Response to SDAA Audit Question

Question Number: A-19.1-20

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Question:

There are several locations in the NuScale SDA in Chapter 19 where editorial changes can provide clarification. Explain the following or clarify the SDA text in the following locations:

1. In Section 19.1.4.1.3, “Success Criteria,” in the discussion of fuel assembly heat removal, the sentences “Depending on the IE and accident sequence, core cooling can be achieved passively by actuation of the DHRS or the ECCS. In the absence of these preferred, automatic methods, operator action can establish CVCS makeup inventory to the RPV or flood the CNV from the CFDS following ECCS success” are unclear. In the 2nd sentence, the statement appears to indicate the absence of ECCS and the last clause of the statement seems to require ECCS.
2. In Section 19.1.5.1.1.2, under “Seismically-Induced Initiators,” there is a parenthetical that states “each seismic event tree represents a portion of the ground motion range from 0.0525g to 4.0g.” A few sentences later, it is stated that for a representative seismic event tree, corresponding peak ground accelerations are from 0.005g to 0.1g. The 0.0525g reference appears to be to the mid-point of the lowest range and not indicative of the lowest ground motion.
3. The first paragraph of Section 19.1.5.1 includes the statement: “A PRA-based SMA provides information related to the dominant contributors to seismic risk by determining plant responses from different ground motion demands, i.e., a range of reference earthquakes (REs).” In contrast, the first paragraph of Section 19.1.5.1.1, includes the statement, “The RE is the CSDRS with a horizontal PGA of 0.5g.” Confirm that there is only one RE with the definition above or provide additional information related to the use of how multiple REs were used to determine the HCLPF value. It is understood that the use of multiple ground motions was used to provide insights to the relative contributions of failures at different ground motions.
4. In Section 19.1.10, Reference 19.1-16, Electric Power Research Institute, “Advanced Light Water Reactor Passive Plant Utility Requirements Document,” Rev. 13, EPRI 3002003129, EPRI, Palo Alto, CA, 2024 appears to either have the wrong report number or be the wrong

document. EPRI 3002004129 is the Advanced Nuclear Technology: Advanced Light Water Reactor Utility Requirements Document, Revision 13. EPRI 3002000507 is the Advanced Nuclear Technology Advanced Light Water Reactor Utility Requirements Document, Revision 12.

Response:

1. The two referenced sentences are intended to demonstrate the defense-in-depth of the US460 design for providing core cooling, including both automatic and manual options. NuScale acknowledges the current text is unclear and proposes a change to the FSAR, as shown in the markup provided below, to clarify that the methods listed for fulfilling the fuel assembly heat removal function are dependent on the initiating event and accident sequence.

2. The ground motion range referenced in the identified statement from Section 19.1.5.1.1.2 of the standard design approval application (SDAA) should read, “0.005g to 4.0g.” This error has been corrected, as shown in the markup provided below.

3. The noted statement from Section 19.1.5.1 of the SDAA describes the process of using multiple ground motions to evaluate seismic risk contributors in a PRA-based seismic margin assessment (SMA). The plant-level high confidence of low probability of failure (HCLPF) capacity is referenced to a single reference earthquake, which is the certified seismic design response spectra (CSDRS) with a horizontal peak ground acceleration of 0.5g. NuScale acknowledges the noted use of the term “reference earthquakes” can create confusion when describing a PRA-based SMA. Section 19.1.5.1 has been revised to remove this term, as shown in the markup provided below.

4. The title of the identified document from Section 19.1.10 of the SDAA should read, “Advanced Light Water Reactor Utility Requirements Document.” The EPRI document number (3002003129) and revision number (13) are correct. This error has been corrected, as shown in the markup provided below.

Markups of the affected changes, as described in the response, are provided below:

for the event tree sequences is performed by defining success in three progressive stages: overall success criterion, functional success criteria, and system success criteria.

The overall success criterion is prevention of core damage. Accident sequences that are considered success or "OK" do not result in core damage for the duration of the mission time defined for the PRA, and end in a stable or improving NPM configuration using the following definitions:

- Mission time is the period of time that a system or component is required to operate successfully to perform its function. Mission times are specified for components that are required to operate following an initiating event. Mission times take into account the time needed to reach a safe, stable, long-term condition, and time needed to establish long-term recovery actions. The PRA mission time is 72 hours.
- Core damage is defined as occurring when:
  - the collapsed level in the reactor has decreased such that active fuel in the core has been uncovered for a sustained period, and
  - a fuel peak cladding temperature (PCT) of 2200 degrees Fahrenheit or higher is reached as defined by the thermal-hydraulic calculation.

Functional success criteria are then developed based on the safety functions necessary to support the overall success criterion. The functional success criteria are the minimum set of functions whose success is needed to prevent core damage and a large release. The safety functions and method of achieving the functions are summarized as follows:

Audit Issue A-19.1-20

- Fuel assembly heat removal: This function refers to the transfer of core heat to the UHS after a module upset. The function can be achieved by safety-related or nonsafety-related systems that can provide core cooling. Depending on the IE and accident sequence, core cooling can be achieved passively by actuation of the decay heat removal system (DHRS) or the ECCS~~. In the absence of these preferred, automatic methods~~, or actively by operator action to~~can~~ establish chemical and volume control system (CVCS) makeup inventory to the RPV or flood the CNV from the CFDS~~ following ECCS success~~.
- Reactivity control: This function refers to the limiting of core power generated by the fission reaction. The function is achieved if the core is rendered subcritical by insertion of control rods as demanded by a reactor trip signal. In an anticipated transient without scram (ATWS) event, as the fuel heats up and the moderator density decreases, core power is reduced; this negative reactivity feedback maintains fuel assembly heat removal while avoiding core damage. In sequences where makeup inventory via CVCS is credited, operators initiate makeup with suction from the boron addition system (BAS), which can be used to support reactivity control. In addition, in sequences with

19.1.5.1 Seismic Risk Evaluation

Audit Issue A-19.1-20

Evaluation of the risk due to seismic events is performed using a seismic margins assessment (SMA) to determine the plant-level high confidence of low probability of failure (HCLPF) ground motion capacity. A PRA-based SMA provides information related to the dominant contributors to seismic risk by determining plant responses from different ground motion demands~~, i.e., a range of reference earthquakes (REs)~~. Because the plant lacks a reliance on electrical power, added water, or operator actions, the design is less susceptible to low capacity accident progressions (i.e., those from small ground motions) than typical operating nuclear power plants. Consequently, seismically-induced major structural failures associated with higher ground motions, which are typically a minor contributor to the seismic risk for operating plants, represent a significant risk contributor for the NuScale design. A PRA-based SMA is developed to confirm that plant responses initiated from large ground motions are accounted for.

The SMA for the NPM is performed in accordance with NRC guidance from Section 19.0 of NUREG-0800, Revision 3 and the applicable SMA guidance in Part 5 of ASME/ANS RA-Sa-2009 as endorsed by Regulatory Guide 1.200.

19.1.5.1.1 Description of the Seismic Risk Evaluation

The primary goal of an SMA is to identify the SSC that contribute to seismic risk. The SSC identification is done by evaluating SSC risk contributors and determining the plant-level HCLPF ground motion capacity.The plant-level HCLPF ground motion capacity must be 167 percent of the RE used for design, or the review level earthquake (RLE). The RE is the CSDRS with a horizontal PGA of0.5g. Thus the plant-level HCLPF ground motion capacity requirement is 0.84g PGA (i.e., 1.67 * 0.5g). There are two main tasks associated with performing an SMA: seismic fragility analysis (structures and components), and seismic plant response analysis (accident sequence analysis and plant level response).

19.1.5.1.1.1 Seismic Analysis Methodology and Approach

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. All SSC modeled in the internal events PRA are included in fragility analysis, with the exception of basic events that are not subject to seismically-induced failure (e.g., phenomenological events, filters, control logic components). No pre-screening is performed to establish a seismic equipment list (SEL) or safe shutdown equipment list (SSEL). SSC that contribute to the seismic margin are determined by applying the MIN-MAX method described in Section 19.1.5.1.2.

The HCLPF ground motion for SSC that contribute to the seismic margin is obtained by performing fragility analysis using the separation of variables the ISRS is used. For SSC located in the RXB, an enveloped floor ISRS for all locations on an elevation is used to describe the SSC seismic demand. For SSC located on or near the NPM, but do not contribute to the seismic margin (e.g., DHRS heat exchangers), broadened ISRS is used at the equipment anchorage location.

Each SSC fragility is calculated based on floor responses. Consequently, each fragility is multiplied by the PGA of the RE (0.5g) to anchor the median capacity to the seismic input defined for design (i.e., the CSDRS). Each component fragility is then determined as a function of design loads, placement, and site response.

The HCLPF is then defined as the acceleration level where there is a 95 percent confidence of less than 5 percent failure probability. The HCLPF can also be approximated as the acceleration with a one percent probability of failure on the mean fragility curve.

Results of the fragility calculation for the NPM supports are shown in Table 19.1-32.

19.1.5.1.1.2 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 initiating events, as illustrated in Figure 19.1-14. These events are modeled using similar logic to corresponding random internal events PRA initiating events. Plant response is modeled only for earthquakes with a non-negligible probability of causing a reactor trip.

Audit Issue A-19.1-20

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 0.005~~25~~g 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 19.1-14 is a representative seismic event tree, corresponding 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

19.1-13 Idaho National Laboratory "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants: 2020 Update," INL/EXT-21- 65055, November 2021.

19.1-14 Sandia National Laboratories, "MELCOR Computer Code Manuals," (Version 2.2), Vol. 1 and Vol. 2, Albuquerque, NM, January 2021.

19.1-15 U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses Project Uncertainty - Analysis of the Unmitigated Short-Term Station Blackout of the Surry Power Station," (Draft Report), Agencywide Documents Access and Management System (ADAMS) Accession No. ML15224A001.

Audit Issue A-19.1-20

19.1-16 Electric Power Research Institute, "Advanced Nuclear Technology: Advanced Light Water Reactor ~~Passive Plant~~ Utility Requirements Document," ~~Rev. 13,~~ EPRI # 3002003129, Revision 13, EPRI, Palo Alto, CA, December 05, 2014.

19.1-17 U.S. Nuclear Regulatory Commission, "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Republication," *Federal Register*, Vol. 51, No. 162, August 21, 1986, pp. 30028-30033.

19.1-18 Nuclear Energy Institute, "Guidance for Post Fire Safe Shutdown Circuit Analysis," NEI 00-01, Revision 2, May 2009.

19.1-19 Nuclear Energy Institute, "Guidance for Post Fire Safe Shutdown Circuit Analysis," NEI 00-01, Revision 3, October 2011.

19.1-20 Nuclear Energy Institute, "External Flooding Integrated Assessment Guidelines," NEI 16-05, Revision 0, April 2016.

19.1-21 Quanterion Solutions Incorporated, "Quanterion Automated Databook: Electronic Parts Reliability Data 2014 (EPRD-2014), Nonelectric Parts Reliability Data 2011 (NPRD-2011), Failure Mode/Mechanism Distribution 2013 (FMD-2013)," Utica, NY.

19.1-22 Electric Power Research Institute, "An Analysis of Loss of Decay Heat Removal and Loss of Inventory Event Trends (1990-2009)," EPRI #1021167, EPRI, Palo Alto, CA, 2010.

19.1-23 Electric Power Research Institute, "Methodology for Developing Seismic Fragilities," EPRI #103959, EPRI, Palo Alto, CA, 1994.

19.1-24 Electric Power Research Institute, "Seismic Fragility Applications Guide Update," EPRI #1019200, EPRI, Palo Alto, CA, 2009.

19.1-25 Electric Power Research Institute, "Advanced Light Water Reactor Passive Plant Utility Requirements Document," Rev. 13, EPRI #3002000507, EPRI, Palo Alto, CA, 2014.