RAIO-0617-54488

June 15, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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Rockville, MD 20852-2738

**SUBJECT:** NuScale Power, LLC Response to NRC Request for Additional Information No. 10 (eRAI No. 8743) on the NuScale Design Certification Application

**REFERENCE:** U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 10 (eRAI No. 8743)," dated April 25, 2017

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. 8743:

- 15.01.05-1

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 Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

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

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8743

**Enclosure 1:**

NuScale Response to NRC Request for Additional Information eRAI No. 8743

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

**eRAI No.:** 8743
**Date of RAI Issue:** 04/25/2017

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**NRC Question No.:** 15.01.05-1

In accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 31, “Fracture prevention of reactor coolant pressure boundary,” the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties; (2) the effects of irradiation on material properties; (3) residual, steady state and transient stresses; and (4) size of flaws.

To meet the requirements of GDC 31, as it relates to the steam line break (SLB) accident resulting in a limiting reactor coolant system (RCS) pressure, the accident analysis should consider appropriate uncertainties for determining conservative temperatures and pressures at the RCPB to show that the probability of rapidly propagating fracture is minimized for this transient.

In FSAR Tier 2, Section 15.1.5.3.2, “Input parameters and Initial Conditions,” under the “Steam Line Break Case Resulting in a Limiting Reactor Coolant System Pressure” heading, the applicant states that beginning-of-life (BOL) steam generator (SG) characteristics are used and a 30% uncertainty is added to the SG heat transfer. The staff notes that BOL SG characteristics and adding to the steam generator heat transfer allows for more heat to be removed from the primary system, ultimately resulting in a less limiting primary system pressure transient. Based on the docketed information, the staff is unable to determine the adequacy of a 30% uncertainty addition to the steam generator heat transfer. Furthermore, the staff is unable to determine if SG tube plugging and fouling is appropriately accounted for in this analysis since BOL SG conditions are used. The staff requests the applicant to either (1) provide justification in the FSAR as to why the applicant adds to the steam generator heat transfer and does not subtract from the steam generator heat transfer or (2) modify the FSAR

as necessary. The staff also requests the applicant to provide justification in the FSAR as to why SG tube plugging and fouling are not accounted for in this steam line break case resulting in a limiting RCS pressure. If SG tube plugging and fouling need to be accounted for, the staff requests the applicant to modify the FSAR as necessary.

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**NuScale Response:**

The steam piping failures are cooldown events and, therefore, are not limiting for primary pressure. For these cooldown events, minimum critical heat flux ratio (MCHFR) and mass release are the parameters of interest. The events in FSAR Section 15.2, "Decrease in Heat Removal by the Secondary Side" provide the bounding events for reactor coolant system (RCS) and steam generator (SG) pressure. The limiting events in FSAR Section 15.2 were evaluated considering the effects of heat transfer uncertainty, SG tube plugging and fouling, both for increased and decreased primary to secondary heat transfer.

The justification for the methodology used for the evaluation of the steam piping failure is presented in the Non-Loss-of-Coolant Accident Analysis Methodology, TR-0516-49416-P, Revision 0, Section 7.2.4. The initial condition biases and bases are presented in Table 7-24 of this report. The report explains that for large steam piping failures, the cooldown reduces RCS pressure, therefore, evaluating effects of changes to initial conditions is unnecessary. For small steam piping failures, the cooldown and depressurizing effects are small such that the loss of offsite power and corresponding loss of feedwater are the dominant effects on RCS pressure. The maximum RCS pressure for the small steam piping break was reported for completeness in the conclusions of FSAR Section 15.1.5 to show that the primary pressure met the acceptance criteria. Variations of individual parameters for inputs, such as steam generator heat transfer, were not evaluated in the steam piping failure analyses for impact on RCS pressure. The pressure presented in the conclusions of FSAR Section 15.1.5 was the highest peak RCS pressure for the scenarios analyzed. Sensitivity studies showed that varying parameters such as heat transfer uncertainty, SG tube plugging and fouling had a very small effect on RCS or SG pressure. Since the impact of these initial conditions on RCS or SG pressure was minor and bounded by the events in FSAR Section 15.2, only the nominal maximum pressures were presented in FSAR Section 15.1.

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