RAIO-0617-54493

June 15, 2017

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. 17 (eRAI No. 8767) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 17 (eRAI No. 8767)," dated May 08, 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. 8767:

- 15.06.06-1

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
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8767

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NuScale Response to NRC Request for Additional Information eRAI No. 8767

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

eRAI No.: 8767
Date of RAI Issue: 05/08/2017

NRC Question No.: 15.06.06-1

In accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 35, “Emergency Core Cooling,” the emergency core cooling system (ECCS) safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities, shall be provided to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. The staff notes that the applicant departs from GDC 35 by adopting Principle Design Criterion (PDC) 35 presented in the Final Safety Analysis Report (FSAR) Tier 2, Section 3.1.

To meet the requirements mentioned above, as they relate to the ECCS system providing abundant core cooling during the inadvertent ECCS actuation event, the accident analysis should show that fuel and clad damage that could interfere with continued effective core cooling is prevented.

In FSAR Tier 2, Section 15.6.6, “Inadvertent Operation of Emergency Core Cooling System,” the applicant does not provide a plot of minimum critical heat flux ratio (MCHFR). In accordance with DSRS Section 15.6.6, Review Procedure 7, the staff cannot determine if the MCHFR is acceptable for this event. The staff recognizes that the MCHFR is reported in FSAR Tier 2, Section 15.6.6.6, “Conclusions.” However, this does not allow the staff to assess the MCHFR throughout the transient to verify when and where the MCHFR occurs. Based on the docketed information, the staff cannot determine if abundant core cooling is provided during the accident and will prevent fuel and clad damage that could interfere with continued effective core cooling. The staff requests the applicant to provide additional plots in the FSAR of MCHFR for the entire transient (“zoomed-in” and “zoomed-out”) that show how the MCHFR behaves as the transient progresses.

NuScale Response:

A plot of the "zoomed-in" critical heat flux ratio (CHFR) has been added to FSAR Section 15.6.6 as FSAR Figure 15.6-67 and a "zoomed-out" CHFR plot has been added as FSAR Figure 15.6-68. FSAR Figure 15.6-67 confirms that the minimum CHFR (MCHFR) is 1.240 and that MCHFR occurs shortly, within one second, after event initiation, as reported in FSAR Section 15.6.6.5. FSAR Section 15.6.6.5 states that the MCHFR is above the CHFR limit. Figure 15.6-68 provides CHFR for the duration of the event.

As stated in FSAR Section 15.6.6.3.1, due to the phenomenological similarities between an inadvertent opening of an ECCS valve and a LOCA, the LOCA evaluation model is conservatively used to evaluate the inadvertent opening of an ECCS valve event. Topical Report TR-0516-494422, Loss-of-Coolant Accident Evaluation Model, Section 4.3 discusses the figures of merit (FOM) for characterizing the plant accident response. The CHFR is an important FOM as it demonstrates there is no significant heatup of the cladding. The MCHFR is above the CHFR limit and occurs within one second of event initiation, therefore, there is no significant heatup of the cladding.

Another FOM is the collapsed liquid level above the core, as it demonstrates there is an adequate supply of liquid water available to the core. Heatup of the fuel will not occur as long as the core is covered with coolant and CHF conditions do not exist. FSAR Figure 15.6-62 shows that the water level remains above the top of the active fuel. And because MCHFR occurs so quickly, the water level in the reactor pressure vessel has little time to respond before MCHFR occurs. Therefore, at the time of MCHFR, the water level is over 30 feet above the top of the active core and stabilizes to approximately 10 feet above the top of active core for the duration of the event. In addition, FSAR Figure 15.6-65 and FSAR Figure 15.6-66 show that the fuel temperatures decrease immediately upon the initiation of the event and continue to decrease throughout the event. Thus, with the two FOMs of MCHFR and collapsed liquid level above the core, effective and abundant core cooling is demonstrated.

Impact on DCA:

FSAR Section 15.6.6 has been revised as described in the response above and as shown in the markups provided in this response.

  - Loss of the normal DC power system (EDNS) and normal AC - Power to the reactor trip breakers is provided via the EDNS, so the primary difference to a loss of normal AC power is that the reactor trip will occur sooner. This scenario is non-limiting for the reasons described above.
  - Loss of the highly reliable DC power system (EDSS), EDNS, and normal AC - This scenario results in an immediate actuation of the reactor trip system, DHRS (although not credited in the analysis), the 24-hour timer for the ECCS valves, and containment isolation. As power to the MPS is lost, the ECCS valve opening is dependent only on the IAB pressure release setpoint. This scenario is non-limiting for the reasons described above.
- Single failure evaluation of a single RVV to open, a single RRV to open, and failure of one ECCS division (one RVV and one RRV) to open was performed to determine the most conservative scenario. The evaluation showed that the single failure cases have no impact on MCHFR or other acceptance criteria evaluated in this analysis. Therefore, the scenario of no single failure is applied in this analysis.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.
- No operator action is credited.

15.6.6.3.3 Results

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Figure 15.6-55 to ~~Figure 15.6-66~~Figure 15.6-68 show the system response to an inadvertent RVV opening event. Table 15.6-17 contains the results of the event. The limiting case is initiated by a spurious opening of an RVV. Sensitivity analysis show that the limiting scenario has all power available and no single failure occurs.

Upon the spurious RVV opening, the large blowdown of the RCS into the containment causes rapid depressurization of the RCS and rapid pressurization of the containment. Spurious RVV flow is shown in Figure 15.6-55 and Figure 15.6-56. The RCS and containment pressures are shown in Figure 15.6-57. The high containment pressure analytical limit is reached shortly after event initiation. The MPS signal actuates control rod insertion, secondary system isolation, and DHRS actuation. However, DHRS operation is conservatively not credited in this analysis.

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The rapid RCS depressurization causes voiding in the core and a momentary decrease in RCS flow (Figure 15.6-58 and Figure 15.6-59), leading to a reduction in CHFR (Figure 15.6-67 and Figure 15.6-68). Reactor power decreases during this time due to control rod insertion and negative void feedback, as seen in Figure 15.6-60. Following the occurrence of transient MCHFR (Figure 15.6-67), a temporary increase in RCS flow is observed due to the increased density gradient from voiding in the riser (Figure 15.6-58).

basis source term bounds the source term, and thus the dose consequences, of this event.

15.6.6.5 Conclusions

The acceptance criteria for an AOO are listed in Table 15.0-2. These acceptance criteria, followed, by how the NuScale Power Plant design meets them, are listed below. Table 15.6-17 provides the results of the limiting scenario of a spurious opening of an RVV.

1) Fuel cladding integrity shall be maintained by ensuring that minimum DNBR remains above the 95/95 DNBR limit. Minimum critical heat flux ratio (MCHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

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The fuel integrity is not challenged by the spurious opening of the RVV. The fuel temperatures decrease upon the reactor trip, as shown in Figure 15.6-65 and Figure 15.6-66, and the water level remains above the top of the active fuel, as shown in Figure 15.6-62. The MCHFR is 1.240, as shown in Figure 15.6-67 and Figure 15.6-68, which is greater than the 95/95 limit of 1.122. As noted in Section 15.6.6.3.1, MCHFR occurs in the high flow correlation range shortly after event initiation as reactor power is still elevated at this time.

2) RCS pressure should be maintained below 110 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2310 psia.

   The RCS pressure is below the acceptance criterion, as shown in Table 15.6-17.

3) The main steam pressure should be maintained below 110 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2310 psia.

   The main steam pressure, presented as steam generator pressure, is below the acceptance criterion, as shown in Table 15.6-17.

4) The event should not generate a more serious plant condition without other faults occurring independently.

   The analysis presented for this event shows that the NPM continues to be cooled with natural circulation through the ECCS valves and the event terminates in a safe, stabilized condition.

The response of the NPM during the long-term cooling phase following the inadvertent opening of an RPV valve is similar to the response of the NPM following a LOCA. The long-term cooling analysis, results and conclusions are discussed in Section 15.6.5.

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Figure 15.6-67: Inadvertent Operation of an Emergency Core Cooling System Valve – Critical Heat Flux Ratio

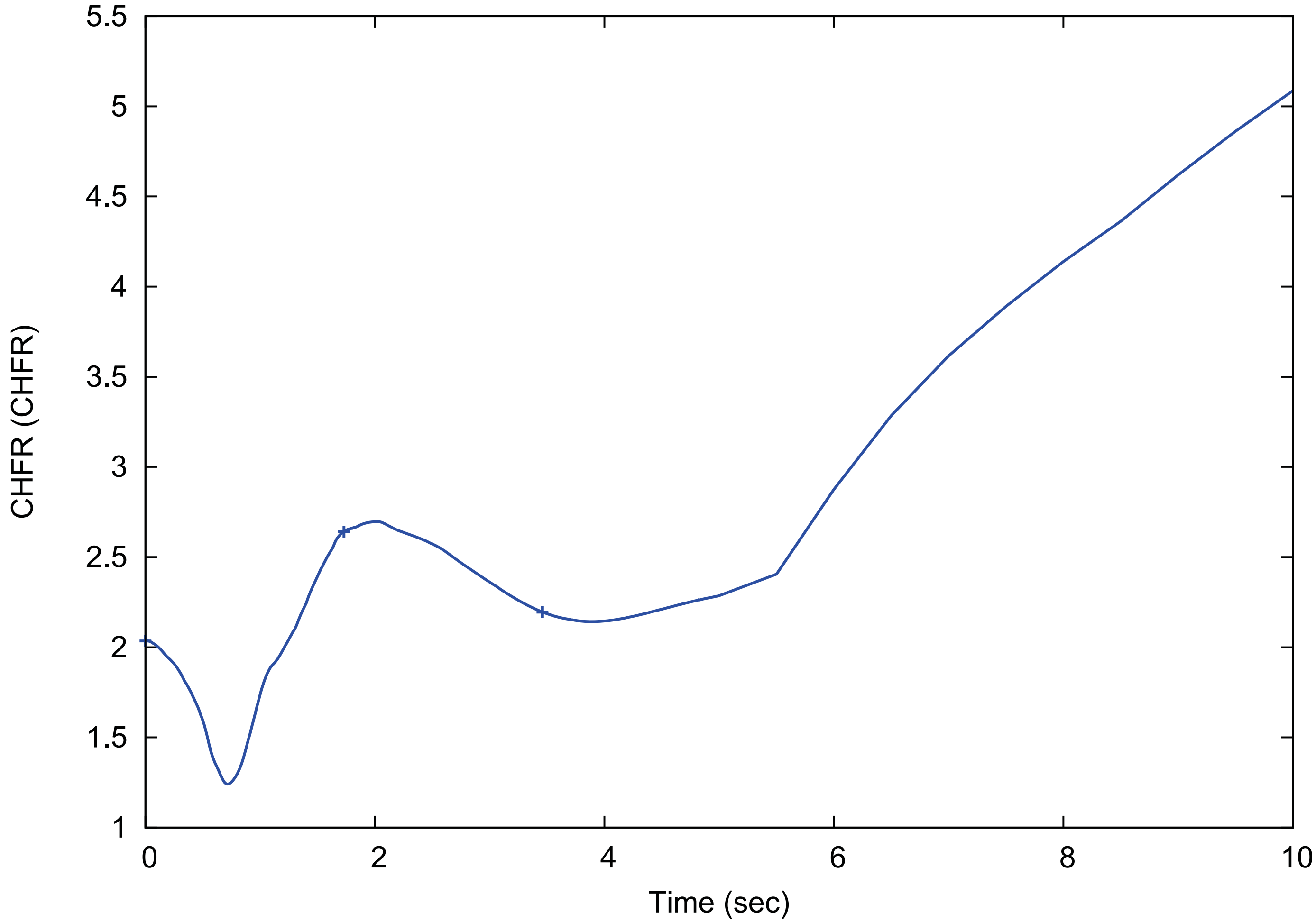

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Figure 15.6-68: Inadvertent Operation of an Emergency Core Cooling System Valve – Critical Heat Flux Ratio