RAIO-0518-60191

May 25, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 418 (eRAI No. 9511) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 418 (eRAI No. 9511)," dated April 11, 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. 9511:

- 15.04.02-5

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,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9511

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9511

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

eRAI No.: 9511
Date of RAI Issue: 04/11/2018

NRC Question No.: 15.04.02-5

General Design Criterion (GDC) 10, "Reactor design," in Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A, requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," provides the staff guidance in determining compliance with GDC 10, among several other GDC, and states that the review should consider the entire power range from low to full power and the allowed extreme range of reactor conditions during the operating (fuel) cycle, including rod configurations, power distribution, and associated reactivity feedback components.

It is not clear from FSAR Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," whether limiting axial and radial power shapes were used in the subchannel analysis for this event. To allow the staff to make a safety finding with regard to GDC 10, confirm that the limiting power distributions were input to the subchannel analysis, and update the FSAR to include a statement that the limiting axial and radial power shapes were used.

NuScale Response:

FSAR Section 15.4.2.3.1 describes the evaluation model used in the subchannel analysis of the uncontrolled control rod assembly withdrawal at power. Limiting axial and radial power shapes are used in the subchannel analysis to ensure a conservative minimum critical heat flux ratio result. As shown in the attached markups, a statement was added to the FSAR to indicate that limiting power shapes are used as well as a reference to the associated subchannel methodology.

Impact on DCA:

FSAR Section 15.4.2 has been revised as described in the response above and as shown in the markup provided in this response.

- high pressurizer pressure

In uncontrolled CRA withdrawal events that result in a reactor trip, the subsequent actuation of the decay heat removal system (DHRS) is credited with maintaining reactor cooling. The MPS signals credited for DHRS actuation are high hot leg temperature and high pressurizer pressure.

There are no single failures that could occur during an uncontrolled CRA withdrawal event that result in a more severe outcome for the limiting uncontrolled CRA withdrawal cases. The diversity, redundancy, and independence of the MPS ensure the system will perform its intended function despite a single failure.

The loss of normal AC power is analyzed for an uncontrolled CRA withdrawal at power. The loss of power scenarios are discussed below:

- Loss of Normal AC - In this scenario, the MPS remains powered, so none of the safety systems are automatically actuated. However, power is lost to the feedwater pumps, CVCS recirculation pumps, pressurizer heaters, and the condenser, resulting in a turbine trip.
  - Loss of normal AC at the time of the event initiation is analyzed in NRELAP5.
  - Loss of normal AC at the time of reactor trip is analyzed in NRELAP5.
- Loss of EDNS and Loss of normal AC - Power to the control rod drive mechanisms is provided via the nonsafety DC power distribution (EDNS), so this scenario is the same as discussed above, with the addition of the CRAs dropping at the time at which power is lost. For this event, this scenario is non-limiting because of the immediate loss of power to the CRDMs, resulting in the drop of the CRAs.
- Loss of EDSS, EDNS and Loss of normal AC - Power to the MPS is provided by the highly-reliable DC power distribution system (EDSS), so this scenario results in an actuation of RTS and all of the engineered safety features. This scenario is non-limiting because of the immediate reactor trip.

15.4.2.3 Thermal Hydraulic and Subchannel Analyses

15.4.2.3.1 Evaluation Models

The thermal hydraulic analysis of the plant response to an uncontrolled CRA withdrawal is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

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The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Limiting axial and radial power shapes are used in the subchannel analysis to ensure a conservative MCHFR result, in accordance with the methodology described in Reference 15.4-1. See Section 15.0.2 for a discussion of the VIPRE-01 code and evaluation model.

15.4.2.3.2 Input Parameters and Initial Conditions

RAI 15.04.01-3, RAI 15.04.02-4

A spectrum of initial conditions is analyzed to find the limiting reactivity insertion due to an uncontrolled CRA withdrawal. Key inputs of the uncontrolled CRA withdrawal evaluation are provided in Table 15.4-5 for the limiting MCHFR case, ~~and in~~ Table 15.4-28 for the limiting RCS pressure case, and Table 15.4-31 for the limiting linear heat generation rate (LHGR) case. The following initial conditions and assumptions ensure that the results have sufficient conservatism.

RAI 15.04.02-1, RAI 15.04.02-2, RAI 15.04.02-4

- Initial power level: 25 percent, 50 percent, 75 percent, and 102 percent of nominal power are analyzed in the uncontrolled CRA withdrawal evaluation. The power level for the limiting MCHFR ~~and RCS pressure~~ case~~s~~ is 75 percent of nominal power. The power level for the limiting RCS pressure and LHGR cases is~~and~~ 102 percent of nominal power~~, respectively~~.

RAI 15.04.02-1, RAI 15.04.02-2, RAI 15.04.06-1

- Reactivity insertion rate: The positive reactivity inserted by the CRA withdrawal is modeled as a constant reactivity addition beginning at the transient initiation. The maximum rod speed of 15 inches/min corresponds to a maximum reactivity insertion of 21 pcm/s. However, to bound the reactivity insertion from possible boron dilution scenarios, a maximum reactivity insertion of 35 pcm/s is analyzed.
  - The reactivity insertion rate for the limiting MCHFR case is 0.9 pcm/s.
  - The reactivity insertion rate for the limiting RCS pressure case is 15.2 pcm/s.

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  - The reactivity insertion rate for the limiting LHGR case is 35.0 pcm/s.
- Time in cycle: The BOC core conditions are implemented in the limiting uncontrolled CRA withdrawal cases. The least negative reactivity coefficients occur at the BOC, and provide the least amount of feedback to mitigate the power increase due to an uncontrolled CRA withdrawal.
- The turbine bypass system is not credited in this analysis to minimize heat removal by the secondary side.
- Conservative scram characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and using a bounding control rod drop rate.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide (RG) 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.