## NuScaleTRRaisPEm Resource

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| **From:** | Bavol, Bruce |
| **Sent:** | Monday, May 08, 2017 3:08 PM |
| **To:** | NuScaleTRRaisPEm Resource |
| **Cc:** | Drzewiecki, Timothy; Travis, Boyce |
| **Subject:** | RE: Topical Report - Request for Additional Information Letter No. 14 (eRAI No. 8807) Section 04.03 - Nuclear Design |
| **Attachments:** | RAI 8807_TR-0616-48793.docx |

Attached please find NRC staff's request for additional information concerning review of the NuScale Design Certification Application.

Please submit your response within 60 days of the date of this RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

Bruce M. Bavol

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**RAI 8807 (Questions 29739, 29749, 29750, 29752, and 29754)**

**TR-0616-48793-P, "Nuclear Analysis Codes and Methods Qualification"**

**(Question 29739) 04.03 - Nuclear Design**

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.

**(Question 29749) 04.03 - Nuclear Design**

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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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.

**(Question 29750) 04.03 - Nuclear Design**

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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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 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.

**(Question 29752) 04.03 - Nuclear Design**

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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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.

**(Question 29754) 04.03 - Nuclear Design**

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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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.