NuScale Power

RAIO-1117-57232

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Docket No. 52-048

U.S. Nuclear Regulatory Commission
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**SUBJECT:** NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 18 (eRAI No. 8778) on the NuScale Design Certification Application

**REFERENCES:**
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 18 (eRAI No. 8778)," dated May 05, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 18 (eRAI No.8778)," dated July 05, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) supplemental response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Question from NRC eRAI No. 8778:

- 04.02-1

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8778

**Enclosure 1:**

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8778

# Response to Request for Additional Information
# Docket No. 52-048

**eRAI No.:** 8778
**Date of RAI Issue:** 05/05/2017

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**NRC Question No.:** 04.02-1

In accordance with 10 CFR 50 Appendix A GDC 10, "Reactor design," the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

To meet the requirements of GDC 10, as it relates to SAFDLs for normal operation, including AOOs, fuel system damage criteria should be included for all known damage mechanisms. Fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. When applicable, the fuel damage criteria should consider high burnup effects based on irradiated material properties data. Complete damage criteria should address, in part, that the cumulative number of strain fatigue cycles on the structural members of the fuel assembly (e.g. grids, guide tubes, thimbles, fuel rods, control rods, etc.) should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles; otherwise, other proposed limits must be justified.

The staff notes that in TR-0816-51127, Revision 1, "NuFuel-HTP2 Fuel and Control Rod Assembly Designs," the applicant has considered load following in the fuel rod fatigue analysis (see Table 4 -3), but no discussion is provided to justify the current thermal-mechanical models for load follow use. For example, it is unclear to the staff from the information provided if the fission gas release model was designed to model load following and was approved for this purpose. However, in FSAR Tier 2, Section 4.3.2.4.16, the applicant states that while power maneuvering operations within the capabilities of the rod control system are anticipated, continuous load following operation using the control rod assemblies is not anticipated. Based on the docketed information, the staff is unable to determine if the NuScale DCA requests approval for load following; therefore, the staff cannot determine if the fuel and control rod assembly designs have been adequately designed to incorporate fatigue effects from load following such that the requirements of GDC 10 are met.

1. Does NuScale request NRC approval for load follow (i.e. power maneuvering) use for the NuScale SMR design?
    a. If no, the staff requests the applicant to clearly identify in FSAR Section 4.2 that load

following will not be used.

b. If yes, the staff requests the applicant to clearly identify in FSAR Section 4.2 that load following will be used, describe the type of load following (e.g. daily load fo llow), and to justify the thermal–mechanical models and analysis for the NuScale fuel design for the requested load follow use. Additionally, the impacts of load follow operation on control rod nuclear lifetime (FSAR section 4.3) and initialization of postulated accident analyses (FSAR Section 15) should be addressed in their respective sections.

c. If the applicant intends to leave the choice for load following operations up to the COL holder, the staff requests the applicant to include an appropriate COL information item that discusses the information needed to be submitted by the COL applicant for NRC review.

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**NuScale Response:**

This response supersedes the original response to RAI 04.02-1 (eRAI 8778) in its entirety.

The NuScale reactor core is designed to perform normal power maneuvers with control rods and/or soluble boron concentration changes. All power maneuvers are performed within the limits of the Technical Specifications, thereby ensuring that the initial conditions for the safety analyses remain valid.

The analyses of the NuScale core in Chapter 4 are based on a representative equilibrium cycle. The inputs to the safety analyses presented in Chapter 15 of the Design Certification Application (DCA) were based on a cycle depletion of this equilibrium cycle that assumes constant power operation. As noted in the Nuclear Regulatory Commission (NRC) RAI Question 04.02-1 (eRAI No. 8778), the fuel rod cladding fatigue was analyzed for power maneuvering in TR-0816-51127, "NuFuel-HTP2 Fuel and Control Rod Assembly Designs."

In response to eRAI Number 8772 (NRC Question No. 04.03-1), NuScale clarified that Technical Specification 5.6.3 references a cycle specific Core Operating Limits Report (COLR), supported by a future topical report on Reload Safety Analysis Methodology. The COLR will be prepared for each cycle, based on the anticipated power maneuvering for the specific plant/module.

The Reload Safety Analysis Methodology topical report will include, or incorporate by reference, the evaluation methodology used to analyze and demonstrate any planned power maneuvering on a cycle-specific basis maintains the fuel within the specified acceptable fuel design limits. The COLR will describe the expected power maneuvering regime and provide limitations or augmentation factors, if any, associated with the assumed operating regime. The evaluation methodology for the power maneuvering is expected to address:

- Applicability of supporting codes and methods
- Alternate cycle depletion reflecting the assumed power maneuvers to capture effects on

burnup, burnup distribution, and power peaking
- Effect of power maneuvering on axial flux shapes and radial peaking
- Control Rod Assembly (CRA) absorber depletion
- Effect of power maneuvers and peaking on fission gas release, fuel melt limits, and transient strain limits
- Fatigue on fuel components (other than fuel rod cladding)
- Fuel assembly guide tube and CRA wear

FSAR Section 1.2 is revised to describe the power maneuvering capabilities of the NuScale Power Module. FSAR Sections 4.2, 4.3, and 4.4 are revised to clarify the power maneuvering analyses that will be performed as part of detailed cycle design.

**Impact on DCA:**

FSAR Sections 1.2, 4.2, 4.3, and 4.4 have been revised as described in the response above and as shown in the markup provided in this response.

- modularization to enable in-shop fabrication of reactor and containment components

#### 1.2.1.1.2 Operating Characteristics

The NPM is designed to operate up to full power conditions using natural circulation as the means of providing reactor coolant flow, eliminating the need for reactor coolant pumps.

The NPMs are partially immersed in a reactor pool and protected by passive safety systems. Each NPM has a dedicated emergency core cooling system (ECCS) and decay heat removal system (DHRS).

Important features of the NPM include the following:

- a small, modular design
- an integral pressurized water reactor (PWR) NSSS that combines the reactor core, SGs, and pressurizer within the RPV, eliminating the need for external piping to connect the SGs and pressurizer to the RPV
- natural circulation provides the driving force for reactor coolant flow, eliminating the need for reactor coolant pumps
- an RPV housed in a steel containment partially immersed in water, providing an effective passive heat sink for long-term decay heat removal
- a steel containment operated at a vacuum, eliminating the need for insulation on the RPV
- passive safety systems that are not reliant on electrical power

Table 1.2-1 presents the overall characteristics of the NuScale Power Plant.

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The NuScale power module is designed to perform normal power maneuvers. Electric power can be adjusted with turbine bypass to the condenser. In addition, core power maneuvering can be accomplished with control rods, soluble boron concentration changes, or a combination of control rods and soluble boron as described in Sections 4.2 and 4.3.

**Nuclear Steam Supply System**

The NSSS consists of a reactor core, two helical-coil SGs, and a pressurizer integrated within the RPV. The RPV is enclosed in an approximately cylindrical CNV that sits in the reactor pool. The reactor core is located below the helical-coil SGs inside the RPV. Using natural circulation, the primary reactor coolant flow path is upward through the central hot leg riser, and then downward around the outside of the SG tubes with return flow to the bottom of the core via an annular downcomer. As the reactor coolant flows across the SG tubes, heat is transferred to the secondary side fluid inside the SG tubes. Concurrently, as the secondary side

## 4.2 Fuel System Design

The NuScale fuel system is designed to satisfy the following criteria:

- The fuel system will not be damaged as a result of normal operation and anticipated operational occurrences (AOOs) [General Design Criterion (GDC) 10].
- Fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when it is required [Principal Design Criteria (PDC) 27].
- Core coolability is always maintained, even after postulated accidents (PDC 35 and 10 CFR 50.34).
- The number of fuel rod failures is not underestimated for postulated accidents (10 CFR 100).

The NuScale fuel assembly design features are similar to those of existing pressurized water reactor (PWR) 17x17 fuel assemblies. The only significant difference between the NuScale fuel design and other PWRs is the shorter fuel assembly length. The effect of the length on fuel design analyses is tested in full-scale fuel assembly tests and analyzed using NRC-approved methods. The results summarized in this section show that the NuScale fuel design, with its shorter length, demonstrates acceptable fuel performance consistent with the other proven features that make up the NuScale fuel design. Because this is a new application of a proven fuel design, post-irradiation inspection will be performed during the initial three cycles of operation of the first licensed module, as described in Section 4.2.4.6.

Section 4.2.1 presents the design bases for the cladding, fuel material, fuel rod performance, spacer grids, fuel assembly structural design, control rod assembly (CRA), and the surveillance programs that will confirm the adequacy of the design. Section 4.2.2 provides a detailed description of the fuel, the components that comprise the fuel assembly, and the CRA. Section 4.2.3 provides the detailed design evaluation that demonstrates how the design bases are met. Section 4.2.4 discusses fuel and CRA testing and inspection.

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The fuel system design evaluations include a fuel rod cladding fatigue analysis that considers power maneuvering in Table 4-3 of Reference 4.2-1. Additional analysis will be performed as part of the detailed design of the cycle-specific core using the methodologies described in Technical Specification 5.6.3. This analysis will include the effects of power maneuvers on CRA absorber depletion, fission gas release, fuel melt limits, transient strain limits, fuel rod components (other than cladding), and guide tube and CRA wear.

### 4.2.1 Design Bases

The fuel rod and fuel assembly design bases establish the performance requirements and damage criteria to satisfy the criteria in Section 4.2 of the NUREG-0800 Standard Review Plan.

The fuel assembly structural integrity is assured by setting limits on stresses and deformations due to various loads and by preventing the assembly structure from interfering with the functioning of other components. Three types of loads are considered:

- non-operational loads, such as those due to shipping and handling

- fluctuation in boron concentration, coolant temperature, or xenon.
- reactivity changes from load changes.
- design basis events with a stuck rod.
- long term shutdown capability.

CRA insertion is restricted to ensure that there is sufficient negative reactivity available to maintain shutdown capability and to limit the amount of reactivity insertion possible during the rod ejection accident. The PDILs are set sufficiently high to meet these criteria while also being low enough that operators have a reasonable range of CRA movement for power maneuvers. In general, deeper CRA insertion is allowed at lower power levels. Operators are alerted if the PDIL is approached. The PDILs are shown in Figure 4.3-2.

Power distribution, rod ejection, and CRA misoperation analyses are based on the arrangement of CRAs shown in Figure 4.3-18.

During a startup, the shutdown bank is withdrawn before the regulating bank withdrawal is initiated. The approach to criticality is initiated by a combination of boron dilution to the appropriate boron concentration and withdrawal of the regulating bank. Additional detail on startup is provided in Section 14.2.

#### 4.3.2.4.13 Burnable Poisons

Gadolinia ($Gd_2O_3$) is used as an integral burnable absorber in selected fuel rods to provide partial control of the excess reactivity available during the cycle. In addition, the burnable absorber also reduces the requirement for soluble boron at the beginning of the cycle, eliminating the possibility of a positive MTC during power operations at the beginning of the cycle (at power). In addition, burnable absorbers reduce power peaking within the fuel assembly.

#### 4.3.2.4.14 Peak Xenon

Startup from the peak xenon condition is accomplished using CRAs and boron dilution.

#### 4.3.2.4.15 Burnup

Cycles are designed with excess reactivity to offset the effect of burnup during the cycle. Control of this excess reactivity is accomplished using soluble boron and burnable poison. The boron concentration is limited during operating conditions to maintain the MTC negative. The end of a fuel cycle is reached when the soluble boron concentration approaches 20 ppm or less as discussed in Section 4.3.1.1.

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#### 4.3.2.4.16 ~~Load Follow~~Power Maneuvering

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Power changes are normally accomplished by means of regulating CRA position or soluble boron concentration. The CRA position is limited by the PDIL and helps maintain the core within the axial offset limits. While power maneuvering operations within the capabilities of the rod control system are anticipated to support power system demands, continuous ~~load following~~power maneuvering ~~operation~~ of the NuScale Power Module where the CRA position is used to achieve power changes ~~is not anticipated.~~was not assumed in the analysis of the representative equilibrium cycle. However, planned power maneuvers will be considered as part of a cycle-specific core design using the methodologies described in Technical Specification 5.6.3. The analysis of the impact of power maneuvering on the nuclear design will include the effects on axial and radial flux shapes that are used in the safety analysis in Chapter 15.

### 4.3.2.5 Control Rod Patterns and Reactivity Worth

The NuScale reactor module design utilizes 16 of the possible 37 assembly locations for CRAs. There are two CRA banks, a regulating bank and a shutdown bank. The regulating bank contains two groups of four CRAs each. The shutdown bank contains two groups of four CRAs each. Figure 4.3-18 shows the location of the CRA banks. Additional details on the CRAs are provided in Section 4.2 and Section 4.6.

The SDM is the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all CRAs (shutdown and regulating banks) are fully inserted (while accounting for the power defect, CRAs being at the PDIL, SDM uncertainties, and flux redistribution), except for the single CRA of highest reactivity worth, which is assumed to remain fully withdrawn.

The design limit on minimum SDM is set by the safety analysis for all power levels (including HZP) and operating modes. The limit assures that there is sufficient negative reactivity following a reactor trip, under all credible operating conditions, to shut the reactor down and prevent exceeding the specified acceptable fuel design limits.

Allowable deviations due to misaligned CRAs are controlled by the technical specifications. The allowance for CRA misalignment is based on the assumed uncertainty in the rod position indication (RPI) system. The nominal RPI system performance requirement is that the position is measured within three steps (out of 224 total); the acceptable analytical CRA misalignment is six steps and accounts for abnormal operating conditions such as a failure of half the sensor coils in the system or failure of an AC power source to the RPI system.

The reactivity insertion during a reactor trip is determined from the CRA drop time and differential reactivity worth versus CRA position. The CRA position versus time of travel after rod release is provided in Figure 4.3-23. This curve is based on a calculation described in Section 4.2. This curve is considered to be a realistic, and yet slightly conservative, rod drop time. A more conservative bounding CRA drop time is used in the Chapter 15 analyses. The reactivity worth versus CRA position is calculated by a series of steady-state calculations at various CRA positions, assuming the CRAs are at the PDIL as the initial position in order to minimize the initial reactivity insertion rate.

#### 4.4.3.3.1 Flow Stability Exclusion Regions

The NuScale flow stability protection solution uses a regional exclusion solution as described in the "Evaluation Methodology for Stability Analysis of the NuScale Power Module" topical report (Reference 4.4-4). The region is defined by a single point specifying riser subcooling margin. The stability exclusion region is protected by automatic MPS protective action.

Section 15.9.2 describes stability analysis application methodology using the PIM computer code. The methodology specifies the type and scope of the generic analysis used to define the exclusion region as well as the margins and the analytical limit for the reactor protection trips required to prevent unstable flow oscillations.

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### 4.4.3.4 ~~Load-Following~~Power-Maneuvering Characteristics

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While power maneuvering operations within the capability of the rod control system are anticipated to support power demands, continuous ~~load following~~ power ~~operation~~maneuvering ~~is not anticipated~~was not assumed in the analysis of the reference equilibrium cycle, as indicated in Section 4.3. However, planned continuous power maneuvering will be considered as part of a cycle-specific core design using the methodologies described in Technical Specification 5.6.3. The limiting axial flux shape that is described in Section 4.4.4.3 will include the impact of planned power maneuvering. Power control is accomplished using boron control and control rod positioning.

Section 4.3.2 describes the analysis used to generate the wide range of normal operation axial power shapes used to establish operating limits for normal steady state and power control operations. These limiting power distributions are controlled during operation by technical specifications that require operation within the axial offset (AO) window and within the power dependent insertion limits (PDILs).

The fixed in-core flux measurements and resulting power distribution continuously displayed in the control room and provide further assurance that the power distributions both axially and radially are not exceeded. Operation outside these limits is not allowed by the plant technical specifications.

### 4.4.3.5 Thermal and Hydraulic Characteristics Summary Table

Reference 4.4-1 summarizes the thermal-hydraulic characteristics of the NPM.

## 4.4.4 Evaluation

Conformance to GDC 10 requirements is demonstrated by establishing SAFDLs and ensuring that the plant stays within the SAFDLs. These limits ensure that the fuel clad is not breached (and thus this fission product barrier remains intact), that fuel system dimensions