RAIO-0618-60534

June 19, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 03 (eRAI No. 8744) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 03 (eRAI No. 8744)," dated April 25, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 03 (eRAI No.8744)," dated June 21, 2017

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. 8744:

- 15.02.08-1

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

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8744

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8744

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

eRAI No.: 8744
Date of RAI Issue: 04/25/2017

NRC Question No.: 15.02.08-1

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 assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle 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 feedwater line break (FWLB) accident resulting in a limiting 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 Final Safety Analysis Report (FSAR) Tier 2, Section 15.2.8.4, "Input Parameters and Initial Conditions," the applicant states that a 30% uncertainty is added to the steam generator heat transfer. However, in FSAR Tier 2, Table 15.2-28, "Biases and Uncertainties – Feedwater Line Break," the applicant states that a -30% uncertainty is used for the steam generator heat transfer. The staff notes that these statements in the FSAR are inconsistent. Based on the docketed information, the staff is unable to determine what uncertainty value is used for the steam generator heat transfer and the adequacy of the reported 30% uncertainty addition to the steam generator heat transfer. The staff requests the applicant to clearly state in the FSAR which uncertainty value is used and provide justification in the FSAR as to why the applicant adds to (or subtracts from) the steam generator heat transfer.

NuScale Response:

NuScale is supplementing its response to RAI 8744 (Question 15.02.08-1) originally provided in letter RAIO-0617-54560 dated June 21, 2017. The Supplemental Response replaces page 15.2-23 and 15.2-24 of the original Draft Revision 1 FSAR markup. In particular, the

supplemental FSAR markup of NuScale's original response is modified to indicate that the failure of the safety related check value to close on the failed steam generator (SG) results in the limiting SG pressure for the feed water line break outside containment.

Original Response

The limiting reactor coolant system (RCS)pressure transient results from the double-ended guillotine (DEG) feedwater line break (FWLB). For cases evaluated to maximize primary pressure, a negative 30% bias to the primary-to-secondary heat transfer rate was applied to maximize the heat retained in the RCS. In addition, the design limit steam generator (SG) fouling factor (10^{-4} hr-ft^2 -°F/BTU) and SG tube plugging (10%) were applied to maximize RCS heatup and, therefore, RCS pressure. The limiting decay heat removal system (DHRS) case was also a DEG FWLB using the same heat transfer bias,fouling and tube plugging values as the limiting RCS pressure case.

For the DEG FWLB evaluated for the limiting SG pressure, the highest pressure was obtained by using a positive 30% heat transfer bias, no fouling and no SG tube plugging. However the DEG FWLB was not the limiting break for SG pressure. Smaller breaks of 8% to 10% of feedwater flow with a negative 30% primary to secondary heat transfer bias, a large fouling factor and 10% tube plugging, resulted in slightly higher SG pressure. The 10% FWLB with failure of the feedwater isolation valve (FWIV) backflow prevention device was selected as the representative limiting SG case for the sequence of events FSAR Table 15.2-25 and SG pressure FSAR Figures 15.2-42 and 15.2-43. Similar FWLBs, 4% to 8% break size, with the same heat transfer biases,resulted in the minimum critical heat flux ratio (MCHFR). The 8% FWLB was selected as the presentative limiting MCHFR case for these sequence of events in FSAR Table 15.2-26 and MCHFR Figure 15.2-44. Note that the title of FSAR Figure 15.2-44 was mislabeled as the MCHFR for the "Maximum RCS Pressure Case" and has been corrected to be the "Limiting MCHFR Case" as shown in the attached FSAR markup.

Therefore, the limiting cases for RCS pressure, DHRS performance, SG pressure and MCHFR all used the heat transfer bias of -30% stated in FSAR Table 15.2-28. The text in FSAR Section 15.2.8.4 was intended to reflect the information in the table. The wording in the attached FSAR markup has been clarified to eliminate the discrepancy. It is noteworthy that varying parameters for heat transfer, SG tube plugging and fouling had less than a 10 psi effect on peak primary or secondary pressure. The limiting calculated RCS pressure (2164 psia) and SG pressure (1328 psia) were well below the acceptance criteria (2310 psia) such that the variation of these parameters was not a significant contributor to determining the acceptability of the results. Additional detail on the basis for sensitivity studies is presented in the Non-Loss-of-Coolant Accident Analysis Methodology, TR-0516-49416-P, Revision 0, Section 4.2 and Section 7.2.12.

During separate discussions with NRC staff, questions were also raised regarding the wording in FSAR Section 15.2.8.4 relating to the assumptions on closure time for the FWIVs. When credited, the FWIVs are assumed to close in the maximum design closure time (7 seconds), not

at the design limit closure rate.For the majority of events, the safety-related check valve (back flow prevention device)will seat such that the closure time of the FWIV is not required to prevent backflow thought he FW system. For these events, the FWIV closure time is not a consideration because the check valve closes very quickly (approximately 1 second).

Therefore,the FWIV closure is removed from the description where the safety-related check valve is credited. The description of these input parameters is clarified in the attached FSAR markup. The description of the sequence of events in FSAR Section15.2.8.2 is also modified to reflect the limiting cases discussed in this response.

In the course of preparing this response, NuScale has included some editorial corrections and clarifications in the attached FSAR markup. In FSAR Section 15.2.8.2, an incorrect characterization of the limiting MCHFR case was eliminated. In FSAR Section 15.2.8.4, the term “Case A” was deleted because it refers to an internal calculation reference. Additionally, wording was added to clarify that the FWIV backflow prevention device is the safety-related FW check valve. The description of the limiting SG pressure event was also corrected to indicate that AC power is lost at the time of the break. In FSAR Section15.2.8.5, a statement was added to clarify that the high pressurizer pressure reactor trip is delayed by biasing pressurizer pressure and level low.

Supplemental Response

The supplemental response is the same as the original. However, the attached FSAR markup has been changed to clarify that the failure of the safety related check value to close on the failed steam generator (SG) results in the limiting SG pressure for the feed water line break outside containment.

Impact on DCA:

Section 15.2.8.2 has been revised as described in the response above and as shown in the markup provided in this response.

heat removal capacity. Table 15.2-25 provides the sequence of events for the limiting secondary pressure case resulting from the 10-percent FWS line break with the failure of the FWIV backflow prevention device.

The limiting DHRS function case involves a DEG break in the feedwater piping inside of containment. Unlike breaks outside of containment, this break results in the complete loss of one train of DHRS. Upon break initiation, pressure inside of the CNV rapidly increases, reaching the high CNV pressure analytical limit and actuating reactor trip, containment isolation, and DHRS. The remaining DHRS loop provides cooling to the module and is sufficient to remove 100 percent of decay heat and drive flow through the core. This event is not limiting for any of the acceptance criteria. The sequence of events for this case is provided in Table 15.2-27.

The MPS is credited to protect the NPM in the event of a FWLB. The following MPS signals provide the plant with protection during a FWLB:

- Low steam pressure
- High pressurizer pressure
- High steam superheat
- High CNV pressure

The actuation of a single RSV is credited for ensuring pressures in the RCS do not exceed the acceptance criteria

RAI 15.02.08-1, RAI 15.02.08-1S1

No single failures have an impact on the limiting ~~primary~~RCS pressure or MCHFR results. The failure of the safety-related check valve (FWIV backflow prevention device) to close on the ~~intact~~failed SG did result in a limiting value for SG pressure. For FWLB inside containment, in the event of the failure of the safety-related check valve, the second nonsafety-related check valve is credited to ensure that adequate inventory is maintained in the intact steam generator and DHRS condenser. Therefore, there are no single active failures that cause the FWLB event to have unacceptable results.

15.2.8.3 Thermal Hydraulic and Subchannel Analyses

15.2.8.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to an FWLB 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 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

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. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.