RAIO-0118-58238

January 17, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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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 Question from NRC eRAI No. 8899:

- 19.01-8

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 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

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

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

NRC Question No.: 19.01-8

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.

The staff reviewed FSAR Section 19.1.5.1.1.3, Subsection "NuScale Power Module Supports," and could not find information it needs to verify the adequacy of the PRA-based SMA. The FSAR states, "Corbel bearing failure is expected to crush the corbel concrete in compression, causing minor axial rotation of the module resulting in a displacement assumed to be no more than 1 inch for the CNV. Because the flexibility in the piping is in the section between the isolation valve and the wall penetration, there is no credible mechanism for the bearing failure displacement to cause piping on top of the CNV to shear off the vessel." The staff requests that the applicant describe the analysis that provides the technical basis for concluding that minor axial rotation of the module results in a displacement assumed to be no more than 1 inch. The applicant should also clarify how the flexibility in the piping prevents the piping on top of the CNV from shearing off of the vessel. The staff additionally requests that the applicant identify the entries in Table 19.1-38 that correspond to the containment isolation valves (CIVs) for this sequence.

NuScale Response:

NuScale provided its original response to RAI 8899, Question 19.01-8, in letter RAIO-0917-55781, dated September 01, 2017. Per discussion with the staff during a public meeting on October 03, 2017, NuScale is revising its original response regarding the evaluation of the Nuclear Power Module (NPM) supports. Accordingly, the following response replaces the original response in its entirety:

The NPM support fragilities have been revised for consistency with the module lug support system, which is described in FSAR Section 3B.2.7.4 and illustrated in FSAR Figure 3B-51, and the NPM base support, which is described in FSAR Section 3B.2.7.3 and illustrated in FSAR Figure 3B-48. Corbel failure modes are not applicable because corbels are not present in the

design.

The controlling fragility for the NPM supports (module lug support system and NPM base support) is the shear failure of the module lug support system shear lugs, which are embedded within the bay wall and pool wall structures.

FSAR Section 19.1.5.1 has been revised to remove discussion related to corbels and their associated failure modes. FSAR Tables 19.1-35, 19.1-37, 19.1-38, and 19.1-40 have been updated to reflect the current analysis results.

Impact on DCA:

FSAR Section 19.1.5.1 and FSAR Tables 19.1-35, 19.1-37, 19.1-38, and 19.1-40 have been revised as described in the response above and as shown in the markup provided in this response.

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 relative~~Consistent 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

~~Component~~Structure, system, and component fragility is referenced to the peak ground acceleration ~~defining~~of the ~~uniform hazard response spectra for a site~~CSDRS, 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 19.01-8S1, 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 SSC 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 SSC that contribute to the seismic margin. Contributing SSC 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 19.01-8S1

The controlling failure mode of these structural events and their direct consequences are shown in Table 19.1-36. 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:

RAI 19.01-8S1

- Structures directly in contact with the reactor module: This applies to the ~~module passive support skirt~~NPM base support ~~ring attached to the reactor pool floor, and the lateral support lug-corbel interface~~and module lug support system;

RAI 19.01-4

- Structures directly connected to the module interface: The reactor bay walls, pool wall, and basemat~~pool floor. The latter two are bounded in terms of fragility by the RXB outer wall failure~~; or

RAI 19.01-4

- 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 19.01-4, 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.

Reactor Building Wall

RAI 19.01-4

The fragility of the RXB as a whole is modeled by separate fragility analyses of each of the wall types, as well as the RXB roof, and basemat:~~a fragility analysis of the structural location experiencing maximal loads (both seismic and normal). Failure is assumed to lead to building collapse, core damage and large release.~~

RAI 19.01-4

~~The RXB is modeled using the controlling failure mode of out-of-plane shear cracking at the base of the outer East-West wall. The outer walls have the highest elevation and fewest lateral supports.~~

RAI 19.01-4

- the four exterior RXB walls
- the four RXB pool walls
- the RXB crane support structure
- the pool bay walls
- the RXB roof
- the basemat

RAI 19.01-4

The locations experiencing maximum loading (combined seismic and non-seismic) for each of the above groups of structures, were evaluated. Failure is assumed to lead to building collapse, core damage, and large release. The controlling failure mode is determined to be out-of-plane shear cracking at the base of the exterior east-west walls.

RAI 19.01-8S1

NuScale Power Module Supports

The two supporting interfaces between the CNV and the reactor pool are:

- ~~The support lugs and wall corbels;~~the module lug support system
- ~~The support skirt and pedestal.~~the NPM base support

RAI 19.01-8S1

The module lug support system is comprised of:

- steel bumpers, which are welded to the wall liner plate
- vertical shear lugs, embedded in the bay and pool walls
- steel liner plates, which cover the pool and bay walls at the lug elevation
- through-bolts, which anchor the liner plate on either side of the bay walls

RAI 19.01-8S1

The NPM base support is comprised of:

- the passive support skirt ring
- a ring of capture bolts embedded in the support skirt
- an embedded steel plate set under the CNV skirt and passive support skirt ring
- a square array of anchors extending from the embedded plate into the concrete foundation

RAI 19.01-8S1

The controlling failure mode for the module supports (for both the lug and floor locations) is evaluated as the shear failure of multiple concrete-embedded shear lugs.

RAI 19.01-8S1

~~The support lug and corbel analysis revealed two controlling failure modes with different consequences: bearing failure of the lugs on the corbel concrete and corbel shear failure.~~

RAI 19.01-8S1

~~Corbel bearing failure is expected to crush the corbel concrete in compression, causing minor axial rotation of the module resulting in a displacement assumed to be no more than 1 inch for the CNV. Because the flexibility in the piping is in the section between the isolation valve and the wall penetration, there is no credible mechanism for the bearing failure displacement to cause piping on top of the CNV to shear off of the vessel. Therefore, the bounding consequences of such a displacement would be stress concentrations on the piping attached to the top of the CNV, resulting in a potential leak of primary coolant outside containment outside the CIVs. This scenario is therefore modeled as a pipe break outside containment with containment isolation available.~~

RAI 19.01-8S1

~~Corbel shear failure is expected to occur at a higher loading than bearing failure. Shear failure on any of the three corbels is the controlling failure mode for the reactor module supports. Support failure is assumed to directly cause core damage and a large release because the integrity of the RPV and CNV cannot be ensured if the module becomes detached from its supports.~~

RAI 19.01-8S1

~~The controlling failure mode for the passive support ring is expected to be horizontal shear force, which is part of the foundation and located inside the outer vessel support skirt ring. Its calculated scale factor of 3.46 is higher than that of the corbel shear failure. It is therefore screened out as a non-controlling failure mode for the reactor supports.~~

Reactor Bay Wall

The reactor module is surrounded by bay walls on two sides and the reactor pool wall on a third. The fourth side is open to the middle of the reactor pool. The bay wall failure is expected to be controlling compared to the reactor pool wall because it is supported only on one end. As for other structural failures, failure of the reactor bay wall is assumed to lead to core damage and a large release.

Bioshield

Each reactor module is covered by a removable bioshield that rests over the module during normal operation. The bioshield consists of a concrete slab attached on three sides by anchor bolts to the bay walls and pool walls.

During refueling, the bioshield of the refueled module is placed on top of an adjacent module. Any operating module, therefore, may have two bioshields stacked over it. A separate fragility calculation is performed for two stacked bioshields and is included in the SMA.

Because the bioshield is a simple slab structure, four potential bioshield failure modes were identified:

- Vertical bioshield failure;
- Horizontal shear flexure;
- Pool wall anchor bolt shear;
- Bay wall anchor bolt shear.

Bioshield failure is expected to cause the entire horizontal slab section to collapse on top of the reactor module, causing core damage and a large release.

Vertical bioshield failure has been screened from analysis because of the bounded consequences of failure. The controlling failure mode would involve detachment from its lower supports against the bay wall, its upper connection to the horizontal bioshield slab, and then sufficient flexing of the bay walls to allow the vertical section to separate from the rest of the bioshield and twist inwards to strike the CNV. Because bay wall twisting and shear cracking failure is evaluated by a separate fragility calculation, this fragility is screened from the analysis.

and the diesels fail to restore power, the ECCS valves open after the DC power holding the valves closed, is removed, and the DHRS or the reactor safety valves (RSVs) depressurize the RPV to the point where the inadvertent actuation block (IAB) allows the ECCS valves to open.

RAI 19.01-8S1

Seismically-induced SGTF is then modeled with a median failure capacity of 2.9g (failure of the support leads to tube failure). The logic is mapped similarly to a randomly occurring SGTF. Other induced failures include LOCAs inside containment (spurious opening of RSVs or ECCS valves), ~~LOCAs~~breaks outside containment (~~corbel bearing failure or~~ CVCS regenerative heat exchanger failure) and (most severely) structural events.

~~As noted in Section 19.1.5.1.1.2, the~~The seismic hazard for the NuScale design has been partitioned into 14 seismic initiating event trees defining the SMA~~, representing different ground motion accelerations. The underlying logic for each tree is identical~~. The underlying logic for each tree is identical. However, each tree represents a different ground motion acceleration. ~~Each seismic ground motion initiator is a SAPHIRE initiating event with a frequency set to unity in order to evaluate the conditional core damage or large release probability associated with that ground motion.~~

Each event tree is assigned a ground motion acceleration increasing monotonically from 0.005g to 4.0g. The seismic initiator event tree provided as Figure 19.1-16 corresponds to a range of peak ground accelerations from 0.005g to 0.1g. The thirteen remaining event trees represent ground motion ranges spaced accordingly up to 4.0g (0.1g to 0.2g, 0.2g to 0.4g,..., 2.0 to 2.5g,...,3.0g to 4.0g). Component failure probabilities are then evaluated at the mid-point of each range (0.0525g for a range of 0.005g to 0.1g, for instance). This methodology supports site-specific estimates of seismic hazard occurrence frequency. Each ground motion initiator is a SAPHIRE initiating event with a frequency set to unity. This allows for an evaluation of conditional core damage or large release probability at each ground motion.

Seismically-induced event trees are initiated ~~based on~~by the failure of a single component or structural ~~event, as described above. Sequences containing these failure events transfer to other event trees (from the seismic initiating event tree) representing plant response to losses of offsite power (Figure 19.1-20), SGTFs (Figure 19.1-19), loss of coolant accidents inside containment (Figure 19.1-17), and pipe breaks outside containment (Figure 19.1-16).~~event. Sequences containing these failure events transfer from Figure 19.1-16 to other seismic event trees that represent plant response to breaks outside containment (Figure 19.1-17), LOCAs inside containment (Figure 19.1-18), SGTFs (Figure 19.1-19), and losses of offsite power (Figure 19.1-20). Figure 19.1-17 and Figure 19.1-19 include a transfer to a loss of DC power event tree (Figure 19.1-20a) to reflect battery depletion at 24 hours. These trees are modified from existing internal events PRA event trees to remove credit for the availability of AC power or for offsite power recovery.

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 19.01-17

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 19.01-8S1

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 19.1-36 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 19.1-36, 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 sequences involving the corbel bearing failure. This leaves corbel shear 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

This section provides brief descriptions of the significant contributors to risk as determined by the SMA.

Structural events are by far the leading contributor to the seismic margin. The bounding structural event is weldment failure on the crane bridge seismic restraints, which is modeled to lead directly to crane collapse, core damage and large release.

RAI 19.01-8S1

A single SMA sequence ~~(sequence SEISMIC-ET-HCLPF: 6-3)~~ contains all structural events and represents 99.8 percent of the large release conditional failure probability after a HCLPF-level earthquake. In accordance with the MIN-MAX method, the lowest HCLPF value between cutsets in the same sequence is controlling. This is why only the Reactor Building crane event HCLPF ~~of 0.88g~~ shows up at the sequence level.

Risk Significance

Potentially risk significant structures, components and operator actions are discussed below.

Significant Structural Failures

RAI 19.01-8S1

Table 19.1-36 lists nine individual structural failure modes for which seismic fragilities are generated. Of these, eight represent ~~single structures that, if they were to fail during a seismic event, are~~structural failure modes assumed to lead directly to core damage and a large release. The fault tree logic for these structures is represented by an "OR" gate with all eight inputs, with any one failure leading to core damage and large release. ~~The accident sequence logic is represented by the first heading of the seismic event tree (Figure 19.1-16).~~ The most risk significant of these structural failures is for yielding of the Reactor Building crane bridge seismic restraint weldments.~~reactor crane bridge seismic restraint weldment yielding, as it has the lowest HCLPF per Table 19.1-36.~~

RAI 19.01-8S1

~~A ninth structural failure mode, corbel bearing failure, can result in a pipe break outside containment. However, additional structural or random failures must occur in the form of failing to isolate containment before core damage would result. Successful isolation enables the ability of the DHRS and the ECCS to provide adequate core cooling. Therefore, the corbel bearing failure is not considered as risk significant as the other eight structural failures.~~

Significant Component Failure Modes

The NuScale unique passive safety features limits the risk associated with failure of active components (such as pumps, compressors and switches) to perform during or after a seismic event. In addition, mitigating systems are largely fail safe, 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 03.02.01-4, RAI 19.01-4, RAI 19.01-8S1

Table 19.1-35: Structural Fragility Parameters and Results

Structural Event	A_m (g)	β_r	β_u	HCLPF (g)	Controlling Failure Mode	Assumed consequence
Reactor Building Crane	2.64	0.28	0.39	0.88	Bridge seismic restraint weldment yielding	Core damage / Large Release
Reactor Building Exterior Walls	~~2.27~~1.92	~~0.20~~0.12	~~0.32~~0.33	~~0.96~~0.92	Out-of-plane shear~~ cracking at base of outer E-W wall~~	Core damage / Large Release
Reactor Module Supports~~ Corbel bearing failure~~	~~1.94~~1.98	0.~~21~~12	~~0.24~~0.35	~~0.68~~0.92	~~Reactor module support lug bearing compressive failure on corbel concrete~~Shear failure of multiple shear lugs	~~Isolable pipe break outside containment~~Core damage/ Large Release
~~Reactor Module Supports - Corbel shear~~	~~2.67~~	~~0.21~~	~~0.38~~	~~1.01~~	~~Corbel concrete diagonal shear failure~~	~~Core damage / Large Release~~
~~Reactor Bay Wall~~	~~2.47~~	~~0.19~~	~~0.42~~	~~1.13~~	~~In-plane gross shear failure~~	~~Core damage / Large Release~~
Bio Shield - horizontal shear flexure - normal operation	11.62	0.28	0.37	~~3.98~~3.99	Horizontal shield slab bending failure	Core damage / Large Release
Bio shield - pool wall bolt failure - normal operation	5.37	0.28	0.35	~~1.90~~1.91	Shear Failure of pool wall Anchor Bolts	Core damage / Large Release
Bio shield - horizontal shear flexure - double stacked for refueling of adj. model	4.05	0.28	0.41	1.30	Bending failure of both stacked shield slabs	Core damage / Large Release when configuration present
Bio shield - pool wall bolt failure - double stacked for refueling of adj. model	3.05	0.28	0.35	1.08	Shear Failure of pool wall Anchor Bolts	Core damage / Large Release when configuration present
Pool Walls	2.31	0.21	0.33	0.95	Out-of-plane shear	Core damage / Large Release
Crane Support Walls	2.61	0.12	0.34	1.23	Out-of-plane shear	Core damage / Large Release
Bay Walls	2.65	0.12	0.31	1.31	In-plane flexure	Core damage / Large Release
Roof	2.22	0.12	0.26	1.20	In-plane shear	Core damage / Large Release
Basemat	3.57	0.27	0.31	1.38	Out-of-plane shear	Core damage / Large Release

A_m = median seismic capacity; βu = uncertainty in the median seismic capacity; βr = randomness of the fragility evaluation; HCLPF = High-Confidence (95%) of a Low Probability (5%) of Failure, Reference 19.1-57~~= A_m exp [- 1.65 (βr + βu)]~~

RAI 03.02.01-4, RAI 19.01-8S1

Table 19.1-37: Seismic Margin Analysis Component Types

Comp ID	Component Description
ACV	AIR OPERATED CONTROL VALVE
AOV	AIR OPERATED VALVE
BAT	BATTERY
BCH	BATTERY CHARGER
BIOBN	BIO SHIELD BAY WALL ANCHOR BOLT (NORMAL OPERATION)
BIOBR	BIO SHIELD BAY WALL ANCHOR BOLT REFUELING OPERATIONS)
BION	HORIZONTAL BIO SHIELD SLAB (NORMAL OPERATION)
BIOPN	BIO SHIELD POOL WALL ANCHOR BOLT (NORMAL OPERATION)
BIOPR	BIO SHIELD POOL WALL ANCHOR BOLT (REFUELING)
BIOR	HORIZONTAL BIO SHIELD SLAB (REFUELING)
BYW	REACTOR POOL BAY WALL
CBH	HIGH VOLTAGE CIRCUIT BREAKER
CBL	LOW VOLTAGE CIRCUIT BREAKER
CBM	MEDIUM VOLTAGE CIRCUIT BREAKER
CKV	CHECK VALVE
~~CORB~~	~~REACTOR MODULE CORBEL~~
CRDGT	CONTROL ROD GUIDE TUBE
CRN	REACTOR BUILDING CRANE
CTG	COMBUSTION TURBINE GENERATOR
DGN	DIESEL GENERATOR
EBA	AC BUS
EBD	DC BUS
HOV	HYDRAULICALLY OPERATED VALVE
HTX	HEAT EXCHANGER
MCC	MOTOR CONTROL CENTER
MDP	MOTOR DRIVEN PUMP
MOV	MOTOR OPERATED VALVE
MSW	MANUAL SWITCH
RBW	REACTOR BUILDING WALL
RRV2	ALL ECCS REACTOR RECIRCULATION VALVES
RSV	REACTOR SAFETY VALVE
RTB	REACTOR TRIP SYSTEM CIRCUIT BREAKER
RVV3	ALL ECCS REACTOR VENT VALVES
SGT	STEAM GENERATOR TUBE
SOV	SOLENOID OPERATED VALVE
SUPP	MODULE SUPPORT
TFM	TRANSFORMER

RAI 19.01-3, RAI 19.01-4, RAI 19.01-8S1, RAI 19.01-9, RAI 19.01-17

Table 19.1-38: Seismic Correlation Class Information

Seismic Transfer Event	Component ID	Elevation (ft)	Location	NuScale Component	Failure Mode Description	A_m (g)	β_r (g)	β_u (g)	HCLPF (g)[1]	Contributes to seismic margin?[2]	Fragility Method[3]
Seismically Induced Initiating Events											
SUPP-75-RXB-SHR-SEIS	SUPP	75	RXB	RXM Supports	Shear Failure of Multiple Shear Lugs	1.98	0.12	0.35	0.92	Yes	DS
HTX---50--RXB---HXF-SEIS[4]	HTX	50	RXB	CVCS Heat Exchanger	Heat Exchanger Failure	6.81	0.32	0.51	1.74	No	Generic
RRV2--50--RXM---FTC-SEIS	RRV2	50	RXM	All ECCS Reactor Recirculation Valves	Fails to Close Fails to Remain Closed Spuriously Open	3.32	0.24	0.32	1.32	No	DS
RSV---75--RXM---FTC-SEIS[4]	RSV	75	RXM	All Reactor Safety Valves	Fails to Close Fails to Remain Closed Fails to Reclose Spuriously Open	3.37	0.24	0.32	1.34	No	DS
RVV3--75--RXM---FTC-SEIS	RVV3	75	RXM	All ECCS Reactor Vent Valves	Fails to Close Fails to Remain Closed Spuriously Open	2.38	0.28	0.5	0.66	No	DS
SGT---50--RXM---BRK-SEIS[4]	SGT	50	RXM	Steam Generators	Tube/Support Failure	2.53	0.28	0.36	0.88	No	DS
TFM---100-SITE--CIF-SEIS	TFM	100	SITE	Offsite Power Transformer	Ceramic Insulator Failure	0.3	0.29	0.47	0.09	No	Generic
Structural Failure Events											
BIOBN-125-RXB---BSF-SEIS	BIOBN	125	RXB	Bioshield Bay Wall Anchor Bolts	Bolt Shear Failure - Normal Operation	4.89	0.28	0.35	1.73	Yes	DS
BIOBR-125-RXB---BSF-SEIS	BIOBR	125	RXB	Bioshield Bay Wall Anchor Bolts	Bolt Shear Failure - Refueling Adjacent Module	2.73	0.28	0.35	0.97	Yes	DS
BION--125-RXB---OPB-SEIS	BION	125	RXB	Horizontal Bioshield	Out of Plane Bending - Normal Operation	11.62	0.28	0.37	3.99	Yes	DS
BIOPN-125-RXB---BTF-SEIS	BIOPN	125	RXB	Bioshield Pool Wall Anchor Bolts	Bolt Tension Failure - Normal Operation	5.37	0.28	0.35	1.91	Yes	DS

Table 19.1-38: Seismic Correlation Class Information (Continued)

Seismic Transfer Event	Component ID	Elevation (ft)	Location	NuScale Component	Failure Mode Description	A_m (g)	β_r (g)	β_u (g)	HCLPF (g)[1]	Contributes to seismic margin?[2]	Fragility Method[3]
BIOPR-125-RXB---BTF-SEIS	BIOPR	125	RXB	Bioshield Pool Wall Anchor Bolts	Bolt Tension Failure - Refueling Adjacent Module	3.05	0.28	0.35	1.08	Yes	DS
BIOR--125-RXB---OPB-SEIS	BIOR	125	RXB	Horizontal Bioshield	Out of Plane Bending - Refueling Adjacent Module	4.05	0.28	0.41	1.3	Yes	DS
BYW-------RXB---FLX-SEIS	BYW	NA	RXB	RXM Bay Wall	In-Plane Flexure Failure	2.65	0.12	0.31	1.31	Yes	DS
CRN---145-RXB---RWF-SEIS	CRN	145	RXB	Reactor Building Crane	Seismic Restraint Weldment Failure	2.64	0.28	0.39	0.88	Yes	DS
RBW-------RXB---OPS-SEIS	RBW	NA	RXB	Exterior Reactor Building Wall	Out of Plane Shear Failure	1.92	0.12	0.33	0.92	Yes	DS
Component Failure Events											
ACV---100-CHILL-FCR-SEIS	ACV	100	CHILL	DWS Recirc Control Valve	Fails to Control	9	0.32	0.52	2.26	No	Generic
ACV---100-RXB---FTO-SEIS	ACV	100	RXB	CFDS Flow Control Valve	Fails to Open	4.41	0.32	0.52	1.11	No	Generic
ACV---100-RXM---FTC-SEIS	ACV	100	RXM	FWS Regulating Valve	Fails to Close	22.13	0.27	0.37	7.72	No	DS
ACV---100-RXM---FTO-SEIS	ACV	100	RXM	CVCS Control Valve	Fails to Open	0.57	0.32	0.52	0.14	No	Generic
AOV---100-CHILL-FTO-SEIS	AOV	100	CHILL	DWS Pump Isolation Valve	Fails to Open	9	0.32	0.52	2.26	No	Generic
AOV---100-RXB---FTC-SEIS	AOV	100	RXB	CFDS Drain Valve	Fails to Close	4.41	0.32	0.52	1.11	No	Generic
AOV---100-RXB---FTO-SEIS	AOV	100	RXB	CVCS Module Heatup Isolation Valve, CFDS Flooding Valve	Fails to Open	0.57	0.32	0.52	0.14	No	Generic
AOV---100-RXM---FTC-SEIS	AOV	100	RXM	MSS Secondary Isolation Valve	Fails to Close	22.13	0.27	0.37	7.72	No	DS
AOV---100-RXM---FTO-SEIS	AOV	100	RXM	CFDS Isolation Valve	Fails to Open	0.57	0.32	0.52	0.14	No	Generic
AOV---50--RXB---FTO-SEIS	AOV	50	RXB	CVCS DWS Supply Isolation Valve	Fails to Open	7.74	0.32	0.52	1.94	No	Generic

Table 19.1-38: Seismic Correlation Class Information (Continued)

Seismic Transfer Event	Component ID	Elevation (ft)	Location	NuScale Component	Failure Mode Description	A_m (g)	β_r (g)	β_u (g)	HCLPF (g)[1]	Contributes to seismic margin?[2]	Fragility Method[3]
RSV---75--RXM---FTC-SEIS[4]	RSV	75	RXM	All Reactor Safety Valves	Fails to Close	3.37	0.24	0.32	1.34	No	DS
					Fails to Remain Closed						
					Fails to Reclose						
					Spuriously Open						
RSV---75--RXM---FTO-SEIS	RSV	75	RXM	All Reactor Safety Valves	Fails to Open	3.37	0.24	0.32	1.34	Yes	DS
RTB---75--RXB---FOP-SEIS	RTB	75	RXB	Reactor Trip Circuit Breaker	Fails to Operate	3.69	0.24	0.39	1.31	No	Generic
SGT---50--RXM---BRK-SEIS[4]	SGT	50	RXM	Steam Generators	Tube/Support Failure	2.53	0.28	0.36	0.88	No	DS
SOV---50--RXM---FTO-SEIS	SOV	50	RXM	ECCS Reactor Recirculation Valve Trip Valve Solenoids	Fails to Open	3.32	0.24	0.41	1.14	No	DS
SOV---75--RXM---FTO-SEIS	SOV	75	RXM	ECCS Reactor Vent Valve Trip Valve Solenoids	Fails to Open	3.23	0.28	0.53	0.85	No	DS
TFM---100-HVSWG-FOP-SEIS	TFM	100	HVSWG	13KV High Voltage Main Power Transformer	Fails to Operate	2.1	0.24	0.39	0.75	No	Generic
TFM---100-LVPDC-FOP-SEIS	TFM	100	LVPDC	Low Voltage Transformer	Fails to Operate	2.1	0.24	0.39	0.75	No	Generic
TFM---100-MVSWG-FOP-SEIS	TFM	100	MVSWG	13KV/4KV Auxiliary Transformer	Fails to Operate	2.1	0.24	0.39	0.75	No	Generic

Notes:

1. All HCLPF values are determined via 5% failure probability on the 95% probability of exceedance fragility curve, Reference 19.1-57.
2. Contribution to the seismic margin is determined via a systematic methodology considering the MIN-MAX HCLPF determination and random CCDP product > 1% criterion described in Table 19.1-41.
3. The methods used to evaluate component fragilities are identified as either "DS" (design-specific) or "Generic". Design-specific fragilities include an evaluation of both the equipment capacity and demand relative to a specific structure or piece of equipment. Generic fragilities constitute fragilities determined via a library/database search of similar equipment types. Such generic fragilities are augmented with ISRS information to include ground motion amplification specific to the NPM and the NuScale reactor building. All component failure modes identified as critical have design-specific fragilities.
4. Three seismically-induced component failure modes are also identified as seismically induced initiating events (HTX---50--RXB---HXF-SEIS, RSV---75--RXM---FTC-SEIS, and SGT---50--RXM---BRK-SEIS). In accident sequences initiated by failure of this equipment, the equipment is not available for mitigation.

RAI 19.01-2, RAI 19.01-5, RAI 19.01-7, RAI 19.01-8, RAI 19.01-8S1, 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 events~~failures 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
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