RAIO-0818-61358

August 10, 2018

Docket No. 52-048

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. 483 (eRAI No. 9516) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 483 (eRAI No. 9516)," dated May 25, 2018

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

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 9516:

- 15-22

The response schedule for the remaining questions of RAI No. 483, eRAI 9516 was provided in an email to NRC (Greg Cranston) dated June 19, 2018.

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 Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9516

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9516

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

eRAI No.: 9516
Date of RAI Issue: 05/25/2018

NRC Question No.: 15-22
10 CFR 50 Appendix A, GDC 34, Residual heat removal, and NuScale's PDC 34, in FSAR Section 3.1.4.5, state,

> "A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded."

The long term cooling technical report (LTC-TR), TR-0916-51299, supports FSAR Section 15.0.5, Long Term Decay and Residual Heat Removal, when the ECCS is used for long term decay heat removal following either a non-LOCA or LOCA event up to 72 hours. The primary acceptance criteria for the analysis are 1) Collapsed liquid level is maintained above the active fuel and 2) fuel cladding temperature is maintained at an acceptable level such that the SAFDLs are preserved.
RG 1.203 describes the EMDAP, and provides guidance, which the NRC staff considers acceptable for use in developing and assessing EMs used to analyze transient and accident behavior, which in Element 4 discusses the process to Assess Evaluation Model Accuracy. Basic principle (4) of the EM development and assessment states:

> *"A key feature of the adequacy assessment is the ability of the EM or its component devices to predict appropriate experimental behavior. Once again, the focus should be on the ability to predict key phenomena, as described in the first principle. To a large degree, the calculational devices use collections of models and correlations that are empirical in nature. Therefore, it is important to ensure that they are used within the range of their assessment."*

The LTC report Executive Summary states:

"The LTC EM uses the proprietary NRELAP5 systems analysis computer code as the computational engine, derived from the Idaho National Laboratory RELAP5-3D© computer code. The models and correlations used by NRELAP5 were reviewed and, where appropriate, modified for use within the long-term cooling EM."

From a review of the LTC-TR, it is not clear which NRELAP5 models and correlations, if any, were modified for use within the LTC EM. In order to follow the RG 1.203 guidance, documentation of the models and correlations modified for use within the LTC EM should be provided in the report for the staff to reach a reasonable assurance finding that GDC 34/PDC-34 is met.

NuScale Response:

The long-term cooling (LTC) methodology as described in the LTC technical report (TR-0916-51299-P, Revision 0) is an extension of the NuScale loss-of-coolant accident (LOCA) evaluation model (EM) (TR-0516-49422, Revision 0). It is applicable to demonstrate long-term cooling capability following both LOCA and non-LOCA design basis events. The transition to LTC occurs once natural circulation between the RPV and containment has been established and the pressure and liquid levels in the CNV and the RPV approach a stable equilibrium condition. This natural circulation pattern consists of coolant upflow through the core producing steam, steam leaving the RPV through the reactor vent valve (RVV) and condensing on the cool containment shell, and the condensate being returned from the containment pool to the RPV through the reactor recirculation valve (RRV). This is a natural transition point into LTC as all events will evolve to this condition. The LTC technical report assesses the EM adequacy from this transition point.

Table 2-1 of the LTC technical report provides a summary of the EMDAP elements as it relates to LTC. An evaluation of Element 4, *Assess Evaluation Model Accuracy,* is provided in Table 2-1. The NRELAP5 computer code models and correlations are documented in the LOCA EM. No specific NRELAP5 code modifications were implemented specifically for LTC. The Executive Summary of Reference 1 was revised to clarify that no code modifications were necessary to NRELAP5 for use in the LTC methodology as indicated in the markup at the end of this response.

Impact on DCA:

Technical Report TR-0916-51299, Long-Term Cooling Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

parameters which are specifically addressed. An extensive PIRT was developed for LTC. Each important parameter is discussed and evaluated in this report as it relates to the LTC EM.

The LTC EM uses the proprietary NRELAP5 systems analysis computer code as the computational engine, derived from the Idaho National Laboratory RELAP5-3D© computer code. The models and correlations used by the NRELAP5 code were reviewed and determined to be~~,~~ ~~where~~ appropriate~~, modified~~ for use within the long-term cooling EM. The NRELAP5 model is validated through the assessment of NIST-1 facility tests and comparison of NRELAP5 predictions to test results. Comparison of the NRELAP5 model to the NIST-1 test results demonstrate that the NRELAP5 code adequately predicts the NPM conditions both in the RPV and the CNV.

The methodology for the NPM thermal-hydraulic response and boron precipitation evaluation are presented in this report. There are two LTC general conditions which address the thermal-hydraulic response and boron precipitation: maximum cooldown to minimize the RPV core liquid volume in the riser region for addressing boron precipitation and the minimum collapsed liquid level above the active fuel, and minimum cooldown to maximize the fuel cladding temperature.

The methodology is demonstrated in the report by presenting the limiting results of a base LOCA case for the letdown line break (LDBRK) utilizing conservative worst case conditions determined by sensitivity calculations. In addition the SGTF results are presented. Sensitivity cases performed considered the following assumptions:

- single active failure, ECCS valve failure to open is the relevant single active failure to consider in the LTC analyses
- decay heat, ranging from no decay heat to 120 percent of nominal
- heat transfer from the RPV to CNV, ranging from adiabatic to 1000 percent of nominal
- heat transfer from the CNV to pool, ranging from 20 percent to 1000 percent
- reactor pool temperature, ranging from 40 degrees F to 210 degrees F
- reactor pool level, down to 45 feet (Nominal at 69 feet)
- reactor pool volume effect on calculated pool temperature heatup from initial conditions
- non-condensable gas effect
- pressurizer level, down to 20 percent level

In all analyzed cases, the core remained covered, with greater than 2.25 feet of collapsed liquid level in the riser above the top of the core. Possible leakage from the CNV was found to have a negligible impact on the results. The cases identified as most limiting, the minimum cooldown, maximum cooldown, and SGTF with decay heat removal system (DHRS), all showed consistently decreasing reactor coolant system (RCS) and cladding temperatures, supporting the conclusion that the ECCS is capable of providing adequate cooling for the 72 hour evaluation period.

In order to evaluate the criterion for maintaining coolable geometry, the possibility of boron precipitation is evaluated in this report. The methodology for determining boron precipitation is