RAIO-0518-60071

May 18, 2018

Docket No. 52-048

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

SUBJECT: NuScale Power, LLC 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 November 27, 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:

- 19.01-2
- 19.01-5

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Samuel Lee, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A
Prosanta Chowdhury NRC, OWFN-8G9A

Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8899

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

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 to replace that response in its entirely. This revised response is provided as a result of discussions with the NRC during the recent PRA audit (ML18053A216). Consistent with those discussions, this response clarifies the use of fourteen seismic event trees (each associated with a specific ground motion) and the high confidence of low probability of failure (HCLPF) screening criteria. This response also deletes references to corbel bearing failure, consistent with NuScale's response to RAI 8899 (Question 19.01-8) provided in letter RAIO-0118-58238 dated January 17, 2018.

The term "PRA-critical" is used to denote structures, systems, and components (SSCs) that contribute to the seismic margin. In the NuScale seismic margin assessment (SMA), the plant-level HCLPF is determined by examining all of the cutset results from all fourteen seismic event trees, which contain both random and seismic failures (each seismic event tree represents a portion of the ground motion range from 0.005g to 4.0g). In each event tree, the initiating event frequency is set to unity to capture the conditional probability of core damage given a seismic event. Multiple event trees are used for two reasons:

1. The different combinations of failure events (combinations of random and seismically-induced failures) reveal which combinations of seismic and random failures are most likely for different earthquake intensities.
2. A systematic process considering several ground motions is useful for avoiding cutset truncation issues if the seismic demand were set too low or too high.

All cutsets are reviewed to screen out those that are not relevant to the determination of the plant-level HCLPF. As indicated in the MIN-MAX screening assumption stated in FSAR 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 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 not be less than one percent. If the combined random failure probability of the cutset is below one percent, the cutset is not a relevant contributor to the HCLPF calculation. The MIN-MAX method is applied to each cutset. Each cutset retained in the HCLPF evaluation corresponds to an SSC that contributes to the seismic margin. Of all the seismic margin contributors, the SSC with the smallest HCLPF value provides the plant-level HCLPF. FSAR Sections 19.1.5.1.1.4 and 19.1.5.1.2 have been modified to clarify the process to determine the SSCs that contribute to the seismic margin (i.e., are PRA-critical), the plant-level HCLPF, and the use of fourteen seismic event trees.

The first sentence of the paragraph cited in the original question does refer to PRA-critical

SSCs. The first sentence of the cited paragraph in FSAR Section 19.1.5.1.1.3 was modified for clarification in FSAR Revision 1 to refer to seismic capacities for “PRA-critical” structures and components. The SMA also includes seismic failures that do not contribute to the seismic margin, i.e., “non-critical” SSCs. Non-critical SSCs either have detailed fragility evaluations confirming their characterization as “non-critical” or have evaluations of fragility determined with generic information, modified with NuScale-specific seismic demand. All reactor building structures modeled in the SMA contribute to the seismic margin (i.e., are PRA-critical).

Impact on DCA:

FSAR Sections 19.1.5.1.1.3, 19.1.5.1.1.4, and 19.1.5.1.2 have been been revised as described in the response above and as shown in the markup provided in this response.

Structure, system, and component fragility is referenced to the peak ground acceleration of the CSDRS, which is the SSE (0.5g).

19.1.5.1.1.3 Seismic Fragility Evaluation

RAI 19.01-1S1, RAI 19.01-2, RAI 19.01-2S1, 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 Section 19.1.5.1.2 and Table 19.1-40.

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 19.1-35. 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). 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 a 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 NPM. Structures selected for analysis meet one of the following criteria:

This methodology was chosen so that NuScale-specific response data is reflected in the evaluation of component fragility.

19.1.5.1.1.4 Systems and Accident Sequence Analysis

Plant response analysis maps the consequences of seismic initiators combined with seismic and random failures. This analysis produces event trees with seismically induced initiating events, component and structural events, and non-seismic unavailability.

The SAPHIRE computer code is used for quantification of the logic models utilized in the NuScale SMA.

Seismically-Induced Initiators

Plant response after a seismic event is mapped using seismically-induced event initiators, as illustrated in Figure 19.1-16. The seismically-induced initiators are modeled using similar logic to their corresponding random internal events PRA initiators. Plant response is modeled only for earthquakes with a non-negligible probability of causing a reactor trip.

The lowest threshold for seismically-induced initiators is a LOOP, which has a median failure capacity of 0.3g. A seismically-induced LOOP credits AC power recovery from the CTG or the BDGs (A_m = 0.65g for both). If both the turbine 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), breaks outside containment (CVCS regenerative heat exchanger failure) and (most severely) structural events.

RAI 19.01-2S1

~~The seismic hazard for the NuScale design has been partitioned into 14 seismic initiating event trees defining the SMA. The underlying logic for each tree is identical. However, each tree represents a different ground motion acceleration.~~The seismic hazard for the NuScale design SMA is partitioned into fourteen seismic event trees. The underlying logic for each event tree is identical; however, each event tree represents a different ground motion acceleration (each seismic event tree represents a portion of the ground motion range from 0.005g to 4.0g). In the SMA, the use of multiple ground motions provides insights into the relative contributions of both seismic and random failures at different ground motions. Figure 19.1-16 is a representative seismic event tree, corresponding to a range of peak ground accelerations from

0.005g to 0.1g. In each event tree, the initiating event frequency is set to unity in the SMA to produce the conditional probability of core damage given a seismic event.

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 by the failure of a single component or structural 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.

Offsite power loss is the most likely induced initiator (a LOOP would occur from lower ground motions than are expected for other induced initiators). As such, credit for offsite power has been removed from the other induced initiator trees. In the event of a LOOP, as illustrated in Figure 19.1-20, credit is considered for the CTG and BDGs. If either survives along with the DC buses, the response to a general reactor trip is considered, as indicated by the transfer "TGS---TRAN--NPC-ET" (Figure 19.1-11). If neither survives, offsite and onsite power has been lost and a station blackout exists. Because backup power is fragile relative to the valves and steam generator tubing for the other induced initiator trees, the existence of power in those situations is not considered in the other seismic initiator trees. If backup power is unavailable due to the seismic event (Sequence 5 of Figure 19.1-20), a transfer is made to the internal event LOOP event tree (Figure 19.1-9).

RAI 19.01-2S1

In developing the SMA, system fault trees also are modified. Seismic failure modes for structures and components are incorporated by inserting transfer gates for each seismic correlation class into each existing fault tree alongside existing randomly occurring events (failure modes). Events representing failure modes without seismically-relevant equivalents remain in the SMA. They are inserted as the union of existing random failure events with the seismic failure of the seismic correlation class. Once complete, the SMA is representative of seismic failures to different component groups located throughout the plant as well as original random failures. Updated fault tree logic is transferred through the logic of each seismic ~~initiating~~ event tree. Because ~~14~~fourteen event trees are utilized to define the seismic hazard, the appropriate ground motion demand corresponding to each event tree is applied with "house" events. These events coincide with the ground motion acceleration modeled with each individual seismic event tree. Project level linkage rules are used to turn house events true or false in order to solve each seismic event tree at the correct ground motion.

When evaluating the SMA model, each seismic event tree may be analyzed independently to determine the conditional core damage cutsets related to a single ground motion acceleration bin. The SMA cutsets contain both random and seismic failures. Cutsets from the model evaluation are subsumed by gathering the cutsets from a particular end state. From the gathered end state interface in SAPHIRE, subsets of cutsets can be viewed by using the SAPHIRE slice function. In the seismic event trees, sequences involving core damage end with "Level2-ET." This indicates a transfer to the containment event tree (Figure 19.1-15), which contains the radionuclide release categories.

In summary, the SMA event trees terminate in:

- OK: No core damage
- Transfer to another event tree
- Transfer to the Level 2 event tree.

RAI 19-4

19.1.5.1.1.5 Effects of Seismically Failed SSC on Surviving SSC

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

RAI 19-4

a) Proximity effects

Safe shutdown of an NPM is ensured by opening of the RSVs, combined with successful passive ECCS valve operation, when there is not a loss of coolant outside the containment boundary. These components have very high seismic capacities and are physically shielded from nonseismically qualified SSC by the seismically qualified CNV. These components fail safe on loss of power and are not located in proximity to nonseismically qualified components.

RAI 19-4

b) Structural failure and falling

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

RAI 19-4

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

RAI 19-4

c) Flexibility of attached lines and cables

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

RAI 19-4

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

19.1.5.1.2 Results from the Seismic Risk Evaluation

RAI 19.01-17

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 SSE, which equates to a 0.84g peak ground acceleration for NuScale.

RAI 19.01-2S1

~~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.~~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 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 not be less than one percent. 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 cutset. Each cutset retained in the HCLPF evaluation corresponds to an SSC that contributes to the seismic margin. Of all the seismic margin contributors, the SSC with the smallest HCLPF value provides the plant-level 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 19.1-35 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

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 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 shows up at the sequence level.

Risk Significance

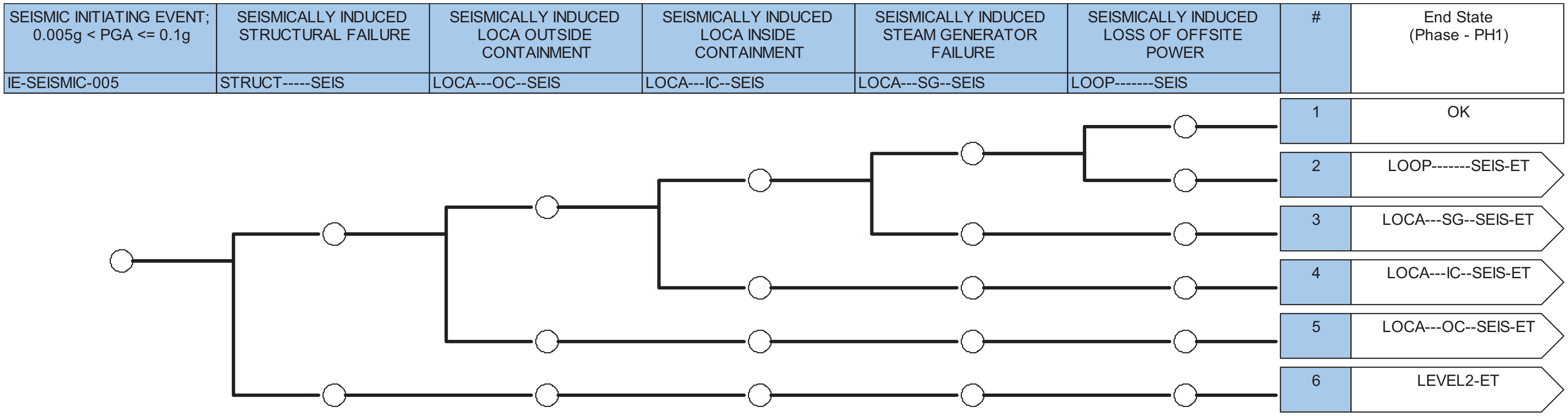
Potentially risk significant structures, components and operator actions are discussed below.

Significant Structural Failures

RAI 19.01-8S1

RAI 19.01-2S1

Figure 19.1-16: Representative Seismic ~~Initiating~~ Event Tree



NuScale Power

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

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

NRC Question No.: 19.01-5

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:

NuScale is supplementing its response to RAI 8899 (Question 19.01-5) originally provided in letter RAIO-0917-55781 dated September 01, 2017 and supplemented in letter RAIO-1117-57364 dated November 27, 2017. This supplemental information is provided as a result of discussions with the NRC during the PRA audit (ML18053A216) regarding the methods of developing structure, system, and component (SSC) fragilities. Information provided in NuScale letters RAIO-0917-55781 and RAIO-1117-57364 related to Question 19.01-5 is unaffected by this supplemental response.

FSAR Revision 1 Section 19.1.5.1.1.3 states that two methods, “hybrid” and “separation of variables”, are used to develop NuScale seismic margin assessment (SMA) fragility calculations for SSCs that contribute to the seismic margin. Consistent with guidance provided by the Electric Power Research Institute (EPRI) on determining a high confidence of low probability of failure (HCLPF) ground motion (Reference EPRI 1002988, Section 2.2.6), the conservative deterministic failure margin method (CDFM) is an element of the hybrid method. The separation of variables method and CDFM were used to calculate HCLPF values, as explicitly endorsed by DC/COL-ISG-020. Accordingly, FSAR Section 19.1.5.1.1.3, with associated references, has been modified to clarify the methodologies used to evaluate HCLPF values.

Impact on DCA:

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

Structure, system, and component fragility is referenced to the peak ground acceleration of the CSDRS, which is the SSE (0.5g).

19.1.5.1.1.3 Seismic Fragility Evaluation

RAI 19.01-1S1, RAI 19.01-2, RAI 19.01-2S1, 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 Section 19.1.5.1.2 and Table 19.1-40.

RAI 19.01-2, RAI 19.01-5S2

~~Seismic capacities~~The HCLPF ground motion 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~~separation of variables and conservative deterministic failure margin (CDFM) methods, as endorsed by Reference 19.1-56. Separation of variables, described in Reference 19.1-57, is a best-estimate methodology to determine SSC fragility parameters (median capacity, randomness, and modeling uncertainty) as a combination of several independently determined factors (e.g., strength and ductility). The fragility parameters are then used to calculate the HCLPF. The CDFM method, described in Reference 19.1-21, uses conservative input parameters (e.g., seismic demands and material properties) to directly establish a conservative estimate of the HCLPF. 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 19.1-35. 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). 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

19.1-16 NUREG/CR-6890 "Reevaluation of Station Blackout Risk at Nuclear Power Plants, Analysis of Loss of Offsite Power Events: 1986-2004" with subsequent updates including: "Analysis of Loss of Offsite Power Events 2013 Update."

19.1-17 EPRI NP-2230 "ATWS: A Reappraisal. Part 3. Frequency of Anticipated Transients."

19.1-18 NUREG/CR-2680 "Seismic Safety Margins Research Program: Equipment Fragility Data Base," Lawrence Livermore National Laboratory, January 1983, U.S. Nuclear Regulatory Commission.

19.1-19 NUREG/CR-3558 "Seismic Safety Margins Research Program: Handbook of Nuclear Power Plant Seismic Fragilities," Lawrence Livermore National Laboratory, June 1985. U.S. Nuclear Regulatory Commission.

19.1-20 NUREG/CR-4659 "Seismic Fragility of Nuclear Power Plant Components," Brookhaven National Laboratory, August 1991, U.S. Nuclear Regulatory Commission.

RAI 19.01-5S2

19.1-21 ~~EPRI 1002989, "Seismic Probabilistic Risk Assessment Implementation Guide," Electric Power Research Institute, Palo Alto, CA, December 2003~~EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)," Electric Power Research Institute, Palo Alto, CA, August 1991.

19.1-22 NUREG/CR-6883 "The SPAR-H Human Reliability Analysis Method," INL/EXT-05-00509, August 2005. U.S. Nuclear Regulatory Commission.

RAI 19-30

19.1-23 NUREG/CR-6928, Initial Issue 2007 and 2010 Update, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 2012.

19.1-24 INL/EXT-16-37873 "Analysis of Loss-of-Offsite-Power Events 1997-2014", Idaho National Laboratory, February 2016.

19.1-25 NUREG/CR-7040 "Evaluation of JNES Equipment Fragility Tests for Use in Seismic Probabilistic Risk Assessments for U.S. Nuclear Power Plants", April 2011, U.S. Nuclear Regulatory Commission.

19.1-26 Nuclear Energy Institute White Paper, "Proposed Methodology and Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone," NEI Small Modular Reactor Licensing Task Force, December 2013.

19.1-27 NUREG/CR-7039 "System Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE)," Version 8, Volumes 1-7, June 2011, U.S. Nuclear Regulatory Commission.

19.1-28 MELCOR Computer Code Manuals, Vol. 1 through Vol. 3, Version 2.1.