October 10, 2019

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 Supplemental Response to NRC Request for Additional Information No. 9306 (eRAI No. 9306) on the NuScale Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 0

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9306 (eRAI No. 9306)," dated April 04, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 9306 (eRAI No.9306)," dated June 04, 2018
3. NuScale Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 0, dated December 2016
4. NuScale Power, LLC Supplemental Response to "NRC Request for Additional Information No. 9306 (eRAI No. 9306)" dated February 21, 2019

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. 9306:

- 15.04.08-1

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 Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

Michael Melton
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9306

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Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9306

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 15.04.08-1

In accordance with 10 CFR 50 Appendix A GDC 28, "Reactivity Limits," the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a rod ejection accident (REA) can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 provides review guidance related to the spectrum of REAs. For an applicant to accurately analyze its plant design for an REA, the underlying software used as part of the applicant's methodology must be properly verified and validated.

Section 3.2.1.4 of Topical Report TR-0716-50350-P, "Rod Ejection Accident Methodology," Revision 0, provides the validation of SIMULATE-3K, which is used to provide a three-dimensional nodal reactor kinetics solution. This section indicates that the SPERT-III benchmark and the NEACRP REA problem were used to validate SIMULATE-3K for the purpose of REA analyses. The references for the validation of SIMULATE-3K against SPERT- III and NEACRP appear to be based on conference proceedings. Neither a summary of results nor an analysis of bias or uncertainty is provided. The referenced conference proceedings are not part of the applicant's Appendix B quality assurance program and, therefore, the robustness of the validation is not demonstrated. As such, the staff makes the following requests:

a. Provide a plot of the comparison between the SIMULATE-3K model and the SPERT-III benchmark results.

b. Provide a summary of the SIMULATE-3K comparison against the NEACRP REA benchmark problem.

c. Provide a reference for a complete verification/validation analysis of SIMULATE-3K under an Appendix B quality assurance program.

NuScale Response:

The original NuScale response was submitted in NuScale correspondence RAIO-0618-60285 and was dated June 4, 2018. A supplement to this RAI response was submitted in NuScale correspondence RAIO-0219-64616, dated February 21, 2019, which detailed the results of a benchmark of the dynamic reactor response simulated by SIMULATE-3K (S3K) to the transient special power excursion reactor test III E-Core experiment (SPERT).

This supplement provides a mark-up to the Rod Ejection Accident Methodology Topical Report (TR-0716-50350), Section 3.2.1.4, which adds a summary of the NuScale SIMULATE-3K to the SPERT III benchmark results as indicated below.

Impact on Topical Report:

Topical Report TR-0716-50350, Rod Ejection Accident Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

and the NEACRP control rod ejection problem computational benchmark (Reference 8.2.22).

The Studsvik SPERT III benchmark provides measured REA transient data for comparison to SIMULATE-3K. SPERT III was a pressurized water nuclear research reactor that analyzed reactor kinetic behavior under conditions similar to commercial reactors. The SPERT III core resembled a commercial reactor, but of a reduced size more closely resembling the NuScale core size. The fuel type (uranium dioxide), moderator, system pressure, and certain initial operating conditions considered for SPERT III are also representative of NuScale. This benchmark demonstrates the ability of SIMULATE-3K to model fast reactivity transients in a PWR core (Reference 8.2.22). Similarities between the NuScale design and the SPERT III core, and notably the small core size, demonstrate applicability and suitability for SIMUALTE-3K REA transient analysis of the NuScale core.

In addition to the Studsvik benchmarks aforementioned, NuScale has performed a benchmark of the dynamic reactor response simulated by SIMULATE-3K of the SPERT III experiment. The original experiment included on the order of one hundred unique tests at five different sets of thermal-hydraulic conditions, with varying initial static worths at each statepoint. One test from each condition set that generally corresponds to the highest static worth for the statepoint has been benchmarked. A comparison of key parameters demonstrates that SIMULATE-3K compares to SPERT with generally excellent agreement; differences are within the experimental uncertainty (with few exceptions), and the major and minor phenomena are correctly predicted.

The NEACRP control rod ejection problem is a computational benchmark that includes a reference solution provided by the PANTHER code, and SIMULATE-3K REA transient results are compared against the reference solution. In this benchmark, a rod ejection accident in a typical commercial PWR at HZP conditions is analyzed. The fuel type (uranium dioxide), moderator, system pressure, and certain initial operating conditions considered for NEACRP are also representative of NuScale. The capability of SIMULATE-3K to model reactivity insertions in the NEACRP benchmark analysis (Reference 8.2.24 and 8.2.25) demonstrates suitability of the code for reactivity transient applications, and specifically REA analysis applications.

The SPERT III and NEACRP benchmarks demonstrate the combined transient neutronic, TH, and fuel pin modeling capabilities of SIMULATE-3K. SIMULATE-3K results for maximum power pulse, time to peak power, inserted reactivity, energy release, and fuel centerline temperature were in excellent agreement with the results from the two benchmark problems. The SIMULATE-3K results for each of these benchmark problems establish the ability of the code to accurately model an REA transient event and predict key reactivity and power-related parameters.

3.2.2 System Response

The NRELAP5 code was developed based on the Idaho National Laboratory RELAP5-3D© computer code. RELAP5-3D©, version 4.1.3 was procured by NuScale and used as the baseline development platform for the NRELAP5 code. Subsequently, features