RAIO-1018-62292

October 25, 2018

Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 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
3. NuScale Power, LLC Supplemental Response to NRC "Request for Additional Information No. 83 (eRAI No. 8899)," dated May 18, 2018

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:

- 19.01-2

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 Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

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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.: 19.01-2

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 lieu of a seismic PRA. 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.

In FSAR Tier 2, Section 19.1.5, the staff identified the use of the terms “PRA-critical” and “Non-critical.” The staff requests that the applicant provide a definition of the terms “PRA-critical” and “non-critical” that are consistent with their usage as listed below and applicability to the PRA-based SMA.

The terms are used in the following sections.

· “Non-critical”

· Section 19.1.5.1.1.3, Page 19.1-54

· “PRA-critical”

· Section 19.1.5.1.1.3, Page 19.1-58

· Section 19.1.5.1.1.3, Page 19.1-59

· Section 19.1.5.1.2, Page 19.1-63

· Section 19.1.5.1.2, Page 19.1-64

Additionally in Section 19.1.5.1.1.3, the 2nd paragraph describes the methodologies used to determine the seismic capacity and demand for the SMA. The staff requests that the applicant clarify if the 1st sentence in that paragraph is referring to PRA-critical structures and components. The applicant should also clarify if non-critical components are modeled in the SMA and whether there are any non-critical structures.

NuScale Response:

NuScale is revising its response to RAI 8899 (Question 19.01-2) originally provided in letter RAIO-0917-55781 dated September 01, 2017 and replaced in its entirely by letter RAIO-0518-60071 dated May 18, 2018. This revised response is provided as a result of discussions with the NRC during a public call on August 28, 2018. Consistent with those discussions, this supplemental response

1. provides supplemental wording to FSAR Section 19.1.5.1 to clarify component and structure boundaries for fragility evaluation in the seismic margin assessment (SMA),
2. provides supplemental wording to FSAR Section 19.1.5.1 to clarify consideration of all cutsets when evaluating SMA risk insights,
3. corrects the erroneous fragility parameter values for seismic correlation class “MOV---100-RXM---FTC-SEIS” in Table 19.1-38, and
4. provides editorial changes.

Impact on DCA:

FSAR Section 19.1.5.1 and Table 19.1-38 have been revised as described in the response above and as shown in the markup provided in this response.

The controlling failure mode of the structural events and their direct consequences are shown in Table 19.1-35. 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). Information for component fragilities is provided in Table 19.1-38.

Seismic Structural Events

RAI 19.01-2S2

~~Structural events are modeled as basic events in the PRA model with median failure acceleration and uncertainty parameters.~~Fragilities for structural failures are modeled as basic events in the SMA model with median failure accelerations and uncertainty parameters. For each structural fragility, boundaries are defined such that all relevant seismically-induced failure mechanisms are accounted for (e.g., failures to supporting sections, intersecting structures, nearby structures). Seismically-induced structural failures are then assumed to lead directly to core damage and large release without opportunity for mitigation. This is a simplifying assumption for modeling catastrophic failure mechanisms. Structural events differ from component failures in that they do not correspond to a random event in the internal events PRA. In 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 NPM. Structures selected for analysis meet one of the following criteria:

RAI 19.01-8S1

- Structures directly in contact with the NPM: This applies to the NPM base support and module lug support system;

RAI 19.01-4

- Structures directly connected to the module interface: The reactor bay walls, pool wall, and basemat; 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.

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

is evaluated by a separate fragility calculation, this fragility is screened from the analysis.

Due to the geometric configuration of the anchor bolts, different failure modes are controlling for each direction:

- East-West: Shear Failure of both bay wall and pool wall anchor bolts;
- North-South: Shear Failure of bay wall anchor bolts, tension failure of pool wall anchor bolts;
- Vertical: Tension failure of bay wall anchor bolts, shear failure of pool wall anchor bolts.

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.

Components

RAI 19.01-2S2

~~For the SMA, seismic~~Similar to fragilities developed for structural failures, fragilities for component failures are modeled as basic events with median failure accelerations and uncertainty parameters. For each component fragility, component boundaries are defined such that all relevant seismically-induced failure mechanisms are accounted for (e.g., anchorage failure, structural collapse affecting component function). Seismically-induced component failures are then mapped to existing random component failure modes from the internal events PRA. 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 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 derived from clipped ISRS, the methodology outlined in EPRI 103959, and generic spectral acceleration capacities developed from EPRI 3002000507 (Reference 19.1-59) and NUREG/CR-2680, NUREG/CR-3558, NUREG/CR-4659, and NUREG/CR-7040 (Reference 19.1-18, Reference 19.1-19, Reference 19.1-20 and Reference 19.1-25, respectively).

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

For components located outside the NPM (e.g., diesel generators), or components that, if failed, would not directly affect safe shutdown, 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.

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 19.01-17,

Seismic risk is evaluated~~quantified~~ in terms of a plant-level HCLPF g-value and a review of SMA accident sequence cutsets for risk insights. ~~SMAs are required to show that the plant level HCLPF is greater than 1.67 times the SSE, which equates to a 0.84g peak ground acceleration for NuScale.~~

RAI 19.01-2S1, RAI 19.01-2S2

The plant-level HCLPF is determined by examining the cutset results from all fourteen seismic event trees. All cutsets are reviewed to screen those that are not relevant to the determination of the plant-level HCLPF. Per the MIN-MAX screening assumption addressed in Table 19.1-40, cutsets are screened out if the combined probability of random failures is less than one percent. This is appropriate because the conditional probability of failure corresponding to the HCLPF (i.e., given an earthquake ground motion equal to the plant-level HCLPF) is required to be greater than or equal to one percent (using the mean fragility curve). Therefore, even if all seismically induced failure probabilities of a particular cutset were 100 percent, the probability of core damage from non-seismic random failures must be greater than or equal to one percent for the cutset to be a relevant contributor to the HCLPF calculation. If the combined random failure probability of the cutset is below one percent, the cutset would not be a relevant contributor to the HCLPF calculation. The MIN-MAX method is ~~applied to each~~then applied to the remaining cutsets to determine the SSC with the limiting HCLPF for each cutset. ~~Each cutset retained in the HCLPF evaluation corresponds to an SSC that~~The limiting SSC identified for each cutset contributes to the seismic margin. Of all the seismic margin contributors, the SSC with the smallest HCLPF value provides the plant-level HCLPF. To demonstrate acceptably low seismic risk at the design certification stage, as indicated by DC/COL-ISG-020, the resultant plant-level HCLPF must be greater than or equal to 0.84g, which is the plant-level HCLPF requirement of 1.67 times the SSE.

RAI 19.01-2S2

All cutsets associated with the corresponding peak ground acceleration HCLPF g-value are reviewed for seismic risk insights. That is, cutsets are not screened from the review process so that all cutsets are considered for potential risk insights.

RAI 19.01-2S2

Plant Level HCLPF

RAI 19.01-8S1, RAI 19.01-2S2

~~The resulting HCLPF acceleration~~Implementation of the screening process described above results in a plant-level HCLPF for the NuScale design ~~is~~of 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. 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

RAI 19.01-2S2

This section provides brief descriptions of the significant contributors to risk as determined by a review of all SMA accident sequence cutsets~~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 RBC collapse, core damage and large release.

RAI 19.01-8S1

A single SMA sequence 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 RBC event HCLPF shows up at the sequence level.

Risk Significance

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

RAI 19.01-2S2

The SMA shows that the current design meets the regulatory HCLPF requirement of 1.67 times the SSE~~, or~~(i.e., 0.84g). A structural failure sequence involving collapse of the RBC is the most important contributor to the seismic margin (and such collapse is relevant only if the RBC is under load within the operating module area of the RXB pool). Other sequences include one or more random failures after the seismic event. These failures occur among the same general components and sequences that lead to core damage in the internal events PRA. An examination of operating nuclear power plant data shows that the seismic survivability of the NuScale design is high because of the low core damage contribution from losses of offsite power. The only significant cutsets contain structural events leading directly to core damage and large release. All other seismically-induced initiating events require multiple seismic or common-cause random failures for core damage. This is largely a consequence of the low degree of reliance on electrical power for achieving safe shutdown. The passive actuation features of safe shutdown functions also imply a low degree of reliance on operator intervention to mitigate a severe accident.

19.1.5.2 Internal Fires Risk Evaluation

An internal fire probabilistic risk assessment (FPRA) for at-power operations has been performed for a single NuScale module. Section 19.1.5.2.1 describes key aspects of the evaluation including methodology and modeling. Section 19.1.5.2.2 provides key results including the CDF, LRF, and CCFP due to internal fire events.

19.1.5.2.1 Description of Internal Fire Risk Evaluation

The internal fire risk evaluation addresses the potential fire events that may originate within the plant boundary and that affect a single module. The FPRA is based on the Level 1 internal events PRA model, which is supplemented by fire-specific failure modes. Because detailed layout information (e.g., cable routing) is not available, detailed fire modeling is not performed.

The internal FPRA applies the methodology provided in NUREG/CR-6850 (Reference 19.1-42); the methodology consists of 16 interrelated tasks. The tasks are implemented as summarized in the following discussion. The discussion

RAI 19.01-2S2, RAI 19.01-3, RAI 19.01-3S1, 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 Correlation Class	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	3.32	0.24	0.32	1.32	No	DS
					Fails to Remain Closed						
					Spuriously Open						
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						
RVV3--75--RXM---FTC-SEIS	RVV3	75	RXM	All ECCS Reactor Vent Valves	Fails to Close	2.38	0.28	0.5	0.66	No	DS
					Fails to Remain Closed						
					Spuriously Open						
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
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	Reactor 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

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

Seismic Correlation Class	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]
EBA---100-HVSWG-FOP-SEIS	EBA	100	HVSWG	13KV AC Bus	Fails to Operate	5.9	0.24	0.39	2.09	No	Generic
EBA---100-LVPDC-FOP-SEIS	EBA	100	LVPDC	BDG Distribution Bus	Fails to Operate	2.8	0.24	0.39	0.99	No	Generic
EBD---86--RXB---FOP-SEIS	EBD	86	RXB	DC Bus Power Channel	Fails to Operate	3.55	0.24	0.39	1.26	No	Generic
HOV---100-RXM---FTC-SEIS	HOV	100	RXM	CVCS, CES, FWS, MSS Containment Isolation Valves	Fails to Close	22.13	0.27	0.37	7.72	Yes	DS
HOV---100-RXM---FTO-SEIS	HOV	100	RXM	CVCS, CFDS Containment Isolation Valves, DHRS Actuation Valves	Fails to Open	0.57	0.32	0.52	0.14	No	Generic
HOV---50--RXM---FOP-SEIS	HOV	50	RXM	ECCS Reactor Recirculation Valves	Fails to Operate (Passive Actuation)	9.52	0.27	0.37	3.32	Yes	DS
HOV---50--RXM---FTO-SEIS	HOV	50	RXM	ECCS Reactor Recirculation Valves	Fails to Open (Valve Body Deformation)	9.52	0.27	0.37	3.32	Yes	DS
HOV---75--RXM---FOP-SEIS	HOV	75	RXM	ECCS Reactor Vent Valves	Fails to Operate (Passive Actuation)	17.45	0.27	0.37	6.09	Yes	DS
HOV---75--RXM---FTO-SEIS	HOV	75	RXM	ECCS Reactor Vent Valves	Fails to Open (Valve Body Deformation)	17.45	0.27	0.37	6.09	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
HTX---50--RXM---HXF-SEIS	HTX	50	RXM	DHRS Heat Exchangers	Heat Exchanger Failure	2.34	0.32	0.51	0.6	No	Generic
MCC---86--RXB---FOP-SEIS	MCC	86	RXB	Low Voltage Motor Control Center	Fails to Operate	3.55	0.24	0.39	1.26	No	Generic
MDP---100-CHILL-FTR-SEIS	MDP	100	CHILL	DWS Pumps	Fails to Run	4.7	0.27	0.43	1.49	No	Generic
MDP---100-RXB---FTR-SEIS	MDP	100	RXB	CFDS Makeup Pumps	Fails to Run	2.3	0.27	0.43	0.73	No	Generic
MDP---50--RXB---FTR-SEIS	MDP	50	RXB	CVCS Makeup Pumps	Fails to Run	4.05	0.27	0.43	1.28	No	Generic
MOV---100-RXM---FTC-SEIS	MOV	100	RXM	CVCS MOV Recirculation Valve	Fails to Close	~~22.13~~ 0.57	~~0.27~~ 0.32	~~0.37~~ 0.52	~~7.72~~ 0.14	No	Generic
MOV---100-RXM---FTO-SEIS	MOV	100	RXM	CVCS MOV Injection Valve	Fails to Open	0.57	0.32	0.52	0.14	No	Generic
MSW---75--CRB---FTC-SEIS	MSW	75	CRB	Manual Division Actuation Switches	Fails to Close	4.78	0.24	0.39	1.7	No	Generic
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						