November 27, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 83 (eRAI No. 8899) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 83 (eRAI No. 8899)," dated July 07, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 83 (eRAI No.8899)," dated September 01, 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 Questions from NRC eRAI No. 8899:

- 19.01-1
- 19.01-5
- 19.01-14
- 19.01-15
- 19.01-16

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. 8899

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8899

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eRAI No.: 8899
Date of RAI Issue: 07/07/2017

NRC Question No.: 19.01-1

10 CFR 52.47(a)(27) states that a Design certification application (DCA) must contain a Final Safety Analysis Report (FSAR) that includes a description of the design-specific probabilistic risk assessment (PRA) and its results. In SECY 93-087, the Commission approved use of the seismic margin approach (SMA) for DCAs in lieu of a seismic PRA. Per the guidance in DC/COL-ISG-20, “Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors,” and Regulatory Guide 1.200, the seismic fragility calculation should use the response spectrum shape defined as the DCA’s Certified Seismic Design Response Spectra (CSDRS).

In FSAR Section 19.1.5.1.1.2, “Seismic Input Spectrum,” only the peak ground acceleration (pga) for the seismic input spectrum is described. The staff requests that the applicant include a definition of the review level earthquake (RLE), including the shape and magnitude, in the FSAR. The applicant should also clarify the following FSAR statements related to the seismic input.

- In FSAR Section 19.1.5.1.1.1, “Seismic Analysis Methodology and Approach,” the applicant states, “The SMA analysis must be performed relative to a review level earthquake of 1.67 times the safe shutdown earthquake (SSE).” The staff requests that the applicant clarify that the RLE is defined relative to the CSDRS.
- In FSAR Section 19.1.5.1.1.2, “Seismic Input Spectrum,” the applicant states, “The component fragility is referenced to the peak ground acceleration defining the uniform hazard response spectra for a site, which is the [safe shutdown earthquake (SSE)].” An identical statement also exists on page 19.1-59 of the FSAR. Because a DCA does not contain site-specific information, the staff requests that the applicant describe the applicability of the uniform hazard response spectra (UHRS) to the PRA-based SMA.
- In the same FSAR statement referenced above, the staff notes that reference is made only to the component fragility. The staff requests that the applicant clarify that the RLE is the seismic input for the fragility evaluation of structures, systems, and components (SSCs).

NuScale Response:

Per discussion with the staff during a public meeting on October 3, 2017, the following information supplements the information provided in NuScale's original response.

The review level earthquake (RLE) spectral shape is defined relative to the certified seismic design response spectrum (CSDRS), as provided in FSAR Figure 3.7.1-1, with a scaling factor of 1.67. FSAR Section 19.1.5.1.1.1 has been modified to clarify this relationship.

Impact on DCA:

FSAR Section 19.1.5.1.1.1 has been revised as described in the response above and as shown in the markup provided in this response.

Section 19.1.5.1 through Section 19.1.5.5 address seismic, internal fire, internal flood, external flood and high-winds hazards, respectively.

COL Item 19.1-7: A COL applicant that references the NuScale Power Plant design certification will evaluate site-specific external event hazards, screen those for risk-significance, and evaluate the risk associated with external hazards that are not bounded by the design certification.

19.1.5.1 Seismic Risk Evaluation

Evaluation of the risk due to seismic events for a NuScale plant is performed using PRA-based seismic margin analysis (SMA). Section 19.1.5.1.1 describes this assessment and outlines the manner in which SMA is applied. Section 19.1.5.1.2 summarizes the results obtained from the PRA-based SMA for the NuScale design.

The scope of the SMA is the evaluation of seismic fragilities for SSC associated with a single module. A ground motion representing high confidence of low probability of failure (HCLPF) is derived for each SSC. Accident sequences from the PRA are solved to produce the combinations of seismic and random failures (cutsets) that could lead to core damage and large releases. Cutset level and plant level HCLPFs are then derived using the MIN-MAX method.

The SMA covers full power and LPSD operating conditions and includes Level 1 (core damage) and Level 2 (large release) consequences.

19.1.5.1.1 Description of the Seismic Risk Evaluation

There are two main tasks associated with performing a PRA-based SMA: seismic fragility analysis (structures and components), and seismic plant response analysis (accident sequence analysis and plant level response). The following sections summarize the SMA approach:

- Seismic Analysis Methodology and Approach (Section 19.1.5.1.1.1).
- Seismic Input Spectrum (Section 19.1.5.1.1.2).
- Seismic Fragility Evaluation (Section 19.1.5.1.1.3).
- Seismic Risk Accident Sequence and System Modeling (Section 19.1.5.1.1.4).

19.1.5.1.1.1 Seismic Analysis Methodology and Approach

RAI 19.01-1, RAI 19.01-1S1, RAI 19.01-7

The PRA-based SMA for the ~~NuScale power plant~~NPM (single module) is performed in accordance with the applicable NRC guidance documents DC/COL-ISG-020 (Reference 19.1-56), and with the applicable PRA-based SMA guidance in ~~the~~Part 5 of ASME-ANS Ra-Sa-2009 (Reference 19.1-2) as endorsed by RG1.200. As discussed in DC/COL-ISG-020, the purpose of a PRA-based SMA is to provide an understanding of significant seismic vulnerabilities and other seismic insights~~, thus establishing the seismic robustness of a standard design~~.

The SMA analysis must be performed relativeConsistent with DC/COL-ISG-020, the seismic margin is evaluated with respect to a review level earthquake (RLE) of 1.67 times the safe shutdown earthquake (SSE). The RLE spectral shape is defined relative to the certified seismic design response spectrum (CSDRS) as provided in Figure 3.7.1-1, with a scaling factor of 1.67. The peak ground acceleration of the CSDRS is the safe shutdown earthquake (SSE).

19.1.5.1.1.2 Seismic Input Spectrum

RAI 19.01-1

ComponentStructure, system, and component fragility is referenced to the peak ground acceleration definingof the uniform hazard response spectra for a siteCSDRS, which is the SSE (0.5g). The certified seismic design response spectra (CSDRS) envelopes this spectrum for the NuScale design with an SSE of 0.5g.

19.1.5.1.1.3 Seismic Fragility Evaluation

RAI 19.01-1S1,RAI 19.01-2, RAI 19.01-17

A seismic fragility analysis is completed as part of an SMA. Fragility describes the probability of failure of a component under specific capacity and demand parameters and their uncertainties. It should be noted that all SSCs modeled in the internal events PRA were included in fragility analysis, with the exception of basic events that are not subject to seismic-induced failure (e.g., phenomenological events, filters, control logic components). No pre-screening was performed to establish a seismic equipment list (SEL) or safe shutdown equipment list (SSEL). The terminology "PRA-critical" is used to denote SSCs that contribute to the seismic margin. Contributing SSCs are determined by applying the MIN-MAX method and the screening assumption described in Table 19.1-41.

RAI 19.01-2

Seismic capacities for PRA-critical structures and components modeled in the SMA are obtained by performing detailed fragility analysis using either the hybrid method or the separation of variables method described in Reference 19.1-21, Reference 19.1-57, and Reference 19.1-58. For non-critical components, fragilities are evaluated using generic capacity values and design-specific response spectra to calculate the demand.

RAI 19.01-5S1, RAI 19.01-10

The controlling failure mode of these structural events and their direct consequences are shown in Table 19.1-36. All structural events identified in Table 19.1-36, except corbel bearing failure, lead directly to core damage and large release. For components, seismic failures are either considered functional failures (all modes) or mapped to specific equivalent random failures (such as a valve failing to open on demand). The in-structure response spectra (ISRS) is produced at each SSC location using the CSDRS as input. Based on available

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NRC Question No.: 19.01-5

10 CFR 52.47(a)(27) states that a DCA must contain an FSAR that includes a description of the design-specific PRA and its results. In SECY 93-087, the Commission approved use of the SMA for DCAs in lieu of a seismic PRA.

The staff reviewed FSAR Tier 2, Section 19.1.5, and finds that the DCA lacks information on equipment qualified via tests. As described in Section 5.1.2 of ISG-20, a description of the procurement specifications (including the enhanced required response spectra (RRS)) should be provided in the DCA. The staff requests that the applicant address the RRS in the DCA or otherwise justify that the procured equipment qualified via tests will have adequate margin.

NuScale Response:

Per discussion with the staff during a public meeting on October 3, 2017, the following information supplements the information provided in NuScale's original response.

Consistent with DC/COL-ISG-020 (FSAR Reference 19.1-56), the seismic demand to equipment is developed from certified seismic design response spectrum (CSDRS)-based seismic inputs. NuScale does not use a test qualification methodology and required response spectra (RRS) to establish component fragility. The seismic demand is quantified using the location-specific in-structure response spectra (ISRS), which has been developed for the purpose of designing equipment with location-specific demand. FSAR Section 19.1.5.1.1.3 has been modified to clarify that the ISRS is used in lieu of the RRS in NuScale fragility calculations.

Impact on DCA:

FSAR Section 19.1.5.1.1.3 has been revised as described in the response above and as shown in the markup provided in this response.

The SMA analysis must be performed relativeConsistent with DC/COL-ISG-020, the seismic margin is evaluated with respect to a review level earthquake (RLE) of 1.67 times the safe shutdown earthquake (SSE). The RLE spectral shape is defined relative to the certified seismic design response spectrum (CSDRS) as provided in Figure 3.7.1-1, with a scaling factor of 1.67. The peak ground acceleration of the CSDRS is the safe shutdown earthquake (SSE).

19.1.5.1.1.2 Seismic Input Spectrum

RAI 19.01-1

ComponentStructure, system, and component fragility is referenced to the peak ground acceleration definingof the uniform hazard response spectra for a siteCSDRS, which is the SSE (0.5g). The certified seismic design response spectra (CSDRS) envelopes this spectrum for the NuScale design with an SSE of 0.5g.

19.1.5.1.1.3 Seismic Fragility Evaluation

RAI 19.01-1S1,RAI 19.01-2, RAI 19.01-17

A seismic fragility analysis is completed as part of an SMA. Fragility describes the probability of failure of a component under specific capacity and demand parameters and their uncertainties. It should be noted that all SSCs modeled in the internal events PRA were included in fragility analysis, with the exception of basic events that are not subject to seismic-induced failure (e.g., phenomenological events, filters, control logic components). No pre-screening was performed to establish a seismic equipment list (SEL) or safe shutdown equipment list (SSEL). The terminology "PRA-critical" is used to denote SSCs that contribute to the seismic margin. Contributing SSCs are determined by applying the MIN-MAX method and the screening assumption described in Table 19.1-41.

RAI 19.01-2

Seismic capacities for PRA-critical structures and components modeled in the SMA are obtained by performing detailed fragility analysis using either the hybrid method or the separation of variables method described in Reference 19.1-21, Reference 19.1-57, and Reference 19.1-58. For non-critical components, fragilities are evaluated using generic capacity values and design-specific response spectra to calculate the demand.

RAI 19.01-5S1, RAI 19.01-10

The controlling failure mode of these structural events and their direct consequences are shown in Table 19.1-36. All structural events identified in Table 19.1-36, except corbel bearing failure, lead directly to core damage and large release. For components, seismic failures are either considered functional failures (all modes) or mapped to specific equivalent random failures (such as a valve failing to open on demand). The in-structure response spectra (ISRS) is produced at each SSC location using the CSDRS as input. Based on available component design information, ISRS is used in lieu of required response spectra for fragility calculations.

Seismic Structural Events

Structural events are modeled as basic events in the PRA model with median failure acceleration and uncertainty parameters. Structural events differ from component failures in that they do not correspond to any random event in the internal events PRA. In nearly all cases, the consequences of structural events are assumed to lead to both core damage and large release without opportunity for mitigation. This is a simplifying assumption for modeling catastrophic failure mechanisms.

The selection of structural failures to model is based on a qualitative assessment of the external mechanisms that can damage the reactor module. Structures selected for analysis meet one of the following criteria:

- Structures directly in contact with the reactor module: This applies to the module passive support skirt ring attached to the reactor pool floor, and the lateral support lug-corbel interface;
- Structures directly connected to the module interface: The reactor bay walls, pool wall and pool floor. The latter two are bounded in terms of fragility by the RXB outer wall failure;
- Structures located above the module, where collapse could lead to physical damage to the module. These include the Reactor Building crane (RBC) and the bioshield. The roof of the RXB and the pool wall fragility is bounded by the outer wall fragility analysis.

Figure 1.2-5 provides perspective on the locations of structural failures included in the SMA.

Reactor Building Crane

The RBC is located over the reactor pool and is suspended by girders. It runs the length of the reactor pool and is used primarily for raising and transporting NPMs to and from the refueling bay.

RAI 19.01-14S1

The crane is designed with seismic restraints. ~~Catastrophic~~As illustrated in Figure 19.1-42, ~~bridge girder failure is preceded by failures of these bridge seismic restraints through yielding of the restraint weldments~~bridge girder failure cannot lead to catastrophic collapse without failure of the bridge seismic restraints. ~~Therefore, failure~~Failure of the bridge seismic restraints is the controlling failure mode by comparison to yielding of the bridge girder itself. The bounding consequence of crane failure is a collapse of the crane structure, which is assumed to impact the top of the module, and lead to core damage and large release. This modeling simplification is ~~required~~conservative because the bioshield, CNV and RPV integrity are not credited following a crane collapse.

Fragility calculations for the bioshield failure modes show that the bioshield controlling failure mode for both the single and double-stacked configurations is shearing of the bay wall anchor bolts.

Table 19.1-36 summarizes the fragility analysis for each of the structural events. Each of the structural event parameters has been calculated using design-specific fragilities.

Components

For the SMA, seismic failures of components are modeled in one of two ways:

RAI 19.01-5S1

- By design-specific fragility analysis. This analysis method uses the material properties and geometry specified by design documents to model the component capacity. It uses ~~in-structure response spectra (~~ISRS~~)~~ data for the seismic demand to calculate the response and safety factors using the separation of variables method.
- By using NuScale-specific response factors and the generic spectral acceleration capacity derived from industry sources (Reference 19.1-59) and ISRS data clipped according to the methodology in Reference 19.1-57.

The first modeling approach is used for PRA-critical components, such as active components located inside the reactor module.

For components located outside the reactor module (e.g., diesel generators), or components that do not show a substantial impact on the plant risk profile, the second method was used. This allows for the use of design-specific ISRS data and generic spectral acceleration capacities to determine the component fragilities.

Components sharing common type, building placement, and elevation within a building are similarly impacted by earthquakes. Because of this, these components are credited in groups for failure purposes during seismic events (Reference 19.1-21). Basic events sharing seismically relevant characteristics are grouped together and modeled as correlated failures. Seismic failures are assigned to groups named seismic correlation classes and are treated as failure events within the SMA.

Component location and elevation are additional criteria for seismic correlation class grouping. Location refers to the building or general area of the component rather than horizontal location within a building. For the purposes of correlation class grouping, all components with the same type in the same building (or general area) with the same elevation class are considered 100 percent correlated. (If at a different elevation, the components were treated as independent.) Table 19.1-37 lists the locations of seismic correlation class components in the SMA.

The component type is the principal grouping criterion for seismic correlation classes. Components that share similar structure, shape, and function are

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NRC Question No.: 19.01-14

10 CFR 52.47(a)(27) states that a DCA must contain an FSAR that includes a description of the design-specific PRA and its results. In SECY 93-087, the Commission approved use of the seismic margin approach (SMA) for DCAs.

During its audit of the PRA (ML17087A109), staff reviewed ER-P040-7026-R0, “Seismic Margin Assessment Notebook,” Appendix J. The applicant summarized the contractor-supplied fragility calculations for PRA-critical SSCs. The staff identified differences between the contractor-supplied HCLPF capacities for the RBC. The results are summarized below.

	Capacity (A_m)	Uncertainty (β_c)	HCLPF (g)
1	2.64	0.48	0.88
2	2.33	0.50	0.74

The HCLPF capacity provided in the FSAR corresponds to the higher HCLPF capacity (in row 1 above) supplied by the contractors. The failure modes considered are for the bridge and trolley seismic restraint assemblies. The controlling HCLPF capacity is determined to correspond to the seismic bridge restraint weldment.

A note is provided below Table J.10 in relation to the lower HCLPF capacity (in row 2 above). The note states that “The fragility for the reactor building crane is with respect to a bridge girder failure in shear. Upon further evaluation, catastrophic bridge girder failure will be preceded by bridge seismic restraint failure. Due to this realization, failure of the bridge seismic restraints are the controlling failure mode, rather than the bridge girder.” This information is also summarized in Section 19.1.5.1.1.3, Subsection “Reactor Building Crane.”

Based on the staff’s review of ER-P-040-7026-R0, Appendix J, during its audit of the PRA (ML17087A109), the staff needs additional information to understand the basis for eliminating bridge girder failure in shear as a controlling failure mode for the RBC fragility evaluation. Because the RBC is PRA-critical and controls the plant level HCLPF capacity, the staff requests that the applicant provide further justification for eliminating the bridge girder failure in shear in the RBC fragility evaluation.

NuScale Response:

Per discussion with the staff during a public meeting on October 3, 2017, the following information supplements the information provided in NuScale's original response.

NuScale has evaluated the potential for "bridge girder failure in shear" as well as "catastrophic bridge girder failure". Girder failure in shear implies that the girder exhibits inelastic deformation, but remains on top of the crane walls because of the span of the girders. Fragility for the bridge girders is calculated up to the yield strength of the girder materials. Any induced buckling or torsion short of complete girder shearing, while potentially rendering the crane inoperable, cannot cause the bridge structure to collapse as long as the seismic restraints hold. Catastrophic bridge girder failure is associated with a seismically-induced vertical jumping motion that is sufficient for the crane bridge to decouple from the rail and slide off the crane wall ledges; the seismic restraints are designed to resist this motion. Analysis shows that seismic restraint failure is the controlling seismic failure mode for the crane collapse and, thus, is reported as the controlling failure mode in FSAR Section 19.1.5.1.1.3 and FSAR Table 19.1-35.

FSAR Section 19.1.5.1.1.3 has been modified to reflect consideration of both failure modes. FSAR Figure 19.1-42 has been added to illustrate the bridge girder, crane wall ledges and seismic restraints.

Impact on DCA:

Figure 19.1-42 has been added to the FSAR and FSAR Section 19.1.5.1.1.3 has been been revised as described in the response above and as shown in the markup provided in this response.

component design information, ISRS is used in lieu of required response spectra for fragility calculations.

Seismic Structural Events

Structural events are modeled as basic events in the PRA model with median failure acceleration and uncertainty parameters. Structural events differ from component failures in that they do not correspond to any random event in the internal events PRA. In nearly all cases, the consequences of structural events are assumed to lead to both core damage and large release without opportunity for mitigation. This is a simplifying assumption for modeling catastrophic failure mechanisms.

The selection of structural failures to model is based on a qualitative assessment of the external mechanisms that can damage the reactor module. Structures selected for analysis meet one of the following criteria:

- Structures directly in contact with the reactor module: This applies to the module passive support skirt ring attached to the reactor pool floor, and the lateral support lug-corbel interface;
- Structures directly connected to the module interface: The reactor bay walls, pool wall and pool floor. The latter two are bounded in terms of fragility by the RXB outer wall failure;
- Structures located above the module, where collapse could lead to physical damage to the module. These include the Reactor Building crane (RBC) and the bioshield. The roof of the RXB and the pool wall fragility is bounded by the outer wall fragility analysis.

Figure 1.2-5 provides perspective on the locations of structural failures included in the SMA.

Reactor Building Crane

The RBC is located over the reactor pool and is suspended by girders. It runs the length of the reactor pool and is used primarily for raising and transporting NPMs to and from the refueling bay.

RAI 19.01-14S1

The crane is designed with seismic restraints. ~~Catastrophic~~As illustrated in Figure 19.1-42, ~~bridge girder failure is preceded by failures of these bridge seismic restraints through yielding of the restraint weldments~~bridge girder failure cannot lead to catastrophic collapse without failure of the bridge seismic restraints. ~~Therefore, failure~~Failure of the bridge seismic restraints is the controlling failure mode by comparison to yielding of the bridge girder itself. The bounding consequence of crane failure is a collapse of the crane structure, which is assumed to impact the top of the module, and lead to core damage and large release. This modeling simplification is ~~required~~conservative because the bioshield, CNV and RPV integrity are not credited following a crane collapse.

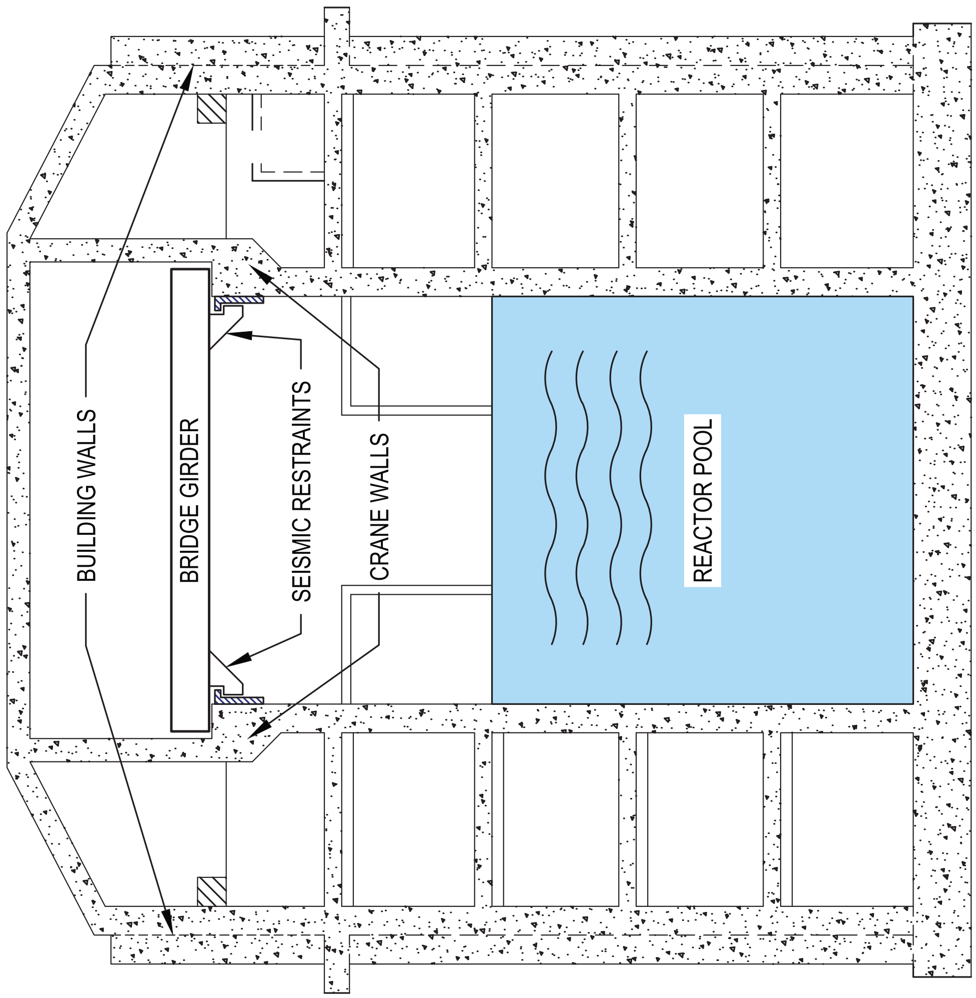

Figure 19.1-42: Simplified Reactor Building Section View

RAI 19.01-14S1

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NRC Question No.: 19.01-15

10 CFR 52.47(a)(27) states that a DCA must contain an FSAR that includes a description of the design-specific PRA and its results. SECY 93-087 approves an alternative approach to seismic PRA for the DCA and ISG 20 provide guidance on the methods acceptable to the staff to demonstrate acceptably low seismic risk for a DC.

Based on the staff's review of NuScale FSAR Tier 2, Section 19.1.5, the staff needs additional information to confirm the validity of the applicant's HCLPF capacities. The staff expectation at the DC stage is that the design of structures within the scope of DC is essentially complete. To evaluate the application, the staff requests that the applicant demonstrate that the seismic margin of 1.67 times the CSDRS is met for the seismic Category I structures against seismic-induced sliding and overturning.

NuScale Response:

Per discussion with the staff during a public meeting on October 3, 2017, the following information supplements the information provided in NuScale's original response.

FSAR Table 3.8.5-5 provides factors of safety (the ratio of resisting force to driving force) for reactor building (RXB) stability under safe shutdown earthquake (SSE) loading. The table illustrates that the safety factors for overturning are significant for all soil types and cracked and uncracked concrete conditions for the RXB. The RXB resistance against sliding has lower factors of safety, as shown in Table 3.8.5-5, so a nonlinear analysis was performed to evaluate the resulting sliding displacements for SSE loading, as discussed in FSAR Section 3.8.5.4.1.2. The results are provided in FSAR Table 3.8.5-12 for the RXB.

For the control building (CRB), both sliding and uplift displacements are shown in FSAR Table 3.8.5-13. The values indicate insignificant displacements compared to the building dimensions.

From the analysis results summarized in FSAR Tables 3.8.5-5, 3.8.5-12, and 3.8.5-13, it is judged that potential displacements associated with seismic loadings up to the plant-level high confidence of low probability of failure (HCLPF) would not challenge the structural integrity of the Category I structures. FSAR Table 19.1-40 has been modified to add an assumption that Category I structures are not vulnerable to sliding and overturning.

Impact on DCA:

FSAR Table 19.1-40 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 19.01-2, RAI 19.01-5, RAI 19.01-7, RAI 19.01-8, RAI 19.01-11, RAI 19.01-14, RAI 19.01-15S1, RAI 19.01-16

Table 19.1-40: Key Assumptions for the Seismic Margin Assessment

Assumption	Basis
Structures are screened out if they are not directly in contact with the reactor module and do not have the potential to collapse on top of it.	Engineering judgment
Systems and components are screened if they are not included in the internal events PRA models (full power and low power and shutdown).	Common engineering practice
Seismic sequences are mapped to those in the internal events PRA but augmented with seismically induced SSC initiating events and seismically induced SSC failures.	Common engineering practice and consistent with the ASME/ANS PRA Standard.
Intra-module component groups have 100 percent correlation provided all components share the same elevation class, general component type and same failure mode. Components not meeting these shared criteria are treated as independent.	Common engineering practice, consistent with the ASME/ANS PRA Standard, and bounding assumption.
Different component failure modes (for the same component or different components of the same type) are not modelled as correlated when the specific seismic failure mode is identified, i.e. "seismic failure to open". When the event is labeled as a functional failure, all failure modes are included and considered correlated.	Common engineering practice, consistent with the ASME/ANS PRA Standard, and bounding assumption.
Seismic component failures are not modelled for fail-safe signal logic, which includes sensors, transmitters, relays, equipment interface modules, safety function modules, actuation priority logic modules, hard-wired modules, scheduling and bypass modules, and scheduling and voting modules. As such, seismically-induced signal logic failures of the MPS are not considered credible.	Common engineering practice
Design-specific fragilities are used for PRA-significant seismic failure eventsfailures that contribute to the seismic margin, including valves located inside the reactor module and structural events.	Common engineering practice, consistent with the ASME/ANS PRA Standard, and engineering judgment.
For SSC that do not contribute significantly to the seismic safety margin, such as components credited in the PRA but not associated with a specific module, design-specific response factors combined with generic capacity values are used.	Engineering judgment and common engineering practice.
Fragility parameters acquired from generic sources, including capacity, randomness, and uncertainty values, are assumed valid and relevant to the NuScale design.	Common engineering practice
Systems are assumed to fail at the ground motion in which they have an 84 percent probability of failure. For ground motions with lower failure probabilities, the success logic is treated as a probability of 1.0.	Simplifying conservative assumption to avoid duplication of success logic in SAPHIRE.
Structural events (e.g., RXB wall), are postulated to directly lead to core damage and large release. The term "structural event" is used in lieu of "structural failure". One exception is a structural failure of the reactor module corbel bearing failure, which is postulated as a LOCA outside containment.	Bounding simplification and engineering judgment.
Control room failure is not included in the SMA because a control room collapse is bounded by the effects of a LOOP that occurs at lower ground motions with higher frequencies. A LOOP results in ECCS valve actuation; a control room collapse results in a signal loss and subsequent ECCS valve actuation.	Bounding assumption
The controlling failure mode of the RBC, which is designed with seismic restraints, is the yielding of the bridge seismic restraint weldments. The bounding consequence of crane failure during low power operations is a collapse of the crane structure on top of the module, leading to core damage and large release.	Bounding assumption

Table 19.1-40: Key Assumptions for the Seismic Margin Assessment (Continued)

Assumption	Basis
During low power and shutdown conditions, the state-specific risk to the module is during the transport phase before and after refueling, when the crane is bearing the load of the module. Other events involving the crane can be screened because the likelihood of the crane being over the module (and not bearing the load of the module) is bounded by the full-power assessment.	Engineering judgment
Failure of the bridge seismic restraints, rather than the bridge girders, is expected to be the controlling failure mode of the crane bridge. ~~Because the seismic restraints do not bear any additional weight from a loaded module, the effect on weldment failure is expected to be negligible.~~	~~Common engineering practice~~Engineering judgement
~~Cutsets~~In the MIN-MAX method, cutsets containing both seismic and random failures are screened if the product of all random failure probabilities is below 1E-2 because the HCLPF is defined as a 1percent failure probability on the mean fragility curve. Thus, it is reasonable to use this value as a screening criterion for the probability of non-seismic failures in the same cutset.	Common engineering practice and consistent with ISG-020.
In a cutset containing multiple seismic failures, the highest HCLPF value determines the cutset HCLPF.	Common engineering practice, application of the MIN-MAX method.
Because the dominant structural events are assumed to lead core damage and a large release, the plant-level core damage HCLPF is the same as the large release HCLPF.	Bounding assumption
Recovery, including the recovery of offsite power, is not credited in the SMA.	Bounding assumption
Extreme stress was considered for operator actions following a seismic event.	Engineering judgment
Fragilities developed via the separation of variables methodology are assumed to be representative of fragilities determined via qualification testing. The separation of variables methodology is based on the same SSC design information, specifications, and analysis as would be used to develop testing information during procurement.	Engineering judgment
The CFT and RFT do not contribute to the seismic margin because the core geometry remains coolable after the CNV top is removed, even if the CFT or RFT were to become damaged by an earthquake.	Engineering judgment
The MLA is modeled as part of the RBC structure and design safety margins preclude it from being the controlling seismic failure.	Engineering judgment
The control rod guide tubes are assumed to be the controlling seismically induced failure associated with the reactor internals. Therefore, seismically induced damage to reactor internals is not considered in the seismic margin.	Engineering judgment
Displacement-induced stresses that result in leakage in piping outside containment are assumed to be isolable by closure of the containment isolation valves.	Engineering judgment
Seismic Category I structures (i.e., the RXB and CRB) are not vulnerable to seismically-induced sliding or overturning (FSAR 3.8.5).	Engineering judgment

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

eRAI No.: 8899
Date of RAI Issue: 07/07/2017

NRC Question No.: 19.01-16

10 CFR 52.47(a)(27) states that a DCA must contain an FSAR that includes a description of the design-specific PRA and its results.
SECY 93-087 approves an alternative approach to seismic PRA for the DCA, and ISG 20 provides guidance on the methods acceptable to the staff to demonstrate acceptably low seismic risk for a DC. In accordance with ISG 20, the operating modes to be considered include at power, low power, and shutdown.

The staff reviewed FSAR Tier 2 Section 19.1.6.3, “Safety Insights from the External Events Probabilistic Risk Assessment for Low Power and Shutdown Operation” and noted that the Containment Vessel Flange Tool (CFT) and the Reactor Vessel Flange Tool (RFT) are not included in the SMA even though in FSAR Section 9.1.5 (1) the RFT and the Module Lifting Adapter are classified as Seismic Category I and (2) the CFT is classified as Seismic Category II. The RFT, the CFT, and the Module Lifting Adapter are not listed in FSAR Table 3.3-2. The staff requests that the applicant include the RFT, the Module Lifting Adapter, and the CFT in the SMA and in FSAR Table 3.3-2 or justify why these components are not listed in the table.

NuScale Response:

Per discussion with the staff during a public meeting on October 3, 2017, the third paragraph of NuScale's original response is modified to add the underlined text, as indicated below.

<u>Original response</u>

Fragilities for the CFT and RFT do not appear in the seismic margin assessment (SMA) because of the limited consequences of their seismic failure. After the module is placed in the CFT and the CNV flange bolts are de-tensioned, the inventory of the ultimate heat sink (UHS) is available for cooling the core through the open emergency core cooling system (ECCS) valves. Therefore, core cooling is assured by natural circulation even if the CFT were to fail seismically. When the module is in the RFT, the RPV head is removed, core cooling is provided by the UHS, irrespective of a seismically-induced RFT failure. As such, the CFT and RFT do not contribute to the seismic margin.

Supplemental response

Fragilities for the CFT and RFT do not appear in the seismic margin assessment (SMA) because of the limited consequences of their seismic failure. As described in FSAR Section 19.1.6.1, the module remains attached to and supported by the reactor building crane, with the CNV flooded up to the pressurizer baffle plate level, until it is placed in the CFT. After the module is placed in the CFT and the CNV flange bolts are de-tensioned, the inventory of the ultimate heat sink (UHS) is available for cooling the core through the open emergency core cooling system (ECCS) valves. Therefore, core cooling is assured by natural circulation even if the CFT were to fail seismically. When the module is in the RFT, the RPV head is removed, core cooling is provided by the UHS, irrespective of a seismically-induced RFT failure. As such, the CFT and RFT do not contribute to the seismic margin.

Impact on DCA:

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