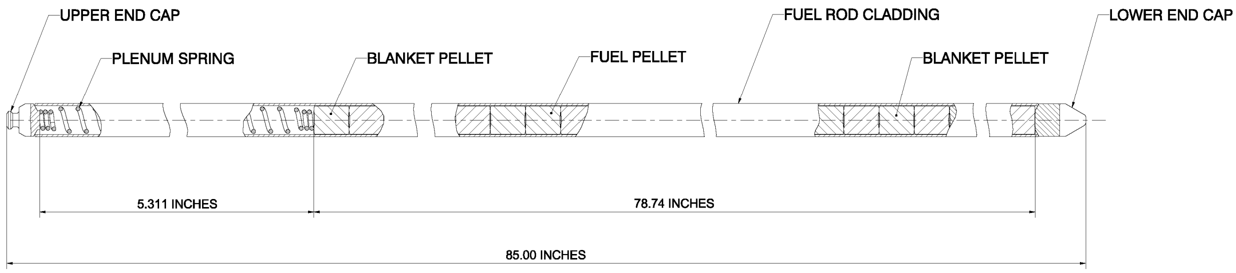

Figure 3-11 Fuel rod assembly

4.0 Design Evaluation

This section evaluates the NuFuel-HTP2™ design against criteria established consistent with Section 4.2 of NUREG-0800 (Reference 9.1.10), to provide assurance that: (1) the fuel system is not damaged as a result of normal operation and AOOs, (2) the fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) core coolability is always maintained. The design criteria are based on AREVA fuel design experience and are consistent with criteria established for previously approved fuel assembly designs.

Applicability of AREVA Fuel Methodology for the NuScale Design, TR-0116-20825-P Revision 1 (Reference 9.1.7) identifies the NRC-approved codes and methods used to evaluate the fuel performance. Table 2-3 of Reference 9.1.7 associates the method with the corresponding SRP acceptance criteria. *NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces*, TR-0716-50351-P (Reference 9.1.8) identifies the method of analysis for the structural response of the fuel assembly to dynamic faulted loads. Use of well-established design criteria and evaluation methods provides assurance of acceptable fuel performance.

The results of the fuel performance analyses are applicable for operation in the NuScale Power Module (NPM).

4.1 Fuel System Damage Criteria

4.1.1 Stress and Loading Limits

Design Criteria

Stress intensities for fuel assembly components shall be less than the stress limits based on ASME Code, Section III criteria. (Reference 9.1.6)

Buckling of the guide tubes shall not occur during normal operation and AOOs.

The cumulative number of strain fatigue cycles on the structural components should be significantly less than the design fatigue lifetime.

NuFuel-HTP2™ Design Evaluation

In the normal operating analysis, a series of mechanical analyses are performed on the NuScale fuel assembly to verify that it can withstand stresses and buckling loads from start-up, steady-state operation, shutdown, and AOOs. Each structural component in the fuel assembly is evaluated against the ASME Code Section III, Level A service limit (Reference 9.1.3). The fuel assembly weight, hold-down spring forces, RCS hydraulic loads, thermal loads during plant heat-up, steady-state operation and cooldown, and control rod assembly (CRA) drop loads (sum of the CRA spring preload and the load from the maximum travel of the CRA spring retainer) create the stress states that are evaluated in the analysis. The operating basis earthquake is less than 1/3 of the safe shutdown earthquake ground motion and is enveloped by the SSE analysis. The fatigue analysis evaluates cyclic loading due to normal operation and AOOs combined with the OBE, for a total of 182 transients over the life of the fuel. The NUREG-0800 criterion for a safety factor of 2 on stress amplitude or 20 on the number of cycles is satisfied by the use of the O'Donnell-Langer curve (Reference 9.1.13) in the analysis.

Guide tube normal operating loads are evaluated for each fuel assembly span. Guide tube spans are separated by spacer grids that begin above the bottom nozzle and extend to the guide tube span below the top nozzle. The loads and material properties of the guide tubes and the other fuel assembly components are based on the bounding outlet temperature in order to conservatively evaluate the loads, design limit, and minimum design margins for fuel assembly components at normal operating conditions.

The guide tube buckling analysis evaluates a maximum guide tube eccentricity to create bounding load predictions when the fuel assembly weight, hold-down spring force, and CRA drop loads are simultaneously applied.

Guide tube stress calculations consider axial loading conditions that cause tensile stresses. Hydraulic loading is conservatively ignored because it reduces the CRA drop and weight loads. Secondary loads are also considered in the form of spacer grid slip loads. These loads are generated as fuel rod slip is resisted by the grids due to differential thermal expansion and irradiation growth between the fuel rods and guide tubes during normal operation. These frictional loads are evaluated for a condition where rods are unseated, or lifted, at beginning of life (BOL) and are conservative for the seated condition at end of life (EOL).

Guide tube primary membrane (P_m), primary membrane + bending (P_m + P_b), and primary + secondary (P_m + P_b + Q) stresses are evaluated against allowable stresses using the ASME Code Level A service limits based on the material yield and ultimate strengths.

The guide tube upper sleeve strength is bounded by the strength of the weld connections with the guide tubes. The guide tube upper sleeve seating appendages are also evaluated for bearing stress and shear stress, considering the loads applied through the top nozzle during a CRA drop.

For welded or threaded joints and various structural connections in the fuel assembly, the maximum load during normal operation is evaluated against ASME Code Level A service limits. The evaluated connections include the guide tube to spacer grid welds, the guide tube to upper sleeve welds, the guide tube upper sleeve to QD retainer weld, the guide tube to guide tube lower end fitting weld, and the threaded connection between the guide tube lower end fitting and the shoulder screw. The threaded connection is evaluated using ASME code methods.

The bottom nozzle and top nozzle strengths are evaluated considering the maximum operating loads from the fuel assembly weight, hold-down spring force, and CRA drop events and the allowable ASME load limit based on prototype component tests. For both the bottom nozzle and top nozzle, testing determined the collapse load for each structural framework. In accordance with the ASME Code, each tested collapse load is multiplied by 2/3 in establishing the allowable load limit.

Table 4-1 provides a summary of the results for normal operation of the NuFuel-HTP2™ fuel assembly design. The results presented in Table 4-1 demonstrate that the assembly components meet the acceptance criteria.

Table 4-1 Summary of results – fuel assembly design margins

Consistent with Reference 9.1.6, the fuel handling qualification includes a 2.5g axial load. The shipping qualification includes 4g axial and 6g lateral loads and grid clamp loads. All loads for shipping and pre-receipt handling are evaluated for fresh fuel conditions. The maximum stresses and loads on the fuel assembly components and structural connections during shipping and handling remain below the specified minimum strength, critical buckling loads, and/or below the ASME Code Level A service limits. For a 4g axial acceleration, the fuel rod plenum spring maintains a force against the fuel stack sufficient to prevent column movement during handling. The evaluation also demonstrates that the fuel rods will not slip through the spacer grids under 4g axial loads.

4.1.1.1 Fuel Rod Cladding Stress and Buckling

<u>Design Criterion</u>

Fuel rod cladding stress shall not exceed the following stress limits defined in Reference 9.1.9:

- **[**

  ]

The fuel rod shall not buckle based on **[** **]** criterion during the limiting overpressure transient at BOL.

<u>NuFuel-HTP2™ Design Evaluation</u>

The fuel rod stress and buckling analysis determines the in-core steady-state stress and buckling performance of the fuel rod design.

[

] Pressure and temperature inputs are chosen so that operating conditions for all normal and AOO are enveloped.

The stress analysis takes into account several sources of cladding stress: pressure differentials, ovality, thermal differentials, flow-induced vibration (FIV), fuel rod growth, and fuel rod-spacer grid (FR-SG) interaction. The following four stress categories are analyzed:

- Primary Membrane (P_m) – Pressure stresses
- P_m + Primary Bending (P_b) – Pressure, ovality and FIV stresses
- P_m + P_b + Local – Pressure, ovality, FIV, and FR-SG stresses

- $P_m + P_b$ + Local + Secondary – Pressure , ovality, FIV, FR-SG , growth, and thermal stresses

At both the inner and outer diameter of the fuel rod, the maximum value of each individual stress is determined. The individual stresses within each stress component (tangential, axial, and radial) are added to find a maximum and minimum stress value. The stress intensity for each category is determined by combining the maximum and minimum stresses. The stress intensity is compared with the allowable stress to determine the margin for the particular stress.

The cladding stress results, listed in Table 4-2, show positive margins for all stress categories. The minimum margin occurs on the cladding outer diameter in compression when combining primary membrane + bending + local stresses.

Table 4-2 Stress results in compression (beginning of life) and tension (end of life)

[ ]

The buckling pressure is calculated to be **[ ]** psi. The buckling pressure is higher than the maximum BOL pressure the fuel rod would experience during an overpressure event (1910 psi). Therefore, the design meets the buckling criterion.

The calculated critical bucking load is **[ ]** lbf. The critical bucking load is greater than the total compressive load of **[ ]** lbf. Therefore, the Euler buckling criterion is also met.

4.1.2 Cladding Fatigue

<u>Design Criterion</u>

The fuel rod cumulative usage factor shall not exceed 0.9. (Reference 9.1.9)

<u>NuFuel-HTP2™ Design Evaluation</u>

A bounding analysis of the NuScale core design is performed using the COPERNIC fuel rod analysis code (Reference 9.1.1) for both UO_2 and UO_2-Gd_2O_3 rods.

Plant operations result in fluctuating thermal, pressure, ovality and pellet-clad contact stresses in the fuel rod cladding. The COPERNIC code predicts changes in cladding diameter, cladding temperature, and fuel rod internal pressure at each time step. These parameters are used to calculate the various stresses used in the fatigue calculation.

The transients considered in the fatigue analysis are provided in Table 4-3. The fuel rod life is conservatively assumed to be 10 years. With this assumption, the fuel rod will experience 10/60 of the number of transients identified in Table 4-3 for a sixty year plant design life.

The fuel rod behavior during each transient is analyzed using the COPERNIC fuel rod code, which predicts changes in cladding diameter and temperature and the fuel rod internal pressure for each time step. These parameters are used to calculate the various stresses used in the fatigue evaluation.

The maximum cumulative usage factor (CUF) for UO_2 fuel rods is **[ ]** and the maximum CUF for UO_2-Gd_2O_3 rods is **[ ]**. Both of these CUFs are well below the limit of 0.9.

Table 4-3 Summary of transients considered in the fuel rod fatigue analysis

Number	Description	Number of Cycles
	Condition I Events (Service Level A)	
1	Reactor heatup to hot standby	200
2	Reactor cooldown from hot standby	200
3	Power ascent from hot standby	500
4	Power descent to hot standby	500
5	Load following	19,750
6	Load regulation	767,100
7	Steady-state fluctuations	5,260,000
8	Load ramp increase	2000
9	Load ramp decrease	2000
10	Step load increase	3000
11	Step load decrease	3000
12	Large step load decrease	200
	Condition II Events (Service Level B)	
13	Turbine trip without bypass	60
14	Loss of all AC power	60
15	Inadvertent main steam isolation valve closure	30
16	Inadvertent operation of decay heat removal system	15
17	Reactor trip from reduced power	180
18	Reactor trip from full power	120
19	Inadvertent control rod assembly drop	60
20	Inadvertent pressurizer spray	15
21	Inadvertent opening of a reactor safety valve	20
	Condition III Events (Service Level C) (only one of the following is considered)	
22	Spurious reactor vent valve actuation	15
23	Spurious reactor recirculation valve actuation	15
24	Small break loss-of-coolant accident	5
25	Steam generator tube rupture	5

4.1.3 Fretting

Design Criterion

Fuel rod failures due to fretting shall not occur, as verified by fretting tests. (Reference 9.1.6)

NuFuel-HTP2™ Design Evaluation

Fretting and vibration performance is validated by the 1000-hour life and wear test performed at the Richland portable hydraulic test facility (PHTF), in addition to other relevant HTP™ vibration and fretting tests.

- The 1000-hour life and wear test performed at the Richland PHTF was run for 1032 hours at a temperature of 300°F at or above the target Reynolds number of 52,000. At the conclusion of the test, **[ ]** fuel rods were examined for grid-to-rod fretting performance. The test results show no wear abnormalities with results **[ ]** well within the performance base for historical test results of proven in-reactor designs. The fretting results are based on a conservative flow configuration with a test-to-reactor momentum flux ratio of approximately **[ ]**. The test assembly replicated the EOL condition **[ ]** The HMP™ grid for the EOL test assembly was relaxed to **[ ]** of the unirradiated grid-to-rod support.
- The predicted vibration response amplitude for the NuFuel-HTP2™ design is **[ ]**, which is significantly less than the maximum measured rod amplitude of **[ ]** in the Hermes-T vibration and wear test performed for the 17x17 HTP™ fuel transition, which considered the effects of bundle-to-bundle cross flow.
- Fretting results for autoclave testing (8005 hours) of the 17x17 HTP™ grid design, which is identical to the HTP™ grid used on the NuFuel-HTP2™ design, show low wear **[ ]** for very conservative imposed vibration amplitudes **[ ]**.
- Fretting results for autoclave testing (1000 hours) of the Advanced W17 HTP™ design with intermediate flow mixers, where the HTP™ and HMP™ grids are similar to the NuFuel-HTP2™ design, show low wear **[ ]** for very conservative imposed vibration amplitudes and **[ ]**.

The predicted small vibration amplitudes for the NuFuel-HTP2™ design are a consequence of much lower axial and cross flow velocities due to the natural circulation (no mechanical pumps) of the NuScale reactor coolant system. The NuScale reactor has a nominal flow rate of 259 gpm per fuel assembly compared to a best-estimate flow of approximately 2050 gpm per fuel assembly for a forced circulation PWR using a 17x17 fuel assembly design. The maximum calculated local cross flow velocity for NuScale fuel assemblies is **[ ]** ft/sec for a uniform core compared to approximately **[ ]** ft/sec for a conventional PWR using a 17x17 fuel assembly design.

The robust fretting characteristics of the NuFuel-HTP2™ design provide confidence in the FIV performance of the fuel. The grid design provides line contact between the fuel rods and the spacer grid with large contact surfaces to mitigate wear. The grid-to-rod support also provides for higher damping to help suppress flow-induced vibration and fretting wear. The grid-to-rod support conditions in the NuFuel-HTP2™ design are similar to the grid-to-rod support of other HTP™ fuel assemblies, of which more than 18,000 have been introduced into operating reactors globally. Thus, the HTP™ test results and operating experience pertaining to rod vibration and fretting is applicable for evaluating the NuFuel-HTP2™ design.

Based on the minimal fretting wear measured during the life and wear test in the Richland PHTF, the small predicted rod vibration amplitudes, and the extensive favorable operating and test experience with the HTP™ fuel design, the NuFuel-HTP2™ design is not expected to experience FIV or wear issues in the NuScale reactor core, and fuel rod failures due to fretting will not occur. There are no limitations with respect to time or burnup for the conditions evaluated.

4.1.4 Oxidation, Hydriding, and Crud Buildup

Design Criterion

The fuel rod cladding peak oxide thickness shall not exceed a best-estimate predicted value of 100 microns. Hydrogen pickup is controlled by the corrosion limit. Crud buildup is limited by inclusion in the oxidation measurement. (Reference 9.1.1)

NuFuel-HTP2™ Design Evaluation

A bounding analysis is performed for NuScale core designs using the COPERNIC fuel rod analysis code (Reference 9.1.1) for UO_2 rods with and without Gd_2O_3. The corrosion of the fuel rods is modeled in order to calculate the oxide thickness that develops on the outer surface of the rods during operation.

An input power history envelope, Figure 4-2, expressed in terms of effective full power hours is used that bounds the individual rod power histories of all the UO_2 and Gd_2O_3 rods in the NuScale equilibrium cores. The corrosion analysis primarily depends on the amount of energy transfer through the cladding and the irradiation time. It shows little sensitivity to the fuel rod design characteristics inside the rods; therefore, only a UO_2 rod is explicitly modeled. The use of a bounding power history envelope makes the analysis equally applicable to all fuel rod types.

The predicted oxide thickness is shown in Figure 4-2. The maximum oxide thickness reaches **[ ]** micrometers, which is well below the design limit of 100 micrometers.

[ ]

Figure 4-1 Bounding power history envelope

Figure 4-2 Predicted corrosion results

The EOL axial corrosion profile is shown in Figure 4-3. The maximum oxide thickness occurs in the lower half of the core, slightly below the middle of the pellet stack. This behavior is because the axial power distributions are consistently peaked towards the lower half of the core, causing the coolant to reach near-saturation conditions under bounding conditions at a core elevation lower than for typical PWR designs.

Figure 4-3 Axial corrosion profile

4.1.5 Fuel Rod Bow

Design Criterion

There is no specific design criterion for fuel rod bow. Fuel rod bowing is evaluated with respect to the mechanical and thermal-hydraulic performance of the fuel assembly. (Reference 9.1.4)

NuFuel-HTP2™ Design Evaluation

Fuel rod bow is the deviation from straightness of the fuel rods in the fuel assembly. The presence of fuel rod bow is identified by the deviation in water channel gap from nominal conditions. The primary effects of rod bow are a decrease in the critical heat flux ratio and an increase in local power peaking. The secondary effects of fuel rod bow can include fuel clad fretting at 100 percent gap closure, although the probability of rod-to-rod contact is minimal.

The NuFuel-HTP2™ design does not introduce any changes from current AREVA PWR fuel assembly designs that might adversely impact rod bow. Operating plant rod bow data for current AREVA PWR designs continue to be adequately covered by the existing rod bow correlation methodology.

The NuFuel-HTP2™ design is within the current experience base with regards to fuel rod bending stiffness and operating temperature and is less limiting regarding end grid slip loads and span length. The mechanical rod bow analysis concludes that fuel assembly performance is within current models and experience.

Rod bow penalties are derived for both linear heat generation rate (LHGR) and critical heat flux (CHF) based on the NRC-approved methodology for quantifying fuel rod bowing and its effects, demonstrated to be applicable to the NuScale design in Reference 9.1.7.

4.1.6 Axial Growth

Design Criteria

For the fuel assembly, the axial clearance between core plates and the top and bottom nozzles shall allow sufficient margin for fuel assembly growth during the assembly lifetime.

For the fuel rod, adequate clearance shall be maintained between the fuel rod and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly. (Reference 9.1.6)

NuFuel-HTP2™ Design Evaluation

The fuel assembly and its components grow during operation. There are two components of the growth: thermal expansion and irradiation growth. The growth analysis is performed for the life of the fuel, in which a representative equilibrium fuel cycle yields an average fuel assembly burnup of {{ }}[2(a)(c),ECI] GWd/mtU with a maximum fuel rod burnup of {{ }}[2(a)(c),ECI] GWd/mtU. Significant margin exists for this representative cycle to allow alternate fuel cycle designs with higher burnups consistent with the approved methodology in Reference 9.1.6.

The minimum clearance between the NuScale fuel rods and the top and bottom nozzles, and the clearance between the fuel assembly and core plates at the EOL condition are determined using worst case fuel rod and fuel assembly growth models and worst case initial dimensions.

[

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The analysis demonstrates significant margin for a representative equilibrium fuel cycle with a maximum average assembly fluence of {{
}}[2(a)(c),ECI] and a maximum fuel rod fluence of {{
}}[2(a)(c),ECI]. Significant margin exists to allow alternate fuel cycle designs with higher burnups consistent with the approved methodology in Reference 9.1.6.

[
]

The minimum clearance between the fuel rod and the top and bottom nozzles, i.e., total shoulder gap (top plus bottom clearance), is **[ ]** inches, at EOL hot conditions.

4.1.7 Fuel Assembly Distortion Evaluation

The NuFuel-HTP2™ fuel has features, including spacer grids, structural connections, and guide tube diameters, similar to current AREVA 17x17 fuel but with a shorter overall length. The shorter length increases the lateral stiffness of the fuel assembly. As validation, fuel assembly lateral stiffness tests were performed for the NuScale fuel design in-air at BOL and EOL conditions.

Similar lateral stiffness tests have been conducted for current AREVA 17x17 fuel designs at EOL and BOL conditions. The test results show that the NuScale fuel assembly lateral stiffness is more than **[ ]** times greater than that of the current AREVA 17x17 fuel design. With this level of lateral stiffness and significantly lower hydraulic loads on the fuel assembly due to natural circulation flow, the NuScale fuel assembly has a high level of resistance to fuel assembly distortion.

Differential fuel rod and guide tube growth rates, coupled with spacer grid slip loads, can contribute to fuel assembly distortion during operation. The NuScale fuel design has the same structural components and fuel rod cladding diameters and material (producing similar slip loads and growth rates) as the current AREVA 17x17 PWR designs; thus, the guide tube stresses from fuel rod and guide tube differential growth are bounded by AREVA design experience. The differential growth stresses imparted to the guide tubes are reduced compared to AREVA experience due to the reduced number of spacer grids over which tensile loads may accumulate on the guide tubes and the reduced length of the fuel rods and guide tubes, which results in lower differences in growth. These design characteristics ensure that the fuel rod and guide tube growth differential effects are within AREVA's recent PWR 17x17 fuel design experience.

Operating experience of the current AREVA 17x17 fuel assembly design demonstrates little in-reactor fuel distortion as evidenced by the absence of incomplete rod insertions and slow-to-settle observations where full insertion of the control rod is delayed. Therefore, control rod drop concerns related to assembly distortion are not expected for the NuFuel-HTP2™ fuel assembly design.

4.1.8 Fuel Rod Internal Pressure

Design Criterion

The internal gas pressure of the peak fuel rod in the reactor shall be limited to a value below that which would cause: (1) the fuel-cladding gap to increase due to outward cladding creep during steady-state operation, and (2) extensive DNB (critical heat flux) propagation to occur. (Reference 9.1.1)

NuFuel-HTP2™ Design Evaluation

A bounding analysis is performed for NuScale core designs using the COPERNIC fuel rod analysis code (Reference 9.1.1) for UO_2 rods with and without Gd_2O_3. The maximum fuel rod internal pressure is conservatively compared to a design limit equal to system pressure (1850 psia). Meeting this criterion demonstrates that the fuel-clad gap does not increase due to cladding outward creep during steady-state operation because a greater pressure external to the rod prevents outward creep and fuel-clad liftoff. This criterion also ensures that extensive DNB propagation does not occur.

The COPERNIC pressure calculation is based on a best-estimate prediction plus an uncertainty allowance to take into account code uncertainties and manufacturing variations. The analysis considers steady-state and Condition I (normal operation) and Condition II (AOO) transients over the full burnup range. The transients are modeled in the COPERNIC input with appropriate axial flux shapes.

The maximum calculated internal pressure over the burnup history (for both UO_2 and Gd_2O_3 rods) is **[ ]** psia compared to a limit of 1850 psia.

4.1.9 Assembly Liftoff

Design Criterion

The fuel assembly shall not lift off from the lower core plate under normal operating conditions and AOOs. (Reference 9.1.6)

NuFuel-HTP2™ Design Evaluation

The fuel assembly lift-off analysis evaluates an {{ }}$^{2(a)(c)}$ AOO. To bound conditions for this event, the maximum flow rates at each corresponding power level are {{ }}$^{2(a)(c)}$. The maximum hydraulic lift is **[ ]** lbf, resulting in a lift margin of **[ ]** lbf. There are large margins against lift-off at all normal operating, startup, and transient (AOO) conditions. Because the NuScale plant design relies on natural circulation of the coolant without any mechanical pumps, the assumed flow rates envelope all operating conditions and AOOs.

4.2 Fuel Rod Failure Criteria

4.2.1 Internal Hydriding

<u>Design Criterion</u>

The fabrication limit for total hydrogen inside a fuel rod assembly is maintained at a minimal level to limit internal hydriding. (Reference 9.1.6)

<u>NuFuel-HTP2™ Design Evaluation</u>

Fuel rod internal hydriding is controlled by fabrication limits for fuel pellet moisture. These controls, typical for AREVA fuel manufacturing, limit the total hydrogen content, including moisture, to ≤ **[ ]** ppm by weight, before rod final closure welding.

4.2.2 Cladding Collapse

<u>Design Criterion</u>

The predicted creep collapse life of the fuel rod shall exceed the maximum expected incore life. (References 9.1.2 and 9.1.9)

<u>NuFuel-HTP2™ Design Evaluation</u>

The cladding creep collapse analysis is performed using the methodology of Reference 9.1.2, extended to M5® applications in Reference 9.1.9.

A bounding analysis of the NuScale core design is performed using the COPERNIC fuel rod analysis code and the CROV creep ovalization code for both UO_2 and Gd_2O_3 rods. Reference 9.1.2 establishes the three collapse criteria to be analyzed: bifurcation buckling pressure, yield stress, and deformation rate.

COPERNIC simulates the performance of the fuel rod throughout the lifetime of the rod to generate the parameters required to perform the creep collapse analysis: rod internal pressure, interior and exterior cladding temperatures, coolant temperature and fast flux.

The CROV code applies the initialization parameters from COPERNIC along with the fuel rod geometry (outside diameter, wall thickness, and ovality) to simulate the cladding creep-down deformations versus time. Consistent with Reference 9.1.2, when the ovality creep rate of the cladding exceeds 0.1 mils/hr, or the generalized stress within the cladding exceeds the yield stress, the cladding is considered to have failed. In addition, the bifurcation buckling pressure must not be exceeded. The three collapse criteria determine the predicted creep collapse life of the fuel, which must exceed the maximum expected incore life.

The CROV analysis demonstrates that the bifurcation buckling pressure limit is not exceeded. A NuScale bifurcation buckling pressure limit of **[ ]** psi is calculated that approximates the limit from the CROV analysis. The CROV analysis does not explicitly calculate the margin to the limit; however, the maximum pressure differential applied to the fuel rod in the CROV runs occurs at BOL and is approximately **[ ]** psi, which indicates that the bifurcation buckling pressure collapse criterion is satisfied with significant margin.

Figure 4-4 demonstrates that the fuel rod will not collapse as a result of stress beyond the yield point over the expected three cycle incore life.

Figure 4-4 CROV results – cladding stress vs. time

Figure 4-5 demonstrates that the fuel rod will not collapse as a result of deformation rate exceeding the limit of 0.1 mil/hr over the expected incore life.

Figure 4-5 CROV results – cladding deformation rate vs. time

All three collapse criteria are met. Therefore, the predicted creep collapse life of the fuel rod exceeds the maximum expected incore life of the fuel rod and cladding creep collapse will not occur.

4.2.3 Overheating of Cladding

Overheating of cladding is evaluated in the NuScale Chapter 15 transient analyses and is not addressed in this report.

4.2.4 Overheating of Fuel Pellets

<u>Design Criterion</u>

Fuel melting during normal operation and AOOs shall be precluded. (Reference 9.1.1)

<u>NuFuel-HTP2™ Design Evaluation</u>

The centerline fuel melt (CFM) analysis is performed using the methodology of Reference 9.1.1.

A bounding analysis of the NuScale core design is performed using the COPERNIC fuel rod analysis code for both UO_2 and UO_2-Gd_2O_3 rods. The COPERNIC code predicts the transient linear heat rates (LHRs) where the onset of fuel centerline melting occurs. **[**

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Using the calculated LHR limits for each of the fuel rods, a bounding envelope is created, as shown in Table 4-4. Figure 4-6, Figure 4-7, Figure 4-8, Figure 4-9, and Figure 4-10 show the bounding envelopes for both transient cladding strain (TCS) and CFM limits for each fuel rod type. **[**

] The LHR limits are used in the NuScale core design.

Table 4-4 Bounding centerline fuel melt limits

Figure 4-6 Centerline fuel melt and transient cladding strain bounding envelopes for UO_2 fuel

Figure 4-7 Centerline fuel melt and transient cladding strain bounding envelopes for 2 wt% Gd_2O_3 fuel

Figure 4-8 Centerline fuel melt and transient cladding strain bounding envelopes for 4 wt% Gd_2O_3 fuel

Figure 4-9 Centerline fuel melt and transient cladding strain bounding envelopes for 6 wt% Gd_2O_3 fuel

Figure 4-10 Centerline fuel melt and transient cladding strain bounding envelopes for 8 wt% Gd_2O_3 fuel

4.2.5 Excessive Fuel Enthalpy

Excessive fuel enthalpy from a reactivity initiated accident is addressed in the NuScale FSAR Chapter 15 analyses.

4.2.6 Pellet/Cladding Interaction

<u>Design Criteria</u>

As stated in NUREG-0800 Section 4.2, there is no generic criterion for fuel failure resulting from pellet/cladding interaction or pellet/cladding mechanical interaction. Cladding strain and fuel melt criteria are applied as a surrogate.

The maximum uniform hoop strain (elastic plus plastic) shall not exceed **[ ]**. Steady-state creep-down and irradiation growth are excluded. (Reference 9.1.1)

The fuel melt criterion is stated in Section 4.2.4.

<u>NuFuel-HTP2™ Design Evaluation</u>

The transient cladding strain (TCS) analysis is performed using the methodology of Reference 9.1.1.

A bounding analysis of the NuScale core design is performed using the COPERNIC fuel rod analysis code for both UO_2 and UO_2-Gd_2O_3 rods. The COPERNIC code predicts the transient LHRs where the cladding uniform hoop strain equals **[ ]**.

[

]

Using the calculated LHR limits for each of the fuel rods, a bounding envelope is created, as shown in Table 4-5. Figure 4-6, Figure 4-7, Figure 4-8, Figure 4-9, and Figure 4-10 show the bounding envelopes for both TCS and CFM limits for each fuel rod type. The LHR limits are used in the NuScale core design.

Table 4-5 Bounding transient cladding strain limits

[ ]

4.2.7 Bursting

Swelling and rupture of the cladding relates to the emergency core cooling system performance evaluation that is part of the NuScale FSAR Chapter 15 analyses and is not addressed in this report.

4.2.8 Mechanical Fracturing

The seismic and loss-of-coolant accident (LOCA) load analysis summarized in Section 4.3.5 addresses externally applied forces on the fuel rod.

4.3 Fuel Coolability

4.3.1 Cladding Embrittlement

Cladding embrittlement relates to the emergency core cooling system performance evaluation and is not addressed in this report.

4.3.2 Violent Expulsion of Fuel

Because reactivity initiated accidents are addressed in the NuScale FSAR Chapter 15 analyses, they are not addressed in this report.

4.3.3 Generalized Cladding Melting

As stated in NUREG-0800 Section 4.2, criteria for cladding embrittlement in Section 4.3.1 are more stringent than generalized cladding melting criteria. Therefore, additional specific criteria are not used.

4.3.4 Fuel Rod Ballooning

Burst strain and flow blockage caused by ballooning of the cladding relates to the emergency core cooling system performance evaluation and is not addressed in this report.

4.3.5 Fuel Assembly Structural Damage from External Forces

Design Criterion

The fuel assembly shall withstand the loads from a safe shutdown earthquake or LOCA. Specific acceptance criteria for fuel assembly components are identified in Reference 9.1.5.

NuFuel-HTP2™ Design Evaluation

The external load analysis is performed using the methodology of Reference 9.1.5 and Reference 9.1.8. The analysis demonstrates positive margins to all criteria.

4.3.5.1 Analysis Inputs

4.3.5.1.1 Lateral Model

The lateral fuel assembly model is developed using the model parameters from Section 6.1 of Reference 9.1.5. Damping coefficients specific to the NuFuel-HTP2™ design are summarized in Table 4-6 and defined in Reference 9.1.8.

Table 4-6 Summary of NuScale fuel assembly damping ratios

[ ]

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

- **[**
  ]
- **[**
  ]

[**]** The application of the free and forced vibration test data from prototypical BOL and EOL NuFuel-HTP2™ fuel assemblies to define the fuel assembly dynamic characteristics is described in Appendix A of Reference 9.1.8.

Dynamic crush testing was performed on prototypical NuFuel-HTP2™ spacer grids in both a non-irradiated (BOL) and simulated-irradiated (EOL) condition to define the external spacer grid stiffness and damping characteristics. The **[** **]** is demonstrated in Appendix A of Reference 9.1.8. The dynamic crush test is also used to establish a grid load limit. **[** **]** The grid loads are reported in Table 4-7.

Fuel assembly lateral impact testing was performed on prototypical NuFuel-HTP2™ fuel assemblies in both a BOL and EOL condition to establish the internal stiffness and damping parameters for the spacer grid.

4.3.5.1.2 Vertical Model

The vertical fuel assembly model is developed using the model parameters from Section 6.2 of Reference 9.1.5. The following model parameters are established through design-specific characterization testing:

- **[**

  ]
- **[**

  ]

Spacer grid slip load tests define the grid-to-fuel rod slider slip load at BOL conditions. In this test, a prototypical NuFuel-HTP2™ spacer grid is loaded with cladding segments and a uniform load is applied across all of the cladding segments. The load at which the fuel rods are observed to begin slipping through the grid is identified as the global slip load. To simulate the EOL condition, **[** **]** The Alloy 718 lower end grid is **[** **]**

An axial stiffness test was performed on full-scale prototypical NuFuel-HTP2™ fuel assemblies in both the BOL and EOL condition. The axial stiffness of the fuel assembly is measured at key axial locations (e.g., spacer grids and top nozzle). The measurements of location-specific axial stiffness are used to benchmark the stiffness of the grid-to-fuel rod slider elements in the vertical model.

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

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

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[

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4.3.5.1.3 Excitation Inputs

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

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

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

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

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

4.3.5.2 Analysis Results

4.3.5.2.1 Lateral Analysis

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

The peak impact loads for the base case seismic inputs, along with margin to the grid impact load limit, from all cases are summarized in Table 4-7. The peak impact loads for SSE and LOCA in a given direction are combined by the square root of the sum of the squares (SRSS) method and margin is calculated against this SRSS impact load. The positive margin for these impact loads confirms that the NuScale spacer grid will not experience plastic deformation that exceeds the limit established in the AREVA methodology (Reference 9.1.5). Thus, the requirements for core coolability and control rod insertion are met.

Table 4-7 Peak grid impact loads and margins

[ ]

4.3.5.2.2 Vertical Analysis

The single assembly vertical model is subjected to vertical core plate displacement time histories corresponding to the SSE and LOCA events. The maximum seismic impact load for the base case inputs, **[ ]** The maximum LOCA impact load is **[ ]** Component loads for the guide tubes, fuel rods, hold-down spring, nozzles, and guide tube connections are extracted from the vertical analysis for further load analysis.

4.3.5.2.3 Stress Analysis

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

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

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The results of the component evaluations are presented in Table 4-8.

Fuel rods

The accident loads on the fuel rod consist of the lateral bending stress, the axial stress from vertical loads, and an additional loading corresponding to impacts to the fuel assembly. The impact induced stress is a localized bending stress resulting from the portion of the grid impacts that are passed through the fuel rods (i.e. internal impact loads). The overall combined stress is the SRSS of the individual components combined with steady state stresses to determine the maximum stress intensity.

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The fuel rods are also evaluated for buckling under compressive loads.

Guide tubes

The accident loads on the guide tube consist of **[** **]** The overall combined stress is calculated as the SRSS of the individual components combined with steady state stresses to determine a maximum stress state. The allowable stress limits are based on ASME Level C service limits, consistent with Reference 9.1.5.

The guide tubes are also evaluated for buckling under compressive loads.

Guide tube-to-spacer connections

The vertical load acting on the guide tube-to-spacer grid connection is **[**

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The combined load is calculated as the SRSS of the individual components. The allowable strength of the guide tube-to-grid connections is based on the ASME Level D service limit and is established through testing.

Guide tube-to-nozzle connections

The loads considered in the evaluation of the guide tube-to-nozzle connections are **[**

] The allowable load for the top nozzle connection is based on the ASME Level C service limit and is established through strength testing. The allowable load for the bottom nozzle connection is based on ASME Level D service limits, because the bottom nozzle cannot affect control rod insertion.

Top and bottom nozzles

The loads considered in the evaluation of the nozzles are the axial loads from the vertical analysis. The allowable strength of the top and bottom nozzle is established by testing. Due to the robustness of the nozzles, the testing is not carried to the extent of failure and thus the allowable load and resulting margin are artificially low.

Table 4-8 Component evaluation margins

[ ]

4.4 Thermal Hydraulic Evaluation

4.4.1 Core Pressure Drop Evaluation

An evaluation is performed of the fuel assembly pressure drop characteristics. The recoverable and unrecoverable pressure drop is determined for each of the 37 fuel assemblies in the reactor core over a wide range of core power and system flow conditions. Pressure loss coefficients based on testing are reported in Table 5-1 and are used in the evaluation.

The results of the pressure drop analysis indicate that the total pressure drop is dominated by the recoverable loss due to elevation change. For the full power best-estimate case, the channel with the greatest unrecoverable pressure drop has an elevation loss approximately **[ ]** times greater than the unrecoverable pressure drop. The overall core total pressure drop for the best estimate case is **[ ]**

Figure 4-11 shows the difference in total pressure drop by channel relative to the average total pressure drop for the core. As shown in the figure, each assembly is represented by one channel. Figure 4-12 shows the difference in the unrecoverable pressure drop by channel relative to the average unrecoverable pressure drop for the core. Figure 4-13 shows the cumulative unrecoverable pressure drop in the channels with the maximum and minimum unrecoverable pressure drop. The results are based on operation at 100 percent power, 100 percent flow with best-estimate assembly-by-assembly distributions for core inlet flow, core inlet temperature, and assembly radial power.

Figure 4-11 Difference in total pressure drop by channel – best estimate

Figure 4-12 Difference in unrecoverable pressure drop by channel – best estimate

Figure 4-13 Cumulative unrecoverable pressure drop – best estimate

4.4.2 Guide Tube Boiling

The guide tube boiling analysis determines the water temperature profile inside the guide tubes of all assemblies, with the limiting conditions found in those fuel assemblies that contain CRAs. The analysis considers the limiting CRA positions – the normal parked position and the power dependent insertion limit. The design criterion is that long term bulk boiling in the guide tube is precluded. This criterion is satisfied by demonstrating that the coolant flow inside the guide tube remains below the saturation temperature under conservative conditions for flow, temperature, pressure, and assembly power.

The guide tube coolant temperatures are dependent on the coolant flow rates inside the guide tubes, the amount of radiation heating of the control rod, the direct heating of the water inside the guide tube, and the direct heating of the guide tube itself. Heat transfer with the adjacent subchannel, by conduction through the guide tube wall and convection, is modeled, consistent with the temperature gradient.

Conservative analyses are performed by evaluating normal operation as a function of power and RCS flow rate. Core parameter uncertainties are considered in the areas of **[**

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The guide tube flow rate analysis determines the coolant flow in the guide tube as a function of elevation. Flow enters the guide tube through holes in the side of the guide tube and from below the bottom nozzle into the guide tube by way of the through-hole in the cap screw.

Analysis results for the 100 percent full power case and 75 percent power case indicate that bulk boiling will not occur at any location within the guide tube under steady state conditions. The results indicate that the 100 percent power case is limiting.

4.4.3 Control Rod Drop Analysis

The control rod drop analysis predicts the insertion rate and impact velocity of the control rod assembly during a reactor trip. The calculated impact velocity is compared to the maximum acceptable impact velocity for the CRA spring in Section 6.2.9.

When the CRA is dropped into a fuel assembly, water in the guide tube is displaced through several flow paths. The rate of displacement depends on the number, size, and location of the holes along the guide tube. The NuScale fuel assembly design has 24 guide tubes, each containing two pairs of side flow holes at the entrance to the dashpot. In addition, water is forced out through the top annulus of the guide tube and through the hole in the cap screw at the bottom of the guide tube assembly.

A best-estimate, mechanistic model is used to evaluate the impact velocity based on the guide tube and control rod geometry, nonlinear coefficients for drag loss (hydraulic and mechanical) and the equation of motion. Drag coefficients are based on control rod drop measurements from a 17x17 PWR plant with a similar fuel geometry. Coolant flow velocity through the guide tube is conservatively assumed to be zero.

The CRA impact velocity limit is defined in Section 6.2.9 based on the CRA spring design. The control rod drop analysis predicts an impact velocity of 2.32 ft/sec, which is below the impact velocity limit, and a drop time of 1.08 sec, where drop time is defined as the time between the start of rod movement and the time of full insertion. Figure 4-14 shows axial position versus time for the CRA drop based on a starting position of 79.280 inches from the top of the fuel assembly bottom nozzle to the lower tip of the control rod. Figure 4-15 shows CRA velocity versus time. For reactivity insertion use in the plant safety analyses, the best estimate curve of Figure 4-14 is conservatively modified to bound potential plant conditions.

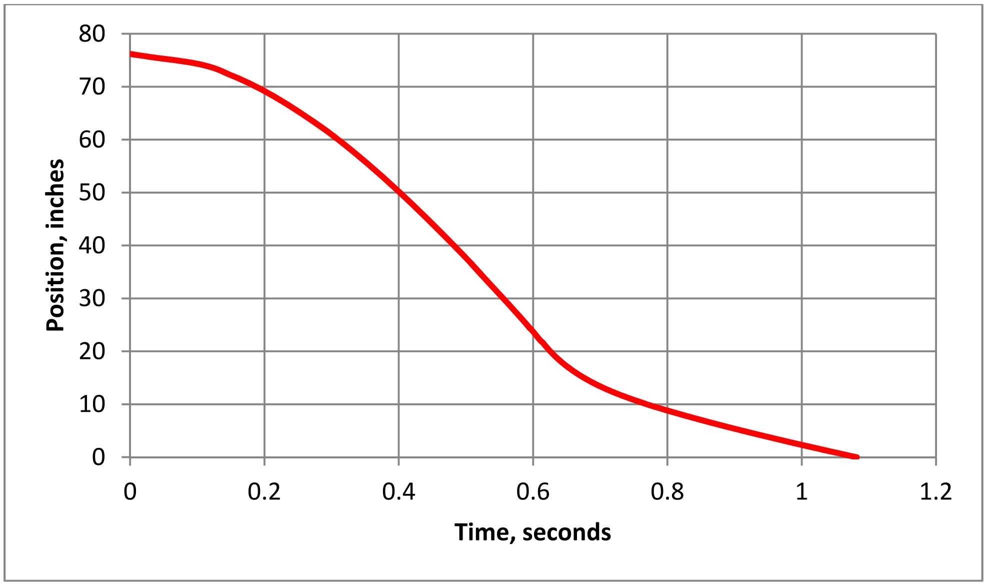

Figure 4-14 Control rod position versus time

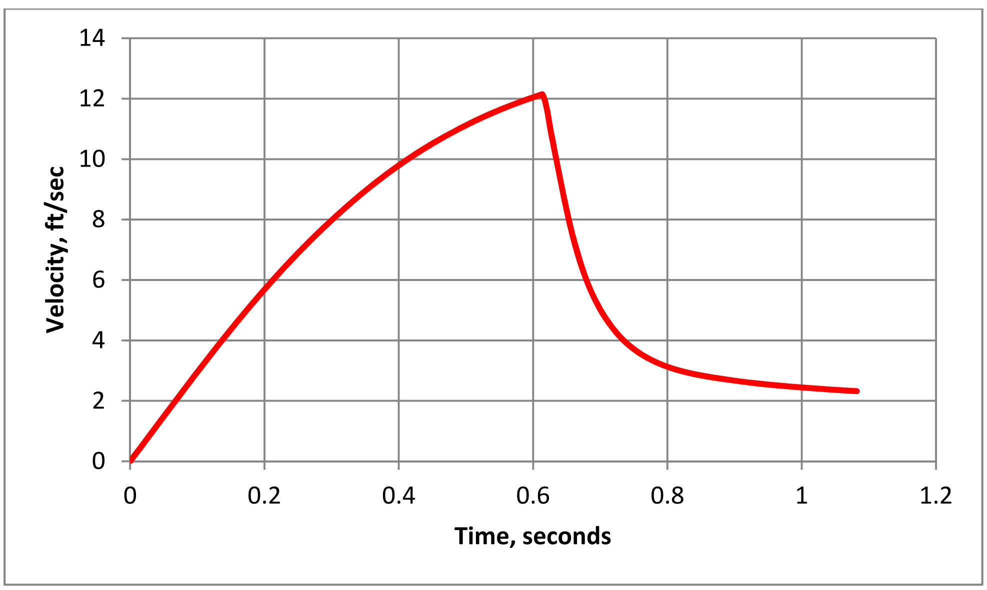

Figure 4-15 Control rod velocity versus time

5.0 Fuel Assembly Testing

5.1 Mechanical Testing Summary

A comprehensive test program was conducted at AREVA's Richland Test Facility to characterize the mechanical performance of the NuFuel-HTP2™ fuel design. The test results are used in the fuel assembly normal operation and seismic analyses to determine the acceptability of the design for in-reactor operation.

Prototypical fuel assemblies for beginning-of-life (BOL) and end-of-life (EOL) conditions were fabricated and mechanically tested. The BOL assembly spacer grids are in the as-fabricated BOL condition. The EOL assembly simulates the EOL conditions with grid cells relaxed and fuel rods seated on the bottom nozzle.

The fuel assembly characterization tests and their use in modeling and analysis are further described in Reference 9.1.5.

5.1.1 Fuel Assembly Lateral Load Deflection (Stiffness) Test (Beginning of Life and End of Life)

This test is performed to characterize the static, lateral structural response of the fuel assembly. The assembly is secured in prototypical upper and lower core plate interfaces. The test is performed by laterally deflecting the center of the test assembly at the second HTP™ spacer grid from the bottom to a displacement along one axis. The force required to deflect the assembly and the corresponding displacement are recorded continuously for the complete loading and unloading cycle.

5.1.2 Fuel Assembly Free Vibration (Lateral Pluck) Test (Beginning of Life and End of Life)

This test is performed to characterize the dynamic, lateral response of the fuel assembly over a large range of amplitudes. The assembly is secured in prototypical upper and lower core plate interfaces. The test is performed by laterally deflecting at the second HTP™ spacer grid from the bottom of the test fuel assembly to a given displacement, and obtaining the response of the assembly when the applied force is suddenly released. Deflection versus time is measured and is used to establish the fuel assembly first mode and damping.

5.1.3 Fuel Assembly Lateral Impact Test (Beginning of Life and End of Life)

This test is performed to characterize the dynamic, lateral impact behavior of the fuel assembly. The test is performed in two phases. In the first phase, the test consists of deflecting the fuel assembly to a given displacement at the second HTP™ spacer grid from the bottom and then suddenly releasing the load allowing it to impact on a baffle plate at the third spacer grid from the bottom location. In the second phase, the test consists of deflecting the fuel assembly to a given displacement at the third spacer grid location from the bottom and then suddenly releasing the load allowing it to impact on a baffle plate at the second spacer grid from the bottom location. The assembly response is obtained in the form of deflection versus time measured at two spacer elevations corresponding to the pull and impact locations. Instrumentation is also used to monitor out of plane movement and twist during the test. A load cell attached between the baffle plate and the support measures the test assembly impact force.

5.1.4 Fuel Assembly Lateral Forced Vibration Test (Beginning of Life and End of Life)

This test is used to characterize the dynamic, lateral response of the fuel assembly. This test complements the free vibration test by providing information on higher modes of the fuel assembly natural frequency, but is typically limited to smaller amplitudes than the free vibration test. This test is performed by applying a dynamic horizontal motion to the test assembly. The fuel assembly is installed on the seismic test stand, and secured in a prototypical support fixture, in order to achieve fixed-fixed end boundary conditions. The input is applied at the first and/or second intermediate spacer grids. These locations are selected in order to be able to excite all modes of interest. For each mode, the fuel assembly response is measured by accelerometers and/or displacement sensors attached at the HTP™ grid locations. **[ ]**, the evolution of frequency versus vibration amplitude is analyzed.

5.1.5 Fuel Assembly Axial Stiffness Test (Beginning of Life and End of Life)

This test is performed to characterize the static, axial structural response of the fuel assembly. The test is performed in the same fixture used for the free and forced vibration testing. The fuel assembly is secured at the top and bottom plates with a simulated core plate fixture. A jack screw is mounted between the simulated core plate and the upper support structure. A load cell is mounted between the lower support plate and the floor plate. The jack is used to apply the load, and the load cell measures the applied load. The axial deflection of the fuel assembly, under load, is measured at key locations with respect to a fixed reference. Instrumentation is also used to monitor any lateral movement of the fuel assembly.

5.1.6 Fuel Assembly Drop Test (Beginning of Life and End of Life)

This test is performed to characterize the dynamic, axial structural response of the fuel assembly. The test fuel assembly is suspended a specified distance above a plate attached to a load cell. The assembly is released and allowed to fall onto the plate and load cell. The displacement of the fuel assembly bottom nozzle is measured throughout the test.

Because the use of this data is to calibrate the impact behavior of the vertical seismic model, it is necessary to collect data on force and displacement as a function of time.

5.1.7 Spacer Grid Tests

The mechanical performance of the spacer grids were confirmed through a series of structural tests on prototype grids.

Dynamic crush tests are performed on HTP™ spacer grids at unirradiated and simulated-irradiated conditions. The tests determine the through-grid stiffness and damping values for the lateral seismic models and the crushing load limits for the grids.

The static crush characteristics (static stiffness and elastic load limit) are used to establish allowable grid clamping loads applied during shipping.

Grid slip load testing defines the grid-to-fuel rod slip load at non-irradiated (BOL) conditions for both the HTP™ and HMP™ grids. Grid slip load testing is not performed at irradiated (EOL) conditions because the fuel rods are not actively restrained in the grid. The slip load values are used in the fuel assembly evaluation.

5.1.8 Top and Bottom Nozzle Tests

Strength testing of the bottom nozzle is performed to establish the axial load limit for evaluation. A prototypical bottom nozzle is tested at room temperature in static axial compression by applying a load to 24 springs on the guide tube positions. The spring stiffness is set to be equal to the guide tubes stiffness in order to simulate the load distribution of the guide tubes.

A maximum room temperature test load is applied without collapse of the structure. This tested maximum load is used to demonstrate the structural adequacy in the design evaluation by comparison with the normal operating and faulted loads.

Strength testing of the top nozzle is also performed to establish the axial load limit for evaluation. A prototypical top nozzle is tested at room temperature in static axial compression by applying a load to top of the top nozzle, which is set on 24 springs at the guide tube positions. The spring stiffness is set to be equal to the guide tubes stiffness in order to simulate the real load distribution of the guide tubes.

A room temperature test load is applied that exceeds the design load and resulted in no plastic deformation of the structure. This tested maximum load is used to demonstrate the structural adequacy in the design evaluation by comparison with the shipping and handling, normal operating and faulted loads.

5.2 Thermal-Hydraulic Testing Summary

5.2.1 Pressure Drop and Liftoff Testing and Pressure Loss Coefficient Development

Pressure drop and liftoff testing was performed on a full-scale NuFuel-HTP2™ prototype fuel assembly in the PHTF at the AREVA NP Richland Test Facility. The testing configuration simulated the upper and lower core supports in the NuScale Power Module. The test data obtained from the pressure drop and liftoff testing are used to develop pressure loss coefficients for subsequent thermal-hydraulic and mechanical analyses.

The pressure loss coefficients for the spacer grids and the overall loss for the NuFuel-HTP2™ fuel assembly are dependent on the Reynolds number. The coolant flow in the

NuScale reactors is driven by natural circulation and at nominal flow conditions, the Reynolds number is approximately 80,000. Pressure drop data are reduced to Reynolds number dependent values and are used in the development of the pressure drop coefficient correlation. The pressure loss coefficients for assembly components and the overall fuel assembly are given in Table 5-1.

In addition to the pressure drop test, a hydraulic liftoff test was performed in the PHTF to acquire data to develop a correlation to be used in fuel assembly hydraulic lift analyses. The liftoff test is performed on a prototypical fuel assembly at six different temperatures for characterization over a range of Reynolds numbers. At each temperature, the flow is adjusted to obtain a conservative lift point, which is defined as the flow and temperature state at which the assembly is barely seated. Pressure drop measurements are taken at the conservative lift point and at a state point where the assembly lifts measurably. An overall pressure loss coefficient for use in hydraulic lift analyses is determined and is given in Table 5-1. This pressure loss coefficient incudes a correction for the assembly residual momentum factor based on the liftoff test results.

5.2.2 Flow-Induced Vibration Testing

A 1000-hour life and wear test was performed at the Richland PHTF to validate the fretting performance of the fuel assembly structure. The life and wear test is run for 1032 hours at a temperature of 300°F at or above the target Reynolds number of 52,000. At the conclusion of the test, **[ ]** fuel rods are examined for grid to rod fretting. The test results are summarized in Section 4.1.3.

Table 5-1 Pressure loss coefficients derived from testing

6.0 Control Rod Assembly

6.1 Control Rod Assembly Description

The NuScale CRA includes 24 individual control rods fastened to a one-piece cast stainless steel spider and coupling hub (see Figure 6-1). Table 6-1 provides major CRA design parameters. The top end of each individual control rod attaches to the spider by a nut, pin, and weld combination. Compared to AREVA's standard rod cluster control assembly (RCCA) design, the individual control rods are shortened to match the NuScale reactor core height, but retain the same basic design features and materials. AREVA's standard RCCA design has been implemented in twelve 17x17 PWRs in the United States, comprising over 600 individual assemblies.

The combination of the pin, nut, upper end plug and spider boss form a flex joint, which provides flexibility to accommodate potential misalignment between the CRA and fuel assembly guide tubes. The upper end plug has a reduced diameter shank and lower shoulder to provide lateral clearance with the interior diameter of the spider finger. The clearance allows for elastic deflection of the upper end plug for any misaligned control rod, fuel assembly, upper internals, or fuel handling equipment.

A preloaded helical spring is assembled into a skirt internal to the bottom of the spider hub and provides for energy absorption during a CRA trip. The spring is preloaded and maintained within the hub by a retaining ring and tension bolt. During a refueling outage or after a reactor trip, the spring retaining ring rests on the fuel assembly top nozzle. The CRA interfaces with the control rod drive mechanism coupling through a cavity at the top of the CRA spider (see Figure 6-2) that matches the male coupling dimensions on the drive shaft, similar to current designs in operation.

A 302 stainless steel plenum spring is used within the individual rods to restrain motion of the absorber materials within the cladding during shipping and handling. The absorber material is a combination of B_4C pellets and silver-indium-cadmium (Ag-In-Cd) bar. A stack support resides within the annulus of the lowermost Ag-In-Cd absorber to reduce compressive loads on the Ag-In-Cd, thus reducing thermal creep of the Ag-In-Cd during operation. The control rod cladding is 304 stainless steel tubing with stainless steel end plugs welded to each end, encapsulating the rod internals to complete the rod assemblies (see Figure 6-3).

The only design differences between the AREVA standard RCCA and the NuScale CRA are the length of the rod components, the cladding material, and spider spring and retainer modifications. AREVA operating experience has identified two life-limiting phenomena for control rod assemblies: cladding strain and cladding wear. Strain behavior of the NuScale CRA is similar given the diametral equivalence of the rods. The minor difference in cladding material should not be a significant factor in the allowable cladding strain. Cladding wear is expected to be acceptable on the NuScale CRA given the low axial flow rates of the NuScale reactor.

The changes to the spider spring and retainer design relative to the standard AREVA RCCA design are made to increase the spring preload and decrease the spring solid height. These changes allow the CRA spider to absorb the energy from a higher impact force than is experienced in typical PWR applicactions.

Control rod assembly materials exposed to reactor coolant are either low carbon stainless steels or Alloy 718, all of which are resistant to corrosion from reactor coolant exposure. These materials have been used extensively and successfully in operating PWRs. The B_4C and Ag-In-Cd absorber materials are encapsulated in a 304L stainless steel tube that is welded at both ends, which protects the absorbers from coolant interaction. Control rod integrity is confirmed by inspection during initial refueling outages. Table 6-2 identifies CRA component materials.

Table 6-1 Control rod design parameters

Parameter	Value
CRA total weight (lb)	43
CRA total height (inch)	94.37
Control rod length – short/medium/long (inch)	87.065 / 87.425 / 87.875
Control rod outer diameter (inch)	0.381
Control rod inner diameter (inch)	0.344
Control rod bottom end plug length (inch)	1.913
B_4C outer diameter (inch)	0.333
B_4C stack length (inch)	62.0
Ag-In-Cd outer diameter (inch)	0.336
Ag-In-Cd stack length (inch)	12.0
Height of CRA spider assembly (inch)	10.387
CRA shaft outer diameter (inch)	1.804

Table 6-2 Control rod assembly materials

Component	Material
Spider	304L stainless steel
Rod end plugs	308L stainless steel
Cladding	304L stainless steel
Solid spacer, lock pin, nuts, tension bolt	304L stainless steel
Spring retainer	17-4 PH stainless steel
Spider spring	Alloy 718
Control rod plenum spring	302 stainless steel
Absorber materials	80% Ag - 15% In - 5% Cd and B_4C
Stack support	Alloy X750

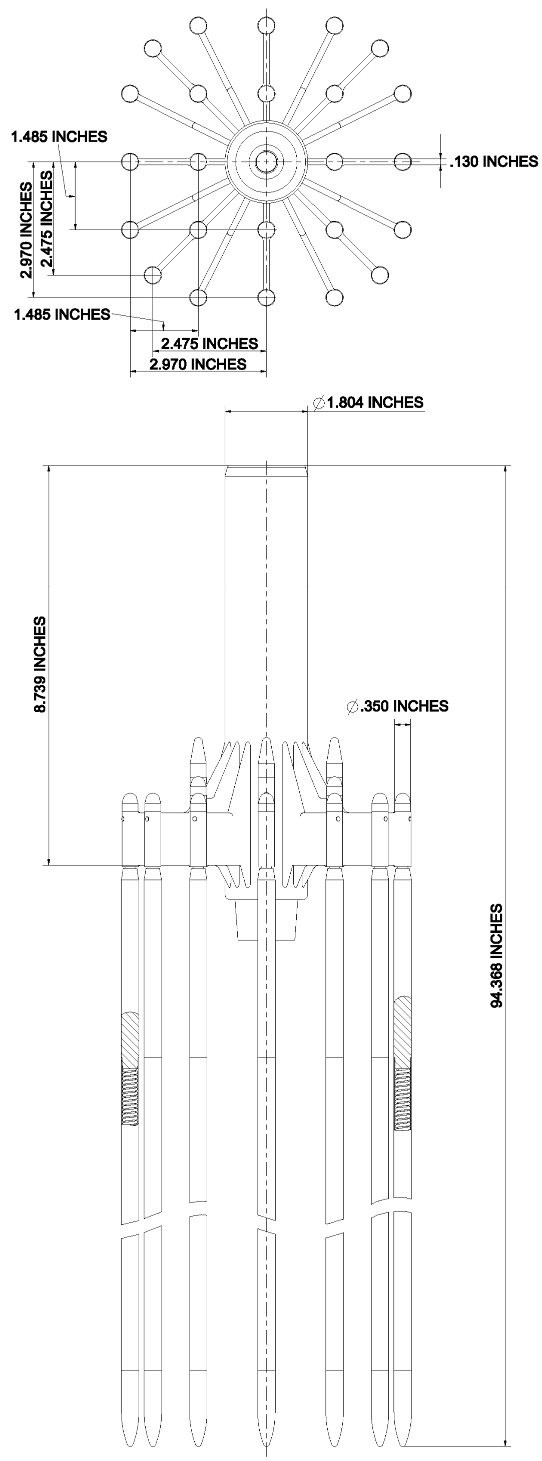

Figure 6-1 Control rod assembly general arrangement

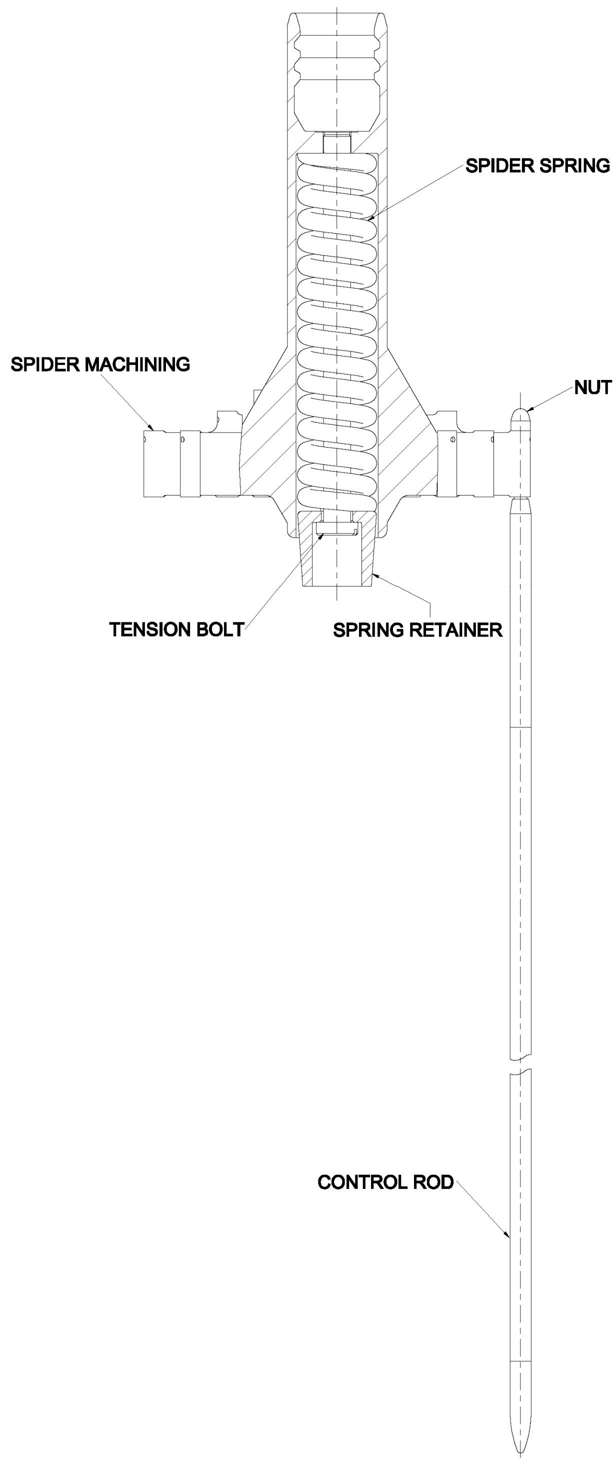

Figure 6-2   Control rod assembly cut-away

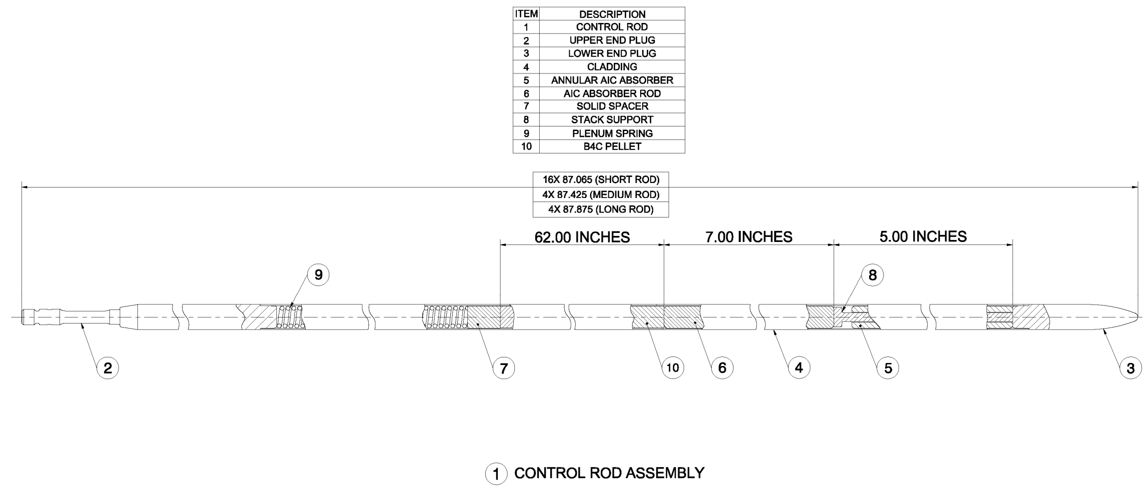

Figure 6-3 NuScale control rod design

6.2 Control Rod Assembly Evaluation

This section evaluates the CRA design against typical criteria to demonstrate acceptable performance under all conditions of operation over a 20 effective full power year (EFPY) design lifetime.

6.2.1 Cladding Strain

The potential for control rod cladding strain is primarily a result of swelling of the control rod absorber material caused by neutron fluence. The analysis considers elevation-specific fluence values and thermal expansion of the absorber and cladding. The calculated cladding and absorber temperatures are based on the flux predicted for various axial positions along the control rods. Volumetric swelling rates of the B_4C pellets and the Ag-In-Cd absorber are based on models benchmarked to measurements from in-reactor control components.

The strain calculation is performed at the following axial elevations:

- Bottom of the annular Ag-In-Cd bar, where the fluence is highest
- Bottom of the solid Ag-In-Cd bar
- Bottom of the B_4C pellet stack

Cladding strain is limited to **[ ]** percent to maintain ductility for irradiated 304L stainless steel cladding. The strain calculation determines that, for swelling rates corresponding to 20 EFPY of operation, the absorbers do not contact the cladding and the cladding strain limit is not exceeded.

6.2.2 Cladding Creep Collapse

Cladding creep collapse evaluations include short-term and long-term collapse analyses.

In the short-term collapse analysis, the differential pressure across the control rod cladding must not exceed the critical buckling pressure. Two critical buckling pressures are calculated:

- Bifurcation buckling pressure of a perfectly circular shell (P_{cr}), to confirm the elastic stability of the cladding
- Yield-point buckling pressure (P_{yp}) accounting for initial tubing ovality

The calculated buckling pressures at hot and cold conditions exceed with substantial margin the system pressure of 1850 psia and the maximum hydrostatic test pressure of 2640 psia respectively.

In the long-term collapse analysis, changes in cladding ovalization and cladding stress over time are predicted using the CROV creep ovalization code. The analysis assumes no cladding support by the control rod internals. Creep collapse is evaluated at the lowest tip of the control rod because this region experiences the largest fast flux.

The CROV analysis demonstrates that after 20 EFPY, the cladding ovality remains within acceptance limits, the cladding stress remains below the material yield strength, and the maximum cladding diameter as a result of ovalization is less than the inner diameter of the guide tube dashpot.

6.2.3 Cladding Stress

Control rod cladding stresses are categorized, calculated, and compared to service limits in accordance with the ASME code. Cladding stresses are calculated based on:

- Differential pressure
- Differential temperature
- Cladding ovality
- Loads from control rod drive mechanism stepping and reactor trip
- Bending due to misalignment
- Seismic conditions
- Flow-induced vibration
- Shipping and handling conditions (evaluated in isolation from other conditions)
- Stuck rod condition (evaluated in isolation from other conditions)

The design stress intensity, S_m, is 2/3 of the cladding yield strength **[** **]**. Table 6-3 defines the allowable stresses for each of the four ASME stress categories. The calculated stresses result in a minimum design margin of

1.47, which occurs for primary membrane stresses at cold conditions, where margin is defined as the allowable stress divided by the calculated stress.

Table 6-3 Control rod cladding allowable stresses

Stress Category	Temperature, °F	ASME Allowable Stress, psi
A: Primary membrane $P_m \le S_m$	70 650	**[ ]** **[ ]**
B: Primary membrane + bending $P_m + P_b \le 1.5S_m$	70 650	**[ ]** **[ ]**
C: Primary and secondary membrane + bending $P_m + P_b + Q \le 3.0S_m$	70 650	**[ ]** **[ ]**
D: Faulted $P_m \le S_m$ $P_m + P_b \le 2.25S_m$	70 650 70 650	**[ ]** **[ ]** **[ ]** **[ ]**

6.2.4 Cladding Fatigue

The control rod cladding is analyzed for fatigue from reactor trips, stepping loads, and flow-induced vibration bending loads, assuming no wear. The analysis conservatively assumes infinite cycles from flow-induced vibration and control rod stepping. The analysis concludes that the fatigue stress is below the endurance stress limit of the cladding material based on the ASME Code fatigue curve (12 ksi), and therefore fatigue failure does not occur.

6.2.5 Cladding Wear

Control rod cladding wear limits are determined by reducing the cladding wall thickness in mechanical analyses until the margin to acceptance limits is reduced to zero. This method is consistent with the industry approach for PWRs. Because of the potential for leaching of the B_4C pellets and subsequent impact on shutdown capability if the cladding barrier is breached, a **[ ]** reduction in the minimum cladding wall thickness is conservatively applied to calculate the wear limits. The calculated wear limits address circumferential wear and azimuthally localized wear.

The following limits are calculated:

- Maximum wear depth of **[ ]** inch, independent of geometry
- Minimum cross-sectional area of **[ ]** inch^2 remaining after wear (uniform circumferential wear)
- Minimum cross-sectional area of **[ ]** inch^2 remaining after wear (azimuthally localized wear, considering **[ ]**)

Wear limits are used in conjunction with wear rates specific to the NuScale design to determine an allowable wear-based design life. Prototype testing using a full-scale CRA is performed to measure CRA rod vibration and the susceptibility to wear. After initial irradiation and operation of the CRA design, inspections are performed so that actual rod wear rates can be compared with the predetermined wear limits to demonstrate acceptable performance.

The operating environment for the NuScale control rods is expected to be less severe with respect to rod wear than the environment typical of operating PWRs. Axial flow rates in the reactor core and in the guide tubes are significantly lower and cross flows at and above the fuel assembly top nozzles are very low (approximately **[ ]** feet/second). The absence of outlet flow nozzles in the upper internals reduces the cross flows compared to a typical PWR. These flow conditions create a more benign flow environment, reducing mechanical interactions with the guide cards and fuel assemblies. Based on this assessment, the CRA design lifetime is not expected to be limited by control rod wear.

6.2.6 Control Rod Internal Pressure

The control rod internal pressure analysis predicts the maximum internal rod pressure using a conservative model that calculates the depletion in the B_4C pellets and releases the helium to the rod plenum volume. The calculation includes helium backfill, residual, and sorbed gases in the determination of the final maximum internal pressure. The Ag-In-Cd material is not a source of gases.

During normal operation, the CRAs are positioned such that the B_4C pellets are located above the active fuel. Over the lifetime of the CRA, there is very low depletion, which results in insignificant helium production. In addition, the large porosity of the B_4C absorber material provides sufficient volume to accommodate any helium produced. However, the analysis conservatively assumes **[ ]** percent release and retention of the helium due to depletion of the B_4C pellets.

The helium generation is conservatively calculated based on the assumption that the lower portion of the B_4C pellet stack is subject to the fluence at the lower tip of the pellet stack. For a 20 EFPY CRA design life, the predicted ^{10}B depletion is **[ ]** percent over the lower portion of the pellet stack. The maximum rod internal pressure at EOL is predicted to be **[ ]** psia, which meets the criterion of being less than RCS pressure (1850 psia).

6.2.7 Component Melt Analysis

The control rod is analyzed to ensure that each component remains below the melt temperature. The analysis uses conservative values for heating rates and gap conductance. The worst case calculated temperatures for all rod components are well below the material melt limits.

6.2.8 Spider Assembly Structural Analysis

The spider assembly structural analysis evaluates the static and fatigue stresses in the CRA spider assembly and the control rod to spider connections. The following loads are analyzed:

- Reactor trip / CRA stepping
- Stuck rod
- Shipping and handling
- Hydraulic load during reactor trip

The following elements of the spider assembly are analyzed:

- Spider arm
- CRA spring
- Spring retainer
- Spring tension bolt
- Spring flange
- Spring housing
- Spider coupling splines
- Upper end plug / nut / spider connection

Calculated stresses are compared to ASME Code minimum material strength values. The results demonstrate positive margins for all components, validating the structural integrity of the spider assembly and connections during normal operation and seismic loads.

6.2.9 Control Rod Assembly Impact Velocity Limit

The kinetic energy absorption capacity of the CRA spring is analyzed to determine the maximum impact velocity of the CRA on the fuel assembly during a reactor trip. The NuScale design has a longer and heavier CRA driveline than is typical for PWRs and has significantly lower axial flow rates, resulting in a higher CRA impact velocity. The spider spring is designed to absorb all of the kinetic energy of the CRA during a reactor trip using the available spring retainer travel to prevent the CRA spider hub from impacting the fuel assembly top nozzle.

The maximum acceptable impact velocity is **[ ]** ft/sec. Section 4.4.3 describes a calculation performed to determine the impact velocity. The calculated velocity is less than the maximum acceptable velocity, justifying the spring design. Testing will confirm that the calculated maximum velocity is conservative.

6.3 Control Rod Assembly Testing

The NuScale CRA is similar to existing 17X17 control rod assemblies except for the shorter length. The CRA drive shaft is longer than typically used in the industry. Prototype testing will be performed to confirm CRA drop times, to assess the propensity for vibration wear, and to ensure that CRA insertion is not affected by the maximum expected misalignment of the fuel assembly guide tubes predicted to occur during a concurrent LOCA and seismic event.

7.0 Design Change Process

Reference 9.1.6 provides generic design criteria for fuel assembly designs and a process to demonstrate compliance. Compliance to these criteria is demonstrated by the following:

- documenting the fuel system and fuel assembly design drawings
- performing analyses with NRC-approved models and methods
- confirming the adequacy of significant new design features using prototype tests or lead test assemblies prior to full reload implementation
- continuing irradiation surveillance programs, including past irradiation examinations, to confirm fuel assembly performance
- using the quality assurance procedures, quality control inspection program, and design control requirements set forth in the NRC approved quality assurance program

The generic design change process described in Reference 9.1.6 and the specific design criteria in this report will be used to justify fuel design changes for the NuScale design without requiring NRC review and approval. Acceptable changes to the NuFuel-HTP2™ fuel design will meet all of the following conditions:

- The change does not result in an un-reviewed safety question.
- Changes in plant technical specifications are not required.
- The applicability of NRC-approved methodologies is demonstrated to be valid.
- Burnup limits are within those approved by the NRC.

Reference 9.1.11 contains examples of design changes that could be made using the change process defined in Reference 9.1.6 without NRC review and approval. NRC concurrence with this process is documented in Reference 9.1.12. The following list summarizes the examples provided in Reference 9.1.11. Both minor design changes and new fuel designs fall within the scope of the design change process such that as long as the criteria continue to be met, subject to the limitations above, design changes can be made without NRC review and approval.

Examples of minor design changes are:

- a change in the attachment of the spacer grid to the guide tubes
- a change in the strip thickness of the spacer grid
- a change in cladding thickness
- the first NuScale use of an assembly design feature previously irradiated in a different lattice
- a change in enrichment

- a change in gadolinia-bearing rod locations

Examples of design changes that constitute new fuel designs are:

- new cladding material
- a spacer grid with a new functional mixing behavior or new rod support mechanism
- a change that would alter the fuel behavior relative to the NRC-approved models, for example rod growth, assembly growth, or clad corrosion

8.0 Summary and Conclusions

This report describes the NuFuel-HTP2™ fuel assembly design and corresponding CRA design. The designs incorporate features with extensive operating experience and are evaluated using NRC-approved evaluation methods. The testing and design evaluations demonstrate that the designs meet regulatory criteria and will perform acceptably in the NuScale Power Module.

9.0 References

9.1 Source Documents

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

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

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

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

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

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

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

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

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

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

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

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

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

9.1.14 TR-0916-51502, "NuScale Power Module Seismic Analysis," January 2017.

9.1.15 TR-1016-51669, "NuScale Power Module Short-Term Transient Analysis," December 2016.

Enclosure 3:

Affidavit of Thomas A. Bergman, AF-0816-51127

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 method by which NuScale develops its fuel design.

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

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

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

(4) The information sought to be withheld is in the enclosed report entitled "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs". The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

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

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

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

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

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

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

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

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 6, 2017.

Thomas A. Bergman

Enclosure 4:

Affidavit of Nathan E. Hottle, AREVA, Inc.

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

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

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

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

3. I am familiar with the AREVA information contained in the following document: TR-0816-51127-P Revision 1, "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. for the control and protection of proprietary and confidential information.

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

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

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

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

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

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

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

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

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

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

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

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

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

Nathan C. Hottle

SUBSCRIBED before me this 7th

day of January, 2017.

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