NUSCALE POWER™

RAIO-0318-59249

March 22, 2018

Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 78 (eRAI No. 8892) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 78 (eRAI No. 8892)," dated June 30, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 78 (eRAI No. 8892)," dated August 22, 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. 8892:

- 19-14

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 Supplemental Response to NRC Request for Additional Information eRAI No. 8892

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NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8892

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

eRAI No.: 8892
Date of RAI Issue: 06/30/2017

NRC Question No.: 19-14

Title 10 Code of Federal Regulations (CFR) 52.47(a)(27) states that a design certification application must contain an final safety analysis report (FSAR) that includes description of the design-specific probabilistic risk assessment (PRA) and its results. In accordance with the Statement of Consideration (72 FR 49387) for the revised 10 CFR Part 52, the staff reviews the information contained in the applicant's FSAR Chapter 19, issues requests for additional information (RAIS) and conducts audits of the complete PRA (e.g., models, analyses, data, and codes) to obtain clarifying information as needed. The staff uses guidance contained in Standard Review Plan (SRP) Chapter 19.0 Revision 3, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors." In accordance with SRP Chapter 19.0 Revision 3, the staff determines whether:

> "The technical adequacy of the PRA is sufficient to justify the specific results and risk insights that are used to support the [Design Certification] DC or [Combined License] COL application. Toward this end, the applicant's PRA submittal should be consistent with prevailing PRA standards, guidance, and good practices as needed to support its uses and applications and as endorsed by the [Nuclear Regulatory Commission] NRC (e.g., [Regulatory Guide] RG 1.200)."

The staff has reviewed the information in the FSAR and examined additional clarifying information from the audit of the complete PRA and determined that it needs additional information to confirm the validity of certain assumptions used in the flooding PRAs. The supporting requirements in the American Society of Mechanical Engineers / American Nuclear Society (ASME/ANS) PRA standard include provisions for documenting sources of model uncertainties and related assumptions. Please address the following questions.

a. FSAR Table 19.1-49, "Assessment of Flood Areas Containing Equipment Modeled in the Probabilistic Risk Assessment," describes the reactor building areas that include flood protection design features to protect equipment from propagating floods. Review of supporting audit information suggests that the required level of flooding protection is determined based on the assumed time available for the operator to successfully isolate the flood source. Please confirm the staff's understanding or provide an alternative explanation.

Additionally, assuming that the staff understands correctly and considering (1) the uncertainties introduced by the current level of plant design as cited in the FSAR (such as the lack of design detail on protective and mitigative features and detailed pipe routing information) and (2) the PRA should consider scenarios beyond the design basis, please explain how operators will always successfully isolate any flood sources in the reactor building.

b. FSAR Table 19.1-48, "Internal Flooding Sources," indicates the Reactor Building Spray System as a potentially significant flood source. The staff reviewed the FSAR and associated audit documentation and was unable to locate information on potential flooding scenarios associated with this flood source. Please describe the potential flooding scenarios associated with this flood source, considering as applicable, the associated potential propagation paths, equipment damage, flooding protection and mitigation features, and operator actions.

c. FSAR Section 19.1.5.4.1 states:

"An external flood could initiate a [Loss of Offsite Power] LOOP or [Loss of Direct Current] LODC because of flooding in areas containing [highly reliable DC power system] EDSS or [13.8 kV and switchyard system] EHVS components."

This statement implies that the EDSS and the EHVS equipment is assumed to be unprotected from floods. Please discuss why flooding protection features assumed to be available for internal flooding scenarios are assumed not to be available for external flooding scenarios.

NuScale Response:

NuScale is supplementing its response to RAI 8892 (Question 19-14) provided in letter RAIO-0817-55577, dated August 22, 2017. This supplemental response is provided in response to discussions with the NRC in a public meeting held on February 13, 2018.

This supplemental response adds an assumption to FSAR Table 19.1-54. The additional assumption is consistent with the response provided in letter RAIO-0817-55577 which stated that the PRA is based on the representative internal flooding analysis described in FSAR Section 3.4.1. To identify affected equipment, the representative analysis uses flood volumes based on the assumption that plant personnel will isolate the flood source (i.e., the flooding source does not continue indefinitely).

Impact on DCA:

Table 19.1-54 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 19-14S1

Table 19.1-54: Key Assumptions for the Internal Flooding PRA

Assumption	Basis
Buildings that are not expected to contain flood sources or are not expected to result in a plant trip are not considered.	Engineering judgment
Pipe routing and physical locations of some equipment are not defined at the design certification stage; flood locations are assumed from plant drawings.	Engineering judgment
An internal flood is capable of resulting in a plant upset and transient initiating event.	Engineering judgment
Flooding frequencies are based on generic data for turbine and auxiliary buildings, including human-induced mechanisms. A lognormal distribution is assumed.	Engineering judgment
No credit is taken for floor drains therefore, maximum flood heights are considered.	Bounding assumption
Operator actions to isolate a flooding event are not modeled. The time required to isolate a flood, however, is used to establish the volume of water involved in the event and establish the depth of water involved in the flood.	Engineering judgment
Flooding effects for equipment located inside the containment are not considered.	Equipment located in containment is designed to operate in harsh environments, including LOCAs.
Safety-related and risk significant equipment is protected from flooding effects.	Although not defined at design certification, flood protection (e.g., flood door, splash protection) for safety-related and risk significant equipment is a design requirement.
Equipment that is not safety-related or risk significant is exposed to flooding in the area where it is located; flood-induced failure mechanisms include spray and submergence.	Bounding assumption
Passive components such as piping, tanks, heat exchangers, manual valves, check valves, relief valves, strainers and filters are not susceptible to flood damage	Passive equipment does not require control to operate
Electrical equipment is susceptible to flood damage which occurs instantaneously when the lowest portion of the equipment is submerged. The most likely failure mechanism for flood water damage is a short-to ground, which results in an open-circuit failure mode.	Bounding assumption
If subjected to a flood, motor operated valves fail as-is and solenoid and air-operated valves fail to their de-energized position.	Engineering judgment
A flood in the RXB will prevent operations from establishing makeup with the CVCS or CFDS.	Bounding assumption
Consistent with the internal events analysis, high stress was considered for operator actions.	Engineering judgment
Equipment affected by a flood is based on the analysis summarized in Section 3.4.1.	Common engineering practice