NuScale Power

RAIO-0717-54897

July 19, 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. 34 (eRAI No. 8785) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 34 (eRAI No. 8785)," dated May 26, 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. 8785:

- 15.06.05-1

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
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8785

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8785

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NuScale Power

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

eRAI No.: 8785
Date of RAI Issue: 05/26/2017

NRC Question No.: 15.06.05-1

10 CFR Part 50 Appendix K, I.C.1 - *Break Characteristics and Flow,* requires that a spectrum of possible pipe breaks be considered in the analyses of loss-of-coolant accidents (LOCAs). Section 15.6.5 of the NuScale Design Specific Review Standard states that, "a spectrum of LOCA break sizes is to be evaluated and the limiting break identified through sufficient analyses …" The applicant indicates in Section 15.6.5.1 of the Final Safety Analysis Report (FSAR) that a spectrum of break sizes were analyzed that were limited to the chemical and volume control system injection and discharge lines, the high point vent line, and the pressurizer spray supply line. NRC staff identified that the reactor vent valves, reactor recirculation valves, and control rod drive mechanism housing were not identified as being considered in the spectrum of possible break locations. This caused NRC staff to question whether the spectrum of pipe-breaks considered in Section 15.6.5 of the FSAR was sufficient to identify the limiting break. NRC staff relies upon the consideration of an adequate spectrum of pipe breaks to establish a finding that the limiting pipe break has been identified. Accordingly, NRC requests that NuScale provide sufficient evidence to justify that a sufficient break spectrum has been considered such that the limiting break size has been identified and that it meets the applicable acceptance criteria.

NuScale Response:

The loss-of-coolant accident (LOCA) analysis presented in FSAR Section 15.6.5 follows the methodology presented in Topical Report TR-0516-49422-P, "Loss-of-Coolant Accident Evaluation Model," Revision 0. A spectrum of possible pipe breaks were considered for the LOCA analysis, consistent with 10 CFR Part 50 Appendix K, I.C.1.a. The reactor coolant system (RCS) piping external to the reactor pressure vessel (RPV) is identified in FSAR Section 5.1.3.2 and consists of the RCS injection line, RCS discharge line, RPV high point degasification line, and two pressurizer spray line branches from a common spray header. The RCS piping identified in FSAR Section 5.1.3.2 was considered for possible pipe breaks for the LOCA analysis. Please see the response to RAI 15.06.05-02 (RAI 8786) for the spectrum of breaks analyzed in the LOCA analysis.

FSAR Section 3.9.3.2 identifies the ASME Class 1 pressure relief valves, which include the

RCS reactor safety valves and the emergency core cooling system (ECCS) valves, consisting of the reactor vent valves (RVVs) and reactor recirculation valves (RRVs). FSAR Section 3.9.3.2 also states that the ASME Class 1 pressure relief valves are located on the reactor pressure vessel (RPV) and are part of the reactor coolant pressure boundary (RCPB). The RCS reactor safety valves and the ECCS valves are Seismic Category I components and are designed as ASME BPVC Section III Class 1 components. FSAR Section 5.1.3.6 also states that the RVVs and RRVs are part of the RCPB, similar to the reactor safety valves discussed in FSAR Section 5.1.3.5. FSAR Section 5.2.2.5 and FSAR Section 6.3.2.2 state that the ECCS valves are welded directly to the RPV nozzle safe ends and their stress analysis is performed in accordance with the requirements for ASME Class 1 components.

10 CFR 50.46(c) defines LOCAs as: "hypothetical accidents that would result from the loss of reactor coolant ... from breaks in pipes in the reactor coolant pressure boundary." In promulgating the final 10 CFR 50.46 rule, the Commission confirmed that "The wording of the definition of a loss-of-coolant accident has been modified to conform to its long-accepted usage, limiting it to breaks in pipes." 6 AEC 1085, 1093 (1973). Branch Technical Position (BTP) 3-3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, provides the definition of piping as:

> Piping is a pressure retaining component consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers).

There are no straight or curved pipe or pipe fittings, as defined by BTP 3-3, between an ECCS valve and the nozzle. Rather, the RVVs and RRVs are welded directly to the RPV nozzle safe ends. ASME code jurisdictional boundaries are established in ASME Design Specifications in accordance with Section III of the ASME Boiler and Pressure Vessel Code, Subsection NCA, paragraph NCA-3254. NuScale's ASME Design Specification for the RPV specifies that for the ECCS valve nozzle locations, the safe ends, the safe-end-to-nozzle welds, and the valve-to-safe-end welds are part of the RPV and analyzed as such. Therefore, the RVVs and RRVs are not required to be analyzed as a LOCA break location per 10 CFR Part 50 Appendix K, I.C.1.a and not part of the LOCA break spectrum. Although NuScale does not consider a break of the RRV and RVV in the LOCA spectrum based on the discussion above, an inadvertent operation of an ECCS valve is analyzed as a design basis event in FSAR Section 15.6.6.

The control rod drive mechanism (CRDM) pressure housings are integral portions of the RPV. The CRDM housings are welded to nozzles that are integrally forged as part of the RPV head. The safe end to nozzle welds and safe end to control rod housing welds are inspected to ASME Class 1 requirements, as stated in FSAR Section 4.6.2. FSAR Section 3.5.1.2 states that a CRDM housing failure is non-credible. The CRDM housing is a Class 1 appurtenance per ASME Section III and is designed in accordance with 10 CFR 50.55a. FSAR Section 3.9.3.1.2 describes the qualifications of the portions of the CRDM providing a RCPB function and FSAR Section 4.5.1.1 and FSAR Section 5.2.3 describe the materials for the pressure boundary portions of the CRDM.

Although NuScale does not consider a CRDM housing failure in the LOCA spectrum based on the discussion above, the inner diameter of the CRDM nozzle is bounded by the flow area of an inadvertent opening of a reactor vent valve analyzed in FSAR Section 15.6.6. In addition, consistent with Standard Review Plan 15.4.8, a failure of the CRDM housing is postulated to provide a limiting reactivity event. The rod ejection accident is analyzed in FSAR Section 15.4.8.

Impact on DCA:

There are no impacts to the DCA as a result of this response.