July 06, 2017

Docket: PROJ0769

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 14 (eRAI No. 8807) on the NuScale Topical Report TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification," Revision 0

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 14 (eRAI No. 8807)," dated May 8, 2017.
2. NuScale Topical Report, "Nuclear Analysis Codes and Methods Qualification," TR-0616-48793, Revision 0, dated August 2016.

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

The Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 8807:

- 29739
- 29749
- 29750
- 29752
- 29754

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 14 (eRAI No. 8807). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses 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,

Jennie Wike
Manager, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, TWFN-6E55
Samuel Lee, NRC, TWFN-6C20
Bruce Bavol, NRC, TWFN-6C20

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8807, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8807, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0717-54772

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8807, proprietary

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 8807, nonproprietary

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8807
Date of RAI Issue: 05/08/2017

NRC Question No.: 29739

Title 10 of the Code of Federal Regulations (10 CFR) Part 52, Section 47 and Section 79 require a final safety analysis report (FSAR) to analyze the design and performance of the structures, systems, and components (SSCs). Safety evaluations, performed to support the FSAR, require reactor physics parameters to determine reactor core performance under normal operations, including anticipated operational occurrences, and accident conditions. An approved nuclear analysis methodology is utilized to provide reactor physics parameters for use in safety evaluations. Additionally, the nuclear analysis methodology is used to establish a partial basis for demonstrating compliance with the following general design criteria (GDCs) of 10 CFR Part 50, Appendix A:

GDC 10, Reactor design, which requires that the reactor core and associated coolant, control, and protection systems 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.

GDC 11, Reactor inherent protection, which requires that the reactor core and associated coolant systems be design so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

GDC 12, Suppression of reactor power oscillations, which requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations which can result sin conditions exceeding specified acceptable fuel design limits (SAFDLs) are not possible or can be reliably and readily detected and suppressed.

GDC 26, Reactivity control system redundancy and capability, which requires, in part, that the control rods be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for stuck rods, SAFDLs are not exceeded

GDC 27, Combined reactivity control systems capability, which requires that the reactivity control systems be designed to have a combined capability of reliably controlling reactivity

changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

The design description and analyses presented in TR-0616-48793, "Nuclear Analysis Codes and Method Qualification," are presented for a single NuScale Power Module (NPM). Due to the presence of several NPMs being in relatively close proximity to each other, and the location of the ex-core detectors for a single NPM, NRC staff is questioning whether multi-module effects need to be considered. Because the methodology presented in TR-0616-48793 is focused on a single NPM, NRC staff needs to establish a finding that multi-module effects do not need to be considered for the nuclear design and analysis of a NPM. Accordingly, NRC staff requests that NuScale provide evidence to show that multi-module effects can be neglected in the nuclear design and analysis of a NPM.

NuScale Response:

The reactor core design and analysis are independent of neighboring modules. The methodology described in the Nuclear Analysis Codes and Methods Qualification topical report, TR-0616-48793, is applicable to design and analysis of a single reactor core and is performed independently of the plant design, including the presence of additional modules. Similar to existing multi-unit plants, the design and analysis of a single reactor core using software such as the CMS5 suite is independent of neighboring units. The methodology assumes there are no neutronic interferences from neighboring NuScale Power Modules (NPMs) that would affect the design and analysis of the reactor core. The topical report has been revised to reflect this assumption.

A conservative analysis was performed to determine the attenuation (reduction of neutron flux) provided by the barriers between modules. Each individual NPM is housed in an isolated reactor bay within the reactor pool. Adjacent NPMs are separated by a concrete barrier several feet thick with several feet of borated pool water on either side between the concrete barrier and the containment vessel of the neighboring NPMs. For a reactor operating at 100% power, the neutron flux is attenuated such that the neutron flux seen by a neighboring module at the exterior of the containment vessel is approximately 5 orders of magnitude below the neutron flux level associated with startup of a module. Across the pool, the nearest opposite NPM is separated by a large expanse of borated pool water through which the flux is reduced such that the neutron flux from a reactor operating at 100% power seen by a NPM opposite the reactor pool at the exterior of the containment vessel is essentially zero. The magnitude of the reduction in neutron flux caused by the barriers between modules ensures that the neutronic effects due to a neighboring module are negligible.

Impact on Topical Report:
Topical Report TR-0616-48793, Nuclear Analysis Codes and Methods Qualification, has been revised as described in the response above and as shown in the markups provided in this response.

8.0 Application

NuScale intends to use the CMS5 code suite to perform nuclear analysis for core design, input to safety analysis, startup physics testing, core follow predictions, and operations support. The details of these applications are described in the following subsections. This topical report is also intended for use by Combined License applicants and licensees to implement core design methodology for their safety analysis calculations and operational support.

The methodology described in this topical report is applicable to the design and analysis of a single core, independent of the presence of additional NPMs (cores). Analysis that considers the plant design, including the presence and location of multiple modules, will be performed by a design certification or COL applicant to confirm that the application of this methodology is not affected by the flux from another operating NPM.

8.1 NuScale Reactor Core Design Methodology

CMS5 is used to determine loading patterns to meet the NuScale energy requirement and calculate core physics parameters that meet design limits to ensure the core will meet safety analysis requirements.

8.1.1 NuScale Fuel Rod and Assembly Lattice Configuration

To meet design constraints, core designs may employ fuel rod and assembly enrichment loading schemes that control the radial and axial power distribution in the core and are designed to help limit power peaking.

NuScale loading pattern designs may include assembly radial enrichment zoning to lower the pin-to-box ratio, and help reduce the overall radial peaking factor in the core. Assembly radial enrichment zoning consists of placing fuel rods of different enrichments in specific lattice locations within an assembly. The fuel rods containing the lower enrichment are generally placed in the high flux regions of the lattice that may include the corners, along the outside edge of the assembly, or around the instrumentation or guide tubes.

Fuel rod axial enrichment zoning is the use of different enrichments within the pellet stack of a fuel rod. Fuel rod axial enrichments are utilized to lower the axial leakage and to shape the axial power profile (reduce axial peaking). “Blankets” are areas of lower pellet enrichment placed at the top and bottom of a pellet stack to reduce leakage. These blankets, in effect, act as a reflector and improve fuel utilization. “Cutback” regions are areas in a BP fuel rod that do not contain BP. The cutback region does not necessarily contain a reduction in the ^{235}U enrichment as is done for the region containing the BP (Section 2.4). Cutback regions are used to shape the axial power shape of fuel rods containing BP. The NuScale reactor core may employ blankets and cutback regions in the fuel.

Cross sections for all assembly lattice configurations and fuel rod loading configurations necessary for the SIMULATE5 model are generated in CASMO5.

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Response to Request for Additional Information
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eRAI No.: 8807
Date of RAI Issue: 05/08/2017

NRC Question No.: 29749

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GDC 10, Reactor design, which requires that the reactor core and associated coolant, control, and protection systems 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.

GDC 11, Reactor inherent protection, which requires that the reactor core and associated coolant systems be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

GDC 12, Suppression of reactor power oscillations, which requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits (SAFDLs) are not possible or can be reliably and readily detected and suppressed.

GDC 26, Reactivity control system redundancy and capability, which requires, in part, that the control rods be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for stuck rods, SAFDLs are not exceeded.

GDC 27, Combined reactivity control systems capability, which requires that the reactivity control systems be designed to have a combined capability of reliably controlling reactivity

changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

In order to satisfy these requirements, NuScale imposes limits on key physics parameters (nuclear reliability factors, or NRFs) to ensure that core characteristics will not exceed calculated limits imposed as initial conditions for the transients of interest.

The discussions of the nuclear reliability factor update methodology in Sections 7.1.2, 7.2.2, 7.3.2 7.4.2, 7.5.2, 7.6.2, 7.7.2, 7.8.2, and 7.9.2 of TR-0616-48793 state that the, "... NRFs will be updated with sufficient measured data when a sufficient minimum number of measurements for acceptable statistics (a minimum of 10) are collected." This language is common across each of the NRFs. This statement is causing NRC staff to question the criteria for acceptable statistics and why a minimum of 10 measurements is sufficient. NRC staff needs to establish a finding that the NRF update methodology provides suitably conservative values for the associated NRFs. Accordingly, NRC staff requests that NuScale provide the criteria (e.g., tolerance limit, confidence level) that are used to determine the NRF and demonstrate how a minimum of 10 samples is sufficient to satisfy these criteria. In addition, NuScale should clarify whether the number of measurements is the only criteria necessary (for instance, are there additional quality standards on the data) to proceed to revise a given NRF.

NuScale Response:

In the Nuclear Analysis Codes and Methods Qualification topical report, TR-0616-48793, the choice of a minimum of 10 data points is an effort to standardize the number of data points across all the NRFs to effectively establish biases and uncertainties based on startup testing and core follow data specific to the NuScale design. The 10 data point criterion is a minimum, provided the tolerance limit and confidence level requirements discussed below are met. The approach described is based on data with a normal distribution and no bias, but the same minimum criterion is also used for non-normal distributions.

The generation of startup testing and core follow data for the validation and update of the biases and uncertainties will follow the normal startup testing and core follow quality standards established for nuclear power plants in accordance with Regulatory Guide 1.68. As described in Section 1.1 and shown in the flow chart in Figure 1-1 of the topical report, a base set of conservative NRFs are derived in the topical report and once NuScale-specific startup test and core follow data are available, the NRFs will be updated. The process that will be used to establish updated NuScale NRFs is identical to that described for the operating plants in Section 6.0 of this topical report.

Fundamentally, NRFs will be updated when sufficient NuScale operating data exists to change them. When NRFs are updated, they will be calculated to a 95/95 confidence/probability interval following the statistical methodology described in Section 4.0.

Equation 1 from Section 4.3.1 of the topical report provides the lower or upper one-sided tolerance limit for a normal distribution:

$$T = \bar{x} + k\sigma$$

Where:

T = tolerance limit

$\bar{x}$ = mean of the observations (bias)

σ = standard deviation of observations

k = tolerance factor for one-sided tolerance limit

Rearranging and discarding the mean for a normal distribution, Equation 1 becomes:

$$k = T / \sigma$$

Equations 2, 3, and 4 from Section 4.3.1 determine the k-factor value from the critical values for confidence and proportion, and number of observations. Therefore, Equations 1 through 4 allow for the determination of the number of observations necessary to produce a desired tolerance limit at a 95/95 confidence/probability level given a known standard deviation.

Although 10 data points is selected as a minimum number of observations necessary to update NRFs, not all NRFs will be updated with only 10 data points. The number of data points will depend on the tolerance band desired and standard deviation determined from the observations. A minimum of 10 data points is chosen to standardize the data points necessary for collection across all NRFs and provide a value for the k-factor that is not overly conservative, assuming the standard deviation of the observations does not vary greatly relative to the number of measurements taken.

Impact on Topical Report:There are no impacts to the Topical Report as a result of this response.

Response to Request for Additional Information
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eRAI No.: 8807
Date of RAI Issue: 05/08/2017

NRC Question No.: 29750

Title 10 of the Code of Federal Regulations (10 CFR) Part 52, Section 47 and Section 79 require a final safety analysis report (FSAR) to analyze the design and performance of the structures, systems, and components (SSCs). Safety evaluations, performed to support the FSAR, require reactor physics parameters to determine reactor core performance under normal operations, including anticipated operational occurrences, and accident conditions. An approved nuclear analysis methodology is utilized to provide reactor physics parameters for use in safety evaluations. Additionally, the nuclear analysis methodology is used to establish a partial basis for demonstrating compliance with the following general design criteria (GDCs) of 10 CFR Part 50, Appendix A:

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changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

The discussion of the nuclear reliability factor (NRF) for kinetics parameters in Section 7.9.1 of TR-0616-48793 states, “Because the neutron lifetime is proportional to the soluble boron worth, the NRFs are taken to be the same as that for the [differential boron worth] DBW.” NRC staff is questioning the underlying basis of this statement. NRC staff needs to establish a finding that the NRFs for the kinetics parameters are suitably conservative. Accordingly, NRC staff requests that NuScale provide evidence that the NRF for differential boron worth is applicable to the kinetics parameters.

NuScale Response:

Kinetics parameter uncertainties including neutron lifetime, delayed neutron fraction, and decay constants are not measured and are difficult to directly evaluate. Therefore, uncertainties in the kinetics parameters are determined from the relationship between the prompt neutron lifetime and differential boron worth (DBW). The DBW is the reactivity change associated with a change in the boron concentration of the water moderator. A simple but accurate way to determine the prompt neutron lifetime is from the insertion of a small amount of 1/*v* absorber (e.g. soluble boron) into the core, resulting in a negative reactivity insertion. The reactivity change after insertion of the 1/*v* absorber into the core is:

Equation 1:

$$\rho = \frac{k_{ref} - k_p}{k_p}$$

where:

ρ is the negative excess reactivity,

k_{ref} is the reference eigenvalue (before 1/*v* absorber insertion), and

k_p is the eigenvalue after insertion.

The prompt neutron lifetime after the insertion of a 1/*v* absorber is [Reference 1]:

Equation 2:

$$l' = \frac{1}{N_a \sigma_a v}\left(\frac{k_{ref} - k_p}{k_p}\right)$$

where:

N_a is atomic density of the 1/*v* absorber (atoms/b-cm),

σ_a is the thermal neutron absorption cross section of the absorber (b),

v is the speed of a thermal neutron (cm/s), and

l' is the neutron lifetime (s) when N atoms/b-cm of a 1/*v* absorber is added to the system.

Soluble boron is a nearly perfect 1/*v* absorber, so the insertion of soluble boron follows the relationship shown in Equation 2. Although the kinetics parameters will not be measured directly, substitution of Equation 1 into Equation 2 demonstrates the proportionality of neutron lifetime to DBW.

Section 5.1.4.2.1 of the Nuclear Analysis Codes and Methods Qualification topical report, TR-0616-48793, presents CMS5 to MCNP6 comparisons of DBW. The results provided in Table 5-4 of the topical report show an upper tolerance limit of {{ }}$^{2(a),(c)}$, and a lower tolerance limit of {{ }}$^{2(a),(c)}$ for CMS5 predictions against MCNP6. Also, Section 6.1.1.2 of the topical report presents a comparison of CMS5 DBW predictions against reported measurements for TMI-1 Cycles 1 and 2. Four measurements were taken and CMS5 predictions are in very good agreement with measured results.

Given the ability of CMS5 to accurately predict DBW and the relationship provided between prompt neutron lifetime and DBW, adopting the differential boron worth NRF ({{ }}$^{2(a),(c)}$) for the neutron lifetime uncertainty is reasonable and conservative due to the fact that the DBW NRF is {{ }}$^{2(a),(c)}$ the bounds of the largest magnitude tolerance limit (Table 7-2 of the topical report).

The effective delayed neutron fraction (β_{eff}) in CASMO5 uses a conventional six delayed neutron group approach based on basic nuclear data ($\beta_{m,i}$ and $\lambda_{m,i}$) for each fissioning nuclide *m* and delayed neutron group *i*. The data are weighted by nuclide fission rate and by energy group importance using the cell average adjoint flux in each energy group. The CASMO5 delayed neutron fraction data is passed to SIMULATE5 through the CMSLINK5 cross section library.

The uncertainty in the calculation of β_{eff} primarily arises from the following:

1. Uncertainty due to basic delayed neutron constants
2. Uncertainty due to delayed neutron spectrum
3. Uncertainty due to energy dependence of delayed neutron yield
4. Uncertainties of the fission cross-sections

The total uncertainty in the calculation of β_{eff} from these components has been shown to be about 5% [Reference 2, 3]. This result aligns with the uncertainty historically accepted by the NRC for the calculation of β_{eff} by CASMO [References 4, 5, 6, 7].

As noted in Section 7.9.1 of the topical report, no uncertainty is applied to β_{eff} when it is associated with other reactivity parameters where a separate uncertainty is applied (i.e. rod worth). Uncertainty is only applied to β_{eff} when it is used in an independent manner, not associated with the uncertainty in another reactivity parameter. When uncertainty is not applied through another reactivity parameter, a {{   }}[2(a),(c)] uncertainty on β_{eff} is reasonable and conservative given the uncertainties associated with the calculation of β_{eff} and the historical acceptance of the CASMO calculation uncertainty.

References:

1. Bretscher, M.M, "Evaluation of Reactor Kinetic Parameters Without The Need For Perturbation Codes," Argonne National Laboratory under DOE contract # W-31-109-ENG-38, Presented at the 1997 International Meeting on Reduced Enrichment for Research and Test Reactors, October 5-10, 1997.
2. Hammer, Ph., Tuttle, R.J., 1979. Proc. IAEA Consultants Meeting on Delayed Neutron Properties, Vienna, p. 277.
3. Zukeran, et al, Evaluation Method for Uncertainty of Effective Delayed Neutron Fraction Beff*, JAERI Conference 1999.
4. Virginia Electric and Power Company, DOM-NAF-1-Rev. 0.0-A, "Qualification of the Studsvik Core Management System Rector Physics Methods for Application to North Anna and Surry Power Stations", June 2003. (ML021710790)
5. Nuclear Regulatory Commission, Safety Evaluation Report for DOM-NAF-1-Rev. 0.0. (ML030700038)
6. Northern States Power Company, NSPNAD-8101, Rev. 2, "Qualification of Reactor Physics Methods for Application to Prairie Island", December 1999. (ML003673823)
7. Nuclear Regulatory Commission, Safety Evaluation Report for NSPNAD-8101. (ML003749539)

Impact on Topical Report:
There are no impacts to the Topical Report as a result of this response.

Response to Request for Additional Information
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NRC Question No.: 29752

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GDC 27, Combined reactivity control systems capability, which requires that the reactivity control systems be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

The discussions of the nuclear reliability factor (NRF) update methodology in Sections 7.1.2, 7.2.2, 7.3.2 7.4.2, 7.5.2, 7.6.2, 7.7.2, 7.8.2, and 7.9.2 of TR-0616-48793 discuss when measurements are taken to update the NRFs. NRC staff did not identify any discussion in TR- 0616-48793 to address the potential situation where a measurement exceeds the tolerance limit for a given NRF. NRC staff needs to establish a finding that the uncertainties accounted for by the NRFs are translated into the appropriate Generic Technical Specifications (GTS) such that unexpected uncertainties, associated with reactor physics parameters, will not result in operation of the NuScale Power Module (NPM) outside the bounds of the safety analysis.

Accordingly, NRC staff requests that NuScale describe how each of the NRFs discussed in TR-0616-48793 is translated into a GTS, or otherwise commit to preventing operation of the NPM outside the tolerance bounds of the NRFs.

NuScale Response:

All of the code biases, uncertainties, and resulting nuclear reliability factors (NRFs) established in Section 7.0 of the Nuclear Analysis Codes and Methods Qualification topical report, TR-0616-48793, (i.e. 7.1.1, 7.2.1, 7.3.1, 7.4.1, 7.5.1, 7.6.1, 7.7.1, 7.8.1, and 7.9.1) have been incorporated into the safety analysis calculations where relevant, and will be accounted for in the determination of appropriate operating limits from the analytical limits used in the safety analysis. As noted in the update methodology portion of each of these sections, the NRFs will be verified and updated as necessary during startup testing and operation of the modules. Startup Testing and Core Follow programs and procedures will be established as part of a licensee's detailed design and construction phases and prior to testing and operation of the first module.

The table below describes how the individual parameters will be addressed as described in the NuScale Design Certification Application and the cycle-specific Core Operating Limits Report (COLR) developed based on the Generic Technical Specifications (GTS) and Limiting Conditions for Operation (LCOs).

Table: NRF Incorporation

Topical Section	Parameter	Incorporation Means	Action if Measurement Exceeds NRF
7.1.2	Critical Boron Concentration (CBC)	Addressed by Startup Testing and Core Follow programs and procedures. Evaluated indirectly in GTS LCO 3.1.1 Shutdown Margin (SDM) and LCO 3.1.2 Core Reactivity.	In accordance with typical plant processes and the corrective action program, a condition report will be written, NRF will be evaluated, and safety analysis impact assessed. Surveillance will indirectly confirm that NRF has been adequately considered and addressed in plant procedures including the predicted CBC.
7.2.2	Differential Boron Worth (DBW)	Addressed by Startup Testing and Core Follow programs and procedures. Uncertainty is evaluated indirectly in GTS LCO 3.1.1 SDM and LCO 3.1.2 Core Reactivity.	In accordance with typical plant processes and the corrective action program, a condition report will be written, NRF will be evaluated, and safety analysis impact assessed. Surveillance will indirectly confirm that NRF has been adequately considered and addressed in plant procedures including the predicted DBW.
7.3.2	Isothermal Temperature Coefficient (ITC) & Moderator Temperature Coefficient (MTC)	Addressed by Startup Testing and Core Follow programs and procedures. Uncertainty will be addressed in the development of the COLR limits and GTS LCO 3.1.3 for MTC.	For ITC, in accordance with typical plant processes and the corrective action program, a condition report will be written, NRF will be evaluated, and safety analysis impact assessed. The MTC LCO will verify operation within the bounds of the safety analysis; NRF will be considered and addressed in the MTC limit established by the COLR.
7.4.2	Power Coefficient & Fuel Temperature Coefficient	The NRF is applied analytically.	Values are set conservatively and no measurement is performed.

Topical Section	Parameter	Incorporation Means	Action if Measurement Exceeds NRF
7.5.2	Control Rod Assembly (CRA) Bank Worth	Addressed by Startup Testing and Core Follow programs and procedures. Uncertainty is evaluated indirectly in GTS LCO 3.1.1 SDM and LCO 3.1.2 Core Reactivity and is accounted for by LCO 3.1.5 Shutdown Group Insertion Limits and 3.1.6 Regulating Group Insertion Limits.	In accordance with typical plant processes andthe corrective action program, a condition report will be written, NRF will be evaluated, and safety analysis impact assessed. NRF will be considered and addressed in the limits established by the COLR.
7.6.2	Assembly Radial Peaking	Addressed by Startup Testing and Core Follow programs and procedures. Radial peaking is conservatively addressed in GTS LCO 3.2.1.	In accordance with typical plant processes and the corrective action program, a condition report will be written, NRF will be evaluated, and safety analysis impact assessed.
7.7.2	Pin Peaking	Addressed by Startup Testing and Core Follow programs and procedures. $F_{\Delta H}$ uncertainty will be addressed in the development of the COLR limits and GTS LCO 3.2.1 Enthalpy Rise Hot Channel Factor in accordance with the COLR.	The NRF will be considered and addressed in the Enthalpy Rise Hot Channel Factor limit established by the COLR. Limiting Condition for Operation in GTS will verify operation is maintained within the bounds of the safety analysis.
7.8.2	Axial Offset (AO)	Addressed by Startup Testing and Core Follow programs and procedures. Uncertainty will be addressed in the development of the COLR limits and GTS LCO 3.2.2 AO in accordance with the COLR.	The NRF will be considered and addressed in the AO limit established in the COLR. Limiting Condition for Operation in GTS will verify operation is maintained within the bounds of the safety analysis.

Topical Section	Parameter	Incorporation Means	Action if Measurement Exceeds NRF
7.9.2	Kinetics	As described in Section 7.9.1, the NRF is applied analytically when the parameter is used independently of another parameter that has an associated NRF.	As described in Section 7.9.2, this parameter will be addressed in tandem with Section 7.2.2. Values are set conservatively and no measurement performed.

Based on the aforementioned discussion, all of the NRFs discussed in this topical report are applied and continually assessed to ensure plant operation within the bounds of the safety analysis.

Impact on Topical Report:
There are no impacts to the Topical Report as a result of this response.

Response to Request for Additional Information
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NRC Question No.: 29754

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GDC 11, Reactor inherent protection, which requires that the reactor core and associated coolant systems be design so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

GDC 12, Suppression of reactor power oscillations, which requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations which can result sin conditions exceeding specified acceptable fuel design limits (SAFDLs) are not possible or can be reliably and readily detected and suppressed.

GDC 26, Reactivity control system redundancy and capability, which requires, in part, that the control rods be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for stuck rods, SAFDLs are not exceeded.

GDC 27, Combined reactivity control systems capability, which requires that the reactivity control systems be designed to have a combined capability of reliably controlling reactivity

changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Part of the methodology described in TR-0616-48793 are nuclear reliability factors (NRFs) which account for the uncertainty on reactor physics parameters. The base NRF for the critical boron concentration (CBC), discussed in Section 7.1.1 of TR-0616-48793, uses initial values only from Three Mile Island, Unit 1. The industry standard NRF for CBC, presented for comparison purposes, is dependent on a "representative" differential boron worth that is used in converting the industry standard values. Additionally, the industry standard NRF for CBC lies outside the bounds of the base NRF proposed by NuScale. NRC staff is questioning how the representative differential boron worth was determined to be suitable for converting the industry standard values and why a base NRF was chosen that does not bound an industry standard value. Accordingly, NRC staff requests NuScale justify the basis for the representative differential boron worth, and to explain how the base NRFs for CBC provide conservative bounds.

NuScale Response:

The measured-to-predicted conservatism in Table 6-1 and Figures 6-1 and 6-2 of the Nuclear Analysis Codes and Methods Qualification topical report, TR-0616-48793, show that CMS5 {{ }}2(a),(c) critical boron concentration (CBC) for the Three Mile Island Unit 1 (TMI-1) simulation during the entire plant cycle. As shown in Table 6-2 of the topical report, this comparison resulted in {{ }}2(a),(c) . The industry values shown in Table 7-1 of the topical report have {{ }}2(a),(c) .

The differential boron worth (DBW) of {{ }}2(a),(c) used for conversion of the industry standard value to units of ppm boron was chosen based on measured differential boron worth values taken from TMI-1 Cycles 1 and 2 (Reference 10.1.30 of the topical report), a higher-order code benchmark (topical report Section 5.1.4.2.1, Table 5-3), and calculated values for the NuScale design (NuScale Standard Plant Design Certification Application Part 2 Tier 2, Section 4.3, Figure 4.3-21). The limited quantity of DBW measurement values from TMI-1 and the values from the higher-order code benchmark produced an average value of approximately {{ }}2(a),(c) . The calculated DBW values for the NuScale design range from approximately 8 pcm/ppmb to 13 pcm/ppmb, {{ }}2(a),(c) . The differential boron worth is not a constant quantity; it is dependent on many factors including fuel enrichment, fuel exposure, and the presence and strength of burnable absorbers and control rods. A single DBW conversion factor was chosen for the industry standard value for the CBC comparison to provide a representative value in consistent units of ppmb. The resulting industry standard NRFs for CBC are not provided as limiting values, since these NRFs will vary depending on the DBW chosen for conversion.

Since the CMS5 calculated values {{

}}[2(a),(c)]. The net effect on the NuScale NRFs is {{
}}[2(a),(c)] whereas the industry value has {{
}}[2(a),(c)]. Therefore, although it appears that the NuScale NRF is outside the bounds of the industry value, the {{
}}[2(a),(c)].

Impact on Topical Report:
There are no impacts to the Topical Report as a result of this response.

Enclosure 3:

Affidavit of Zackary W. Rad, AF-0717-54772

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

1. I am the Director, 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 Request for Additional Information response reveals distinguishing aspects about the methodology by which NuScale develops its nuclear analysis codes and methods qualification.

    NuScale has performed significant research and evaluation to develop a basis for this methodology 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 Request for Additional Information No. 14, eRAI 8807. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.
5. The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).
6. Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:
   a. The information sought to be withheld is owned and has been held in confidence by NuScale.
   b. The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.
   c. The information is being transmitted to and received by the NRC in confidence.
   d. No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on 7/6/2017.

Zackary W. Rad