Response to SDAA Audit Question

Question Number: A-19.1-53

Receipt Date: 11/20/2023

Question:

In Section 19.1.7, regarding drop of a module on an operating module, the SDA states, “If the NPM is struck near the top, the CVCS injection line and DHRS heat exchanger piping at the front of the NPM is likely to be damaged, rendering these systems unavailable. If the NPM is struck near its bottom, the low speed of the impact and distance from important components will allow safety systems to be nominally available. In both cases, the CNVs of both NPMs are unlikely to be breached due to the relatively low velocity of impact, caused by the dropped NPM falling only a short distance through the resistive medium of reactor pool water.”

Standard Review Plan, Chapter 19.0, Revision 3, page 19.0-22 identifies the need for in-depth NRC review of refueling operations for small, modular reactors, which are different from traditional LWRs, to ensure that the PRA model is of acceptable scope and level of detail. This same page also directs staff to verify that applicants for plants with multiple modules use a systematic process to identify accident sequences, including significant human errors that could lead to core damage or large release from multiple modules.

a) NuScale is requested to explain if the capability of the containment isolation valves (CIVs) to close is compromised, given that the strike to the operating module has sufficient force to cause pipe breaks. In this context, NuScale is requested to describe any protection for the CIVs provided by the Module Top Support Structure and the Lower Block Assembly (LBA), located at the bottom of the main hoist which have been re-designed in the SDA compared to the DCA.
b) NuScale is requested to explain if the RTS is compromised given that the strike to the operating module has sufficient force to cause pipe breaks. NuScale is requested to describe any protection for the RTS provided by the Module Top Support Structure and the Lower Block Assembly (LBA), located at the bottom of the main hoist which have been re-designed in the SDA compared to the DCA.
c) NuScale is requested to explain whether the success criteria for LOCAs outside

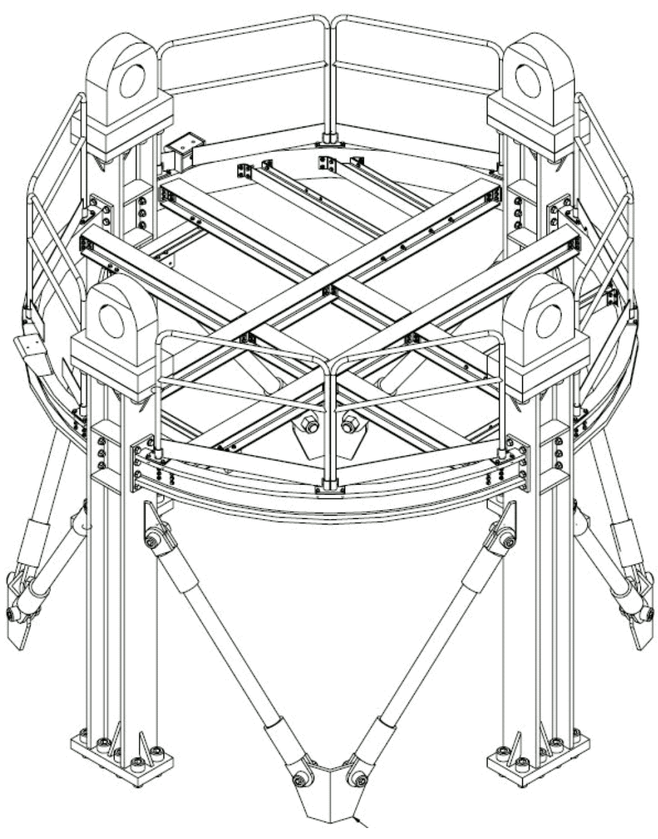
containment changes if containment isolation fails given that the DHRS could be failed simultaneously with breaks in the CVCS inlet and discharge piping, and the CFDS piping.

Response:

a) The safety-related chemical and volume control system (CVCS) containment isolation valves are located on top of the containment vessel and under the NuScale Power Module (NPM) top support structure (TSS). As shown below in Figure 1, Containment Vessel Top Support Structure Assembly, and in FSAR Figure 9.1.5-3, the TSS is composed of diagonal lifting braces and lifting lugs, and provides structural support for piping and valves. The lower block assembly is located at the bottom of the main hoist and interfaces with the TSS; the lower block assembly provides the connection method for the Reactor Building crane to connect to, lift, and carry an NPM between an operating bay and the refueling bay. {{

}}2(a),(c)

Figure 1. Containment Vessel Top Support Structure Assembly



b) The reactor trip system provides both automatic and manual trip capability. A reactor trip system signal opens the reactor trip breakers, which interrupts the control rod drive power supply, and the control rods drop into the core. As shown in FSAR Figure 4.6-1, the control rod drive mechanisms (CRDMs) are mounted on top of the reactor pressure vessel, inside the containment vessel; control rod assemblies connect to the CRDMs at the bottom of the drive shaft. {{

}}2(a),(c)

c) Accident sequence logic for CVCS injection and discharge line breaks outside containment is shown in FSAR Figures 19.1-2 and 19.1-3, respectively. Following a postulated break, a reactor trip on low pressurizer level or low pressurizer pressure is expected. Low pressurizer level will actuate containment isolation, including isolation of the CVCS lines. If the break is not isolated by one of the two redundant, safety-related containment isolation valves, continued loss of inventory will result in emergency core cooling system actuation on low reactor pressure vessel riser level. Success of one train of the decay heat removal system and both trains of the emergency core cooling system can mitigate the break without makeup inventory. Without at least one train of the decay heat removal system operating (to help reduce pressure inside the reactor pressure vessel), makeup inventory would be needed to mitigate an unisolated CVCS line break outside containment.

In response to the clarification call with the NRC and NuScale held on 01/24/2024, NuScale is providing the following additional information:

Consistent with FSAR Section 19.1.7.4, in the event that a dropped NPM strikes an operating NPM, piping at the front of the operating NPM has the potential to be impacted, which includes CVCS (i.e., pressurizer spray) and decay heat removal system piping.

As indicated by FSAR Figure 6.2-2b, Containment Vessel Assembly, the containment flooding and drain system piping is located on the back of the containment vessel head, the CVCS injection and discharge piping is located on the side, and the pressurizer spray piping is located on the front. NuScale has revised the description of an NPM strike in FSAR Section 19.1.7.4.

In response to the clarification call with the NRC and NuScale held on 05/24/2024, NuScale is providing the following additional information:

At the request of the staff, NuScale is attaching the relevant pages from Revision 2 of the FSAR that indicate

- the lower block assembly (as part of the Reactor Building crane) is classified as B1 (Table 9.1.5-2, Classification of Structures, Systems and Components).
- the top support structure is classified as B1 (Table 17.4-1, Design Reliability Assurance Program Structures, Systems, and Components Functions, Categorization, and Categorization Basis).
- tables identifying the classification of structures, systems, and components are provided in applicable FSAR sections (Section 3.2, Classification of Structures, Systems, and Components).

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

9.1.5 Overhead Heavy Load Handling Systems

The overhead heavy load handling system (OHLHS) consists of equipment that lifts loads whose weight is greater than the combined weight of a single fuel assembly and control rod assembly. Loads weighing more than 900 lbs are defined as heavy loads. The primary purpose of the OHLHS is to support movement of a NuScale Power Module (NPM) for refueling.

The principal equipment of the OHLHS consists of the Reactor Building crane (RBC), and various hoists and heavy load handling devices used in the Reactor Building (RXB):

- Reactor Building crane
- traveling jib crane (TJC)
- articulating traveling jib crane (ATJC)
- dry dock jib crane
- module access platform (MAP) jib crane
- auxiliary wet hoist (AWH)

Additional equipment used to inspect, assemble, and disassemble the NPM for refueling is also discussed in this section:

- NPM top support structure (TSS)
- lower riser lifting and torque tool (LRLTT)
- other refueling devices:
  - reactor flange tool (RFT)
  - containment flange tool (CFT)
  - module inspection rack

The OHLHS also includes instrumentation, physical stops, electrical interlocks, and associated administrative controls.

The RBC is classified as nonsafety-related, risk-significant.

The remaining components of the OHLHS are classified as nonsafety-related and not risk-significant. Table 9.1.5-2 identifies SSC classifications for the OHLHS.

Critical load handling is defined as the handling of a heavy load where inadvertent operations or equipment malfunctions, separately or in combination, could cause a release of radioactivity, a criticality accident, the inability to cool fuel within the reactor vessel or spent fuel pool, or prevent safe shutdown of the reactor.

COL Item 9.1-4: An applicant that references the NuScale Power Plant US460 standard design will describe the process for handling and receipt of critical loads including NPMs.

Table 9.1.5-2: Classification of Structures, Systems, and Components

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	Augmented Design Requirements (Note 2)	Quality Group/Safety Classification (Ref RG 1.26 or RG 1.143) (Note 3)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 4)
OHLHS, Overhead Heavy Load Handling System					
Reactor building crane (RBC main hoist and lower block assembly, and RBC sister hook)	RXB	B1	ASME NOG-1	N/A	I
Reactor building crane auxiliary hoist	RXB	B1	ASME NUM-1	N/A	I
• Traveling jib crane • Articulating traveling jib crane • Dry dock jib crane • Module access platform jib crane	RXB	B2	ASME NUM-1	N/A	II
Auxiliary wet hoist	RXB	B2	ASME NUM-1	N/A	II

Note 1: Acronyms used in this table are listed in Table 1.1-1

Note 2: Additional augmented design requirements, such as the application of a Quality Group, Radwaste safety, or seismic classification, to nonsafety-related SSC are reflected in the columns Quality Group / Safety Classification and Seismic Classification, where applicable. Environmental Qualifications of SSC are identified in Table 3.11-1.

Note 3: Section 3.2.2.1 through Section 3.2.2.4 provides the applicable codes and standards for each RG 1.26 Quality Group designation (A, B, C, and D). A Quality Group classification per RG 1.26 is not applicable to supports or instrumentation that do not serve a pressure boundary function. Section 3.2.1.4 provides a description of RG 1.143 classification for RW-IIa, RW-IIb, and RW-IIc.

Note 4: Where SSC (or portions thereof) as determined in the as-built plant that are identified as Seismic Category III in this table could, as the result of a seismic event, adversely affect Seismic Category I SSC or result in incapacitating injury to occupants of the control room, they are categorized as Seismic Category II consistent with Section 3.2.1.2 and analyzed as described in Section 3.7.3.8.

If the dropped NPM remains partially upright, such as if it is supported by another NPM or RXB structure, it is assumed that core damage is avoided; conversely, if it is not supported and falls to the floor core damage is assumed to occur.

The effects of a NPM being struck by a dropped NPM are determined by engineering judgment. The closest analog for an accident sequence for a struck NPM is a general reactor trip. It is reasonable to expect that operators will be closely monitoring a NPM transport, and will manually trip nearby NPMs if the RBC fails.

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If the NPM is struck near the top, the CVCS ~~injection line~~ and DHRS ~~heat exchanger~~ piping at the front of the NPM is likely to be damaged, rendering these systems unavailable. If the NPM is struck near its bottom, the low speed of the impact and distance from important components will allow safety systems to be nominally available. In both cases, the CNVs of both NPMs are unlikely to be breached due to the relatively low velocity of impact, caused by the dropped NPM falling only a short distance through the resistive medium of reactor pool water. Likewise, either struck NPM being dislodged from its operating bay is not judged to be credible as the seismic restraints limit horizontal motion, and the weight of the NPM and downward angle at which it is struck will prevent it from being lifted high enough to escape its bay.

19.1.8 Probabilistic Risk Assessment-Related Input to Other Programs and Processes

The PRA supporting the standard design has been used to support the NuScale design. The following sections summarize the uses of the PRA.

19.1.8.1 Probabilistic Risk Assessment Input to Design Programs and Processes

As discussed in Section 19.1.1.1 the uses of the PRA during the design phase are summarized in Table 19.1-1, which also indicates the applicable section in which the PRA application is discussed. The following sections address specific applications of the PRA.

19.1.8.2 Probabilistic Risk Assessment Input to the Maintenance Rule Implementation

Use of the PRA in supporting the Maintenance Rule is determined by the program for monitoring the effectiveness of maintenance, which is addressed in Section 17.6.

19.1.8.3 Probabilistic Risk Assessment Input to the Reactor Oversight Process

The Reactor Oversight Process, the NRC program to assess the safety of an operating commercial nuclear power plant, is based in part on risk insights. The PRA developed for the standard design provides the basis for an as-built, as-operated PRA. The site-specific PRA is used to support the Reactor Oversight Process, including specific safety and performance metrics.

3.2 Classification of Structures, Systems, and Components

Structures, systems, and components (SSC) are classified according to nuclear safety classification, seismic category, and quality group. This classification aids in the determination of the appropriate quality standards and the identification of applicable codes and standards. The SSC classifications are based on a consideration of both safety-related functions (consistent with the definition of safety-related in 10 CFR 50.2) and risk-significant functions determined as part of the Design Reliability Assurance Program (D-RAP).

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Table 3.2-1 identifies the buildings associated with the site layout and their seismic classification. Table 3.2-2 identifies a list of Seismic Category I SSC that provide pressure integrity functions or their supports, for the reactor coolant pressure boundary (RCPB). Table 3.2-2 also provides the applicable Quality Assurance Program (QAP) requirements and quality group classification. Discussion of systems comprised of Seismic Category II and III SSC are provided in the applicable chapters. “Classification of Structures, Systems, and Components” tables identifying SSC classification, augmented design requirements, quality group, and seismic category are provided in applicable FSAR sections.

Seismic and quality group classification criteria are described in Section 3.2.1 and Section 3.2.2, respectively. The D-RAP process and categorization criteria are described in Section 17.4. Descriptions of the risk-significant Design Reliability Assurance Program structures, systems, and components classifications are found in Table 17.4-1.

The SSC classification process is applied at the component level based upon the system functions performed. At the system level, system functions are designated as safety-related or nonsafety-related, and risk-significant or not risk-significant. Components are then classified commensurate with the safety and risk-significance of the system function(s) they support. A system that primarily performs safety-related or risk-significant functions may include nonsafety-related, not risk-significant components, on the basis of those components only supporting nonsafety-related, not risk-significant secondary system functions. Similarly, components that support multiple system functions may include multiple design features, each related to the different system functions. Components with any safety or risk design feature are classified on the basis of that feature.

Safety-related SSC and risk-significant SSC are subject to the QAP requirements described in Section 17.5. Applicable portions of 10 CFR 50 Appendix B have been applied to some nonsafety-related SSC where specific regulatory guidance applies (e.g., Regulatory Guide (RG) 1.29 “Seismic Design Classification for Nuclear Power Plants”).

In addition to safety and risk-significance, the classification methodology includes consideration for “augmented” requirements for those SSC that are by definition nonsafety-related (based on the definition in 10 CFR 50.2). The selection of augmented requirements is based on a consideration of the important functionality to be performed by the nonsafety-related SSC and regulatory guidance applicable to the functionality (e.g., consistent with the functionality specified in General Design Criterion 60 for

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Table 17.4-1: Design Reliability Assurance Program Structures, Systems, and Components Functions, Categorization, and Categorization Basis

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
Containment System (CNTS)			
• Provides a barrier to contain mass, energy, and fission product release by closure of the containment isolation valves (CIVs) upon containment isolation signal • Provides a sealed containment and thermal conduction for the condensation of steam that provides makeup water to the reactor coolant system (RCS) • Transfers core heat from reactor coolant in containment to the ultimate heat sink (UHS) • Provides safety-related signals	A1	CNTS SSC with the exception of the following: • CIV close and open position sensors: - Containment evacuation system, inboard and outboard - Containment flooding and drain system (CFDS), inboard and outboard - Chemical and volume control system (CVCS) inboard and outboard pressurizer (PZR) spray line - CVCS, inboard and outboard RCS discharge - CVCS, inboard and outboard RCS injection - CVCS, inboard and outboard high-point degasification - Reactor component cooling water system, inboard and outboard return and supply - Main steam system (MSS) and MSS backup • Reactor pressure vessel (RPV) high point degasification solenoid valve close and open position sensors • CVCS discharge air operated valve close and open position sensors • CFDS piping inside containment • Containment air resistance temperature detectors • Piping from systems (containment evacuation system, CFDS, CVCS, condensate and feedwater system, MSS, reactor component cooling water system) CIVs to disconnect flange (outside containment)	Determination by expert panel and informed with input from PRA, deterministic, and other methods of analysis

Table 17.4-1: Design Reliability Assurance Program Structures, Systems, and Components Functions, Categorization, and Categorization Basis (Continued)

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
CNTS (Continued)			
		• CVCS piping (outside containment): - RPV high point degasification solenoid valve to disconnect flange - PZR spray flow check valve to disconnect flange - Injection flow check valve to disconnect flange - Discharge air operated valve to disconnect flange • Containment pressure transducers (wide range) • Feedwater isolation check valves • Feedwater resistance temperature detectors • CNTS top support structure • Containment vessel, control rod drive mechanism support frame • RPV support ledge • Passive autocatalytic recombiner	
• Provide lifting attachment points for the Reactor Building crane for transport of the NuScale Power Module~~Provides backup isolation capability for containment isolation lines that may result in a loss-of-coolant event.~~	B1	• CNTS top support structure~~RPV high point degasification solenoid valve~~ • ~~PZR spray flow check valve~~ • ~~CVCS injection flow check valve~~ • ~~CVCS discharge air operated valve~~ • ~~CVCS piping: (outside containment)~~ - ~~RPV high point degasification CIV to reducer~~ - ~~Reducer to RPV high point degasification solenoid valve~~ - ~~PZR spray CIV to PZR spray flow check valve~~ - ~~Injection CIV to injection flow check valve~~ - ~~Discharge CIV to discharge air operated valve~~	Determination by expert panel and informed with input from PRA, deterministic, and other methods of analysis
Reactor Core System (RXC)			
• Contains fission products and transuranics within the fuel rods to minimize contamination of the reactor coolant • Maintains a coolable geometry under normal operating and design-basis event conditions	A1	• Fuel assembly	Determination by expert panel and informed with input from PRA, deterministic, and other methods of analysis