RAIO-0717-55142

July 31, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

**SUBJECT:** NuScale Power, LLC Response to NRC Request for Additional Information No. 51 (eRAI No. 8854) on the NuScale Design Certification Application

**REFERENCE:** U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 51 (eRAI No. 8854)," dated June 02, 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 Questions from NRC eRAI No. 8854:

- 19-3
- 19-4
- 19-5

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

Distribution: Gregory Cranston, NRC, OWFN-8G9A
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8854

**Enclosure 1:**

NuScale Response to NRC Request for Additional Information eRAI No. 8854

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NuScale Power

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

**eRAI No.:** 8854
**Date of RAI Issue:** 06/02/2017

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**NRC Question No.:** 19-3

10 CFR 52.47(a)(27) states that a design certification (DC) application must contain a final safety analysis report (FSAR) that includes a description of the design-specific probabilistic risk assessment (PRA) and its results. SECY-93-087, Section II.N, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs,” Agency-Wide Document Access and Management System (ADAMS) Accession No. ML003708021, dated April 2, 1993, and the related staff requirements memorandum, ADAMS Accession No. ML003708056, dated July 21, 1993, also provide guidance on use of a sequence-level seismic margins analysis (SMA) in lieu of a seismic PRA. In particular, the Commission approved the Staff’s recommendation that since a seismic PRA cannot be performed until a plant is built, DC applicants should use PRA insights to support a margin-type assessment of seismic events. 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, and issues requests for additional information (RAI) 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: 1) The applicant has performed a PRA-based SMA to determine the seismic capacity of the plant and for each sequence that may lead to core damage or large release; and 2) The design-specific plant system and accident sequence analysis for a PRA-based SMA is performed in accordance with, at a minimum, the Capability Category I requirements of Section 5-2.3 of Part 5 of the ASME/ANS PRA Standard ,with the exceptions that the analysis does not need to be based on site-specific and plant-specific information and does not have to rely on an as-built and as-operated plant.

The staff has reviewed the information in the FSAR and examined additional clarifying information from an audit of the complete PRA-based SMA and determined that it needs additional information to confirm that the system modeling portion of the PRA-based SMA was performed in accordance with, at a minimum, the Capability Category I requirements for supporting requirements SPR-B8 and B9 of Section 5-2.3 of Part 5 of the ASME/ANS PRA Standard. These requirements deal with modeling system recovery and restoration. Restoration of safety equipment by plant personnel can be inhibited by any of several types of causes related to a seismic event.  Treatment of the impact of earthquake damage on system restoration was not discussed in the FSAR.   Please describe the extent to which system recovery was credited in the PRA.  Please identify systems recovered and the accident sequences in which the recovery of systems, if any, was credited.  Please justify credit for any system restorations.

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**NuScale Response:**

System recovery is not credited in the SMA, as indicated in Table 19.1-40. The passive design for long-term cooling and the absence of reliance on electrical power to maintain safe shutdown means that system recovery is not an important consideration in reducing the potential for core damage after a seismic event. For clarity, the text in Section 19.1.5.1.2 has been modified to state that system recovery is not credited in the SMA.

**Impact on DCA:**

FSAR Section 19.1.5.1.2 has been revised as described in the response above and as shown in the markup provided in this response.

resulting in their actuation on loss of power or control. As such, very few component failures have the potential to contribute to seismic risk.

RAI 19-3, RAI 19-4, RAI 19-5, RAI 19-10

Moreover, component fragilities reported in Table 19.1-38 show very low seismic failure probabilities~~a high degree of component seismic robustness~~. The fail-safe design of PRA-critical components means that the only credible seismic failures of the valves required to achieve safe shutdown involves physical deformation of the valves themselves, which only occurs under extreme stresses ~~concentrations~~. As a result, component failures (either seismic or random) do not contribute significantly to the potential for core damage or releases following a seismic event. Rather, similar to the internal events PRA, CCF of key functions have the most potential for controlling risk, e.g., common cause events leading to failure of reactor trip, ECCS valve CCFs and failures to isolate containment (in response to seismically induced SGTF or breaks~~pipe break~~ outside containment).

Significant Operator Actions

The SMA model implements HFE probabilities in the same manner as the internal events PRA. Individual system-specific HFE events are first inserted into cutsets using sequence logic; no seismic-specific operator actions were added to the SMA models.

The internal events human error probabilities of each HFE in the SMA models are multiplied by a factor of 5 for the SMA, to account for the assumed "extreme stress" environment associated with any seismic event (per SPAR-H methodology, NUREG/CR-6883, Reference 19.1-22). This is performed regardless of ground motion, meaning the HEPs at lower ground motion levels are conservative.

RAI 19-3

The NuScale design incorporates a significant amount of passive safety features, requiring ~~little or~~ no operator intervention to initiate or maintain operation. As a result, seismic cutsets containing HFEs also include other seismically induced or random failures that limit the importance of operator actions. ~~Despite the increase in seismic HEPs described above,~~There are no recovery actions credited in the SMA. Although the HEPs are increased for the SMA, there are no operator actions that play a substantial role in contributing to, or mitigating, the conditional core damage probability results for the SMA.

Key Assumptions

Table 19.1-40 summarizes the key assumptions associated with the SMA.

Uncertainties

Parameters representing aleatory and epistemic uncertainty are used directly in evaluating the plant-level HCLPF. Each SSC in the SMA is modeled with a lognormal uncertainty distribution using randomness ($\beta_r$) and epistemic uncertainty ($\beta_u$) parameters. For PRA-critical SSC that are the subject of detailed fragility,

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**NRC Question No.:** 19-4

10 CFR 52.47(a)(27) states that a design certification (DC) application must contain a final safety analysis report (FSAR) that includes a description of the design-specific probabilistic risk assessment (PRA) and its results. SECY-93-087, Section II.N, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," Agency-Wide Document Access and Management System (ADAMS) Accession No. ML003708021, dated April 2, 1993, and the related staff requirements memorandum, ADAMS Accession No. ML003708056, dated July 21, 1993, also provide guidance on use of a sequence-level seismic margins analysis (SMA) in lieu of a seismic PRA. In particular, the Commission approved the Staff's recommendation that since a seismic PRA cannot be performed until a plant is built, DC applicants should use PRA insights to support a margin-type assessment of seismic events. 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, and issues requests for additional information (RAI) 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: 1) The applicant has performed a PRA-based SMA to determine the seismic capacity of the plant and for each sequence that may lead to core damage or large release; and 2) The design-specific plant system and accident sequence analysis for a PRA-based SMA is performed in accordance with, at a minimum, the Capability Category I requirements of Section 5-2.3 of Part 5 of the ASME/ANS PRA Standard ,with the exceptions that the analysis does not need to be based on site- specific and plant-specific information and does not have to rely on an as-built and as-operated plant.

The staff has reviewed the information in the FSAR and examined additional clarifying information from an audit of the complete PRA-based SMA and determined that it needs additional information to confirm that the PRA-based SMA was performed in accordance with, at a minimum, the Capability Category I requirements for high level requirement HLR-SPR-A of Section 5-2.3 of Part 5 of the ASME/ANS PRA Standard. It is clear to the staff that the SMA accounts for seismic failures due to the seismic load exceeding the seismic capacity of an SSC. The staff could not identify any information in the FSAR nor in clarifying documents as to whether the SMA also accounts for failures of SSCs that survive the loads from the earthquake, but are damaged from interaction with a SSC that does not have sufficient capacity to survive the seismic hazard (e.g., a seismic class I SSC damaged by a non-seismic class SSC). Please describe how such seismically induced failures of robust SSCs, if any, are treated in the SMA.

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**NuScale Response:**

Consideration of potential failure of seismically qualified components due to physical interaction with a nonseismically qualified SSC is evaluated consistent with the definition of “spatial interaction” as defined by the ASME/ANS PRA standard “spatial interaction”:

a) Proximity effects

Safe shutdown of a NuScale Power Module (NPM) is ensured by opening of the RSVs, combined with successful passive ECCS valve operation, when there is no loss of coolant outside the containment boundary. These components have very high seismic capacities and are physically shielded from nonseismically qualified SSCs by the seismically qualified CNV. These components fail safe on loss of power and are not located in proximity to nonseismically qualified components.

b) Structural failure and falling

The potential for failure and falling interactions between surviving seismically qualified SSCs and seismically failed SSCs is limited by the nature of the NuScale design. The NPM is physically protected by the pool water, pool walls, bay walls, and, during power operation, the bioshield. Seismically-induced damage to the bay walls and biodshield is modeled in the SMA; the SMA demonstrates that these structures have higher HCLPF values than potential components that could fail due to a seismic event. Thus, these structures would provide a physical barrier between potentially failed components and the NPM.

When the bioshield is removed from an operating bay prior to NPM transport for refueling, piping penetrations atop the CNV, as well as the DHRS piping and heat exchangers on the side of the NPM, could be impacted by a falling or swinging object. However, the module is shut down and flooded prior to its bioshield being removed. In this configuration, safe shutdown is maintained by conduction from the RPV through to the CNV and reactor pool. In addition, piping penetrations atop the module are designed to be protected from pipe whip and jet projections which may be postulated from a seismically caused failure.

c) Flexibility of attached lines and cables

Seismically-induced pipe breaks outside containment are modeled in the SMA and encompass the effects of pipe leaks caused by stresses induced by structural displacements or failing objects.

The NPM is not precluded from achieving safe shutdown as a result of loss of electrical power or signaling logic. As such, the SMA model does not credit systems requiring electrical power at ground motion levels sufficient to cause both loss of offsite power and failure of backup power sources.

Section 19.1.5.1.1.5 has been added to reflect this discussion.

**Impact on DCA:**

FSAR Section 19.1.5.1.1.5 has been revised as described in the response above and as shown in the markup provided in this response.

~~The~~In summary, the SMA event trees terminate in ~~one of four end states~~:

- OK: No core damage
- ~~CD: Core Damage~~
- ~~NR: Negligible Release~~Transfer to another event tree
- ~~LR: Large Release~~Transfer to the Level 2 event tree.

RAI 19-4

### 19.1.5.1.1.5 Effects of Seismically Failed SSCs on Surviving SSCs

Potential failures of seismically qualified components due to physical interaction with a nonseismically qualified SSCs are evaluated consistent with the definition of "spatial interaction," as defined by the ASME/ANS PRA standard:

RAI 19-4

a) Proximity effects

Safe shutdown of an NPM is ensured by opening of the RSVs, combined with successful passive ECCS valve operation, when there is no loss of coolant outside the containment boundary. These components have very high seismic capacities and are physically shielded from nonseismically qualified SSCs by the seismically qualified CNV. These components fail safe on loss of power and are not located in proximity to nonseismically qualified components.

RAI 19-4

b) Structural failure and falling

The potential for failure and falling interactions between surviving seismically qualified SSCs and seismically failed SSCs is limited by the nature of the NuScale design. The NPM is physically protected by the pool water, pool walls, bay walls, and, during power operation, the bioshield. Seismically-induced damage to the bay walls and bioshield is modeled in the SMA; the SMA demonstrates that these structures have higher HCLPF values than potential components that could fail due to a seismic event. Thus, these structures would provide a physical barrier between potentially failed components and the NPM.

RAI 19-4

When the bioshield is removed from an operating bay prior to NPM transport for refueling, piping penetrations atop the CNV, as well as the DHRS piping and heat exchangers on the side of the NPM, could be impacted by a falling or swinging object. However, the module is shut down and flooded prior to its bioshield being removed. In this configuration, safe shutdown is maintained by conduction from the RPV through to the CNV and reactor pool.

RAI 19-4

c) Flexibility of attached lines and cables

Seismically-induced pipe breaks outside containment are modeled in the SMA and encompass the effects of pipe leaks caused by stresses induced by structural displacements or failing objects.

RAI 19-4

The NPM is not precluded from achieving safe shutdown as a result of a loss of electrical power or signaling logic. As such, the SMA model does not credit systems requiring electrical power at ground motion levels sufficient to cause both loss of offsite power and failure of backup power sources.

#### 19.1.5.1.2 Results from the Seismic Risk Evaluation

RAI 19-3, RAI 19-4, RAI 19-5

Seismic risk is quantified in terms of a plant-level HCLPF g-value. SMAs are required to show that the plant level HCLPF is greater than 1.67 times the design basis SSE, which equates to a 0.84g peak ground acceleration for NuScale.

The SMA cutsets are assessed using the MIN-MAX method to determine the sequence level fragility. In this method, a group of inputs combined using OR logic (such as different sequences) is assigned the minimum fragility of the group. Conversely, inputs combined with AND logic (such as seismic events within a sequence) are determined by the maximum fragility of the group. The MIN-MAX method is evaluated at the sequence level. This means that the lowest HCLPF cutset value within a sequence determines the seismic margin. In a cutset containing multiple seismic failures, the highest HCLPF value determines the cutset HCLPF.

RAI 19-3, RAI 19-4, RAI 19-5

The resulting HCLPF acceleration for the NuScale design is 0.88g. Structural events are the leading contributor to the seismic margin because of their immediate consequences and relatively low PGA-grounded median capacities as compared to component failures. Table 19.1-35 summarizes the fragility analysis for each of the structural events. Each of the structural event parameters has been calculated using design specific fragilities. From Table 19.1-35, the structural event with the lowest HCLPF is corbel support bearing failure ~~at 0.68g~~. While this structural event results in a ~~pipe~~ break outside containment, it is isolable and the seismic capacity of the isolation valves results in a much higher HCLPF for these sequences ~~involving the corbel bearing failure~~. This leaves ~~corbel shear~~reactor bay wall failure and RBC failure as having the limiting HCLPFs. The SMA assumes that failure of major structures leads to sufficient damage to the modules such that core damage and a large release would result.

Significant Sequences

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**NRC Question No.:** 19-5

10 CFR 52.47(a)(27) states that a design certification (DC) application must contain a final safety analysis report (FSAR) that includes a description of the design-specific probabilistic risk assessment (PRA) and its results. SECY-93-087, Section II.N, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs,” Agency-Wide Document Access and Management System (ADAMS) Accession No. ML003708021, dated April 2, 1993, and the related staff requirements memorandum, ADAMS Accession No. ML003708056, dated July 21, 1993, also provide guidance on use of a sequence-level seismic margins analysis (SMA) in lieu of a seismic PRA. In particular, the Commission approved the Staff’s recommendation that since a seismic PRA cannot be performed until a plant is built, DC applicants should use PRA insights to support a margin-type assessment of seismic events. 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, and issues requests for additional information (RAI) 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: 1) The applicant has performed a PRA-based SMA to determine the seismic capacity of the plant and for each sequence that may lead to core damage or large release, has shown that the design satisfies the Commission's goal to have seismic margin greater than or equal to 1.67 times the safe shutdown earthquake; and 2) The design-specific plant system and accident sequence analysis for a PRA-based SMA is performed in accordance with, at a minimum, the Capability Category I requirements of Section 5-2.3 of Part 5 of the ASME/ANS PRA Standard ,with the exceptions that the analysis does not need to be based on site-specific and plant-specific information and does not have to rely on an as-built and as-operated plant.

The staff has reviewed the information in the FSAR and examined additional clarifying information from an audit of the complete PRA-based SMA and determined that it needs additional information to confirm that the plant level seismic capacity determined by the applicant is in conformance with the Commission's goal of being at least 167% of the safe shutdown earthquake level. The applicant has determined the plant level seismic capacity to be.88g which is only slightly larger than .84g which corresponds to 1.67 times the safe shutdown earthquake for the NuScale design. The discussion in Section 19.1.5.1.2 of the FSAR indicates that uncertainties in the SMA were assessed, but no results are given. Please provide a description of the uncertainty analysis that was conducted and a summary of the results and include this information in the FSAR.

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**NuScale Response:**

After the plant-level HCLPF is determined, uncertainty analysis is performed by setting the seismic demand to the HCLPF level, sampling from the probability distributions for all all events (fragilities and random events), and re-calculating the CCDP. Results are compared to the HCLPF definition (95 percent confidence of less than 5 percent probability of failure or 1 percent failure probability on the mean fragility curve).

The following table shows the calculated distribution results for the conditional probability of seismic core damage evaluated at the plant-level HCLPF of 0.88g.

**SMA Uncertainty Results**

| End State | Point Estimate | Mean | 5% | Median | 95% |
|---|---|---|---|---|---|
| S-CD* | 7.86E-05 | 2.68E-02 | 3.09E-5 | 1.46E-03 | 1.36E-01 |
| *S-CD: Sequence with a seismic initiator terminating in core damage | | | | | |

These results show a mean conditional core damage probability that is higher than the 1E-2 theoretical CCDP value on the mean fragility curve. The 95 percent curve value is similarly higher than the 5 percent failure probability HCLPF definition. Three factors explain this difference:

- βr and βu have different values for most seismic events. This difference causes failure probabilities to vary considerably between sampled seismic events. This results in some seismic events contributing more to seismic conditional core damage probability in some sample iterations.
- The MIN-MAX SMA method uses only the lowest-fragility cutset as HCLPF input, whereas the uncertainty calculation simulates all possible cutset outcomes. As such, the random sampling captures the contribution of structural events other than the controlling one. This effectively increases the conditional core damage probability higher than what would be predicted from the HCLPF.
- The results include contributions to CCDP uncertainty from random failure uncertainties as well as seismic failures. These uncertainties do not contribute to the plant-level HCLPF evaluated by the MIN-MAX method.

Overall, the CCDP uncertainty distribution shows reasonable agreement between the controlling failure HCLPF evaluated with the MIN-MAX method, and the mean and median CCDP values. The results confirm that the HCLPF value is reasonable. The SMA uncertainties discussion in FSAR Section 19.1.5.1.2 has been modified to summarize these uncertainty results.

**Impact on DCA:**

FSAR Section 19.1.5.1.2 has been revised as described in the response above and as shown in the markup provided in this response.

uncertainty parameters are also assigned to each sub-factor that contributes to the overall safety factor.

The SMA contains uncertainty from many sources, including:

- Ground motion variability
- Uncertainty in soil-structure interaction
- Uncertainty in structural response factors
- Spectral shape (motion frequency) uncertainty
- SSC capacity uncertainty (material strength and inelastic energy absorption)

The modeling of seismic uncertainty is divided into two composite factors, $\beta_r$ and $\beta_u$. Both $\beta_r$ and $\beta_u$ are included in each seismic event, along with the median capacity $A_m$.

RAI 19-5

In addition to parametric uncertainty, the completeness of the selection of SSCs is a consideration in the performance of the SMA.

With respect to evaluation of structures, the SMA specifically considers the capacity and effects of failure of:

- Structures directly in contact with the reactor module
- Structures directly connected to the module interface
- Structures located above the module

RAI 19-5

After the plant-level HCLPF is determined, uncertainty analysis is performed by setting the seismic demand to the HCLPF level, sampling each event in the SMA (fragilities and random events), and re-calculating the CCDP. Results are compared to the HCLPF definition (95 percent confidence of less than 5 percent probability of failure or 1 percent failure probability on the mean fragility curve).

RAI 19-5

The CCDP uncertainty distribution demonstrated agreement between the controlling failure HCLPF (seismic restraint weldment) evaluated with the MIN-MAX method. Results from the uncertainty analysis confirm that the HCLPF value is reasonable.

Sensitivity Studies

No sensitivities were performed for the SMA.

Key Insights