RAIO-0419-65197

April 11, 2019

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. 518 (eRAI No. 9659) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 518 (eRAI No. 9659)," dated March 04, 2019

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. 9659:

- 19-39

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 Rebecca Norris at 541-452-7539 or at rnorris@nuscalepower.com.

Sincerely,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9659

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NuScale Response to NRC Request for Additional Information eRAI No. 9659

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

eRAI No.: 9659

Date of RAI Issue: 03/04/2019

NRC Question No.: 19-39

Regulatory basis

10 CFR 52.47(a)(27) requires a description of the design-specific probabilistic risk assessment (PRA) and its results.

Discussion

Standard Review Plan, Chapter 19.0, Revision 3, page 19.0-22 identifies the need for in-depth NRC review of refueling operations for small, modular reactors, which are different from traditional LWRs, to ensure that the PRA model is of acceptable scope and level of detail. This same page also requires staff to verify that applicants for plants with multiple modules use a systematic process to identify accident sequences, including significant human errors that could lead to core damage or large release from multiple modules.

Section 19.1.7.4 of the NuScale Design Certification Application (DCA) discusses how a module dropped during refueling transport might impact other modules. Rev. 1 of the DCA states that if the module is dropped on an operating module near the top, it could damage the DHRS piping or heat exchangers. In Rev. 2 of the DCA, NuScale added that "additional pipe breaks may occur, leading to a [chemical and volume control system] CVCS line break outside containment." Additionally, Rev. 1 of the DCA states that if the operating module was struck near the bottom, the safety systems would remain nominally available, whereas Rev. 2 replaced this conclusion with "the collision is expected to cause a torque about the module support lugs, resulting in similar stresses to the piping on top of the operating module." The risk insights from this evaluation, which is the same in both revisions, are that a dropped module may incur core damage while the struck modules incur initiating events at full power.

Because Rev. 2 of the DCA postulates additional damage to the operating module beyond what was described in Rev. 1, the staff needs additional information to conclude the qualitative multi-module risk assessment is technically adequate and complete.

Request for Additional Information

Provide justification that multi-module risk insights for the struck module that is assumed to be operating at full power are unaffected by the additional damage described in Rev. 2 of the DCA. Specifically, describe which pipes in the CVCS, decay heat removal system and the containment flooding and drain system are assumed to fail and why. Also explain if the capability of the containment isolation valves to close is compromised, given that the strike to the operating module has sufficient force to cause pipe breaks.

NuScale Response:

The risk insights are unaffected by the changes in the description of potential multiple-module effects included in FSAR Revision 2. As described in FSAR Section 19.1.7.4, the effects of a module being struck by a dropped module are determined by engineering judgment. Accordingly, the modified FSAR wording in Revision 2 is intended to depict a range of possible effects on an operating module rather than a prediction of a specific effect.

Potential damage to the decay heat removal system (DHRS) heat exchangers is identified in the FSAR because the heat exchangers are located external to the containment; damage to these heat exchangers could affect secondary side heat removal, which is considered in the secondary side line break initiating event TGS---FMSLB-UD, described in FSAR Table 19.1-8.

Potential damage to the chemical and volume control system (CVCS) piping is identified because it could result in a pathway from the reactor coolant system to outside of containment, as considered in the CVCS pipe break outside containment initiating events CVCS--ALOCA-COC and CVCS--ALOCA-LOC, described in FSAR Table 19.1-8. Several conditions are necessary to result in such an impact-induced pipe break on an operating module. For example, in a postulated "controlled" module drop (i.e., the module is still attached to the module lifting adapter and crane) the dropped module is likely to remain vertical, precluding impact on an operating module. In a postulated "uncontrolled" module drop (e.g., catastrophic failure of both of the redundant rope reeving systems) which could result in module tipping, the geometry of the refueling pool, operating bay, and bay walls is such that a dropped module is unlikely to impact an operating module; additionally, the steel module platform columns and cross braces minimize the potential for a significant impact.

Considering only the probability of a load drop, the contribution of a potential module drop to the initiating event frequency of an operating module is judged to be negligible both in absolute terms and in comparison to the frequency of a randomly occurring pipe break outside of containment. The frequency of a dropped module is about 1E-7 per year as indicated in Table 19.1-68 and the initiating event frequency for a CVCS pipe break outside containment is about 4E-4 per module critical year as indicated in FSAR Table 19.1-8.

As discussed in FSAR Section 6.2.4.1, there are two containment isolation valves (CIVs) in each line that penetrates the containment boundary and connects to the reactor coolant pressure boundary. The CIVs on the CVCS line are open during power operation. The CIVs fail closed on loss of power or loss of hydraulic pressure and are located on top of the containment vessel head and under the module platform. Thus, if failure of a CVCS line on an operating module were postulated in response to a dropped module, impact damage to piping would likely occur downstream of the CIVs, and therefore the probability of failure-to-close of the CVCS CIVs is judged to be small. Because the containment flooding and drain system (CFDS) is normally isolated during module operation (i.e., the CFDS CIVs are closed) and not connected to the reactor coolant system, a postulated impact-induced CFDS pipe break would not lead to a loss of coolant or compromise the ability of the CIVs to maintain containment integrity.

Impact on DCA:

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