December 27, 2022

Docket No. 52-050

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

SUBJECT: NuScale Power, LLC Submittal of the NuScale Standard Design Approval Application Part 2 – Final Safety Analysis Report, Chapter 17, “Quality Assurance and Reliability Assurance,” Revision 0

REFERENCES:

1. NuScale letter to NRC, “NuScale Power, LLC Submittal of Planned Standard Design Approval Application Content,” dated February 24, 2020 (ML20055E565)
2. NuScale letter to NRC, “NuScale Power, LLC Requests the NRC staff to conduct a pre-application readiness assessment of the draft, ‘NuScale Standard Design Approval Application (SDAA),’” dated May 25, 2022 (ML22145A460)
3. NRC letter to NuScale, “Preapplication Readiness Assessment Report of the NuScale Power, LLC Standard Design Approval Draft Application,” Office of Nuclear Reactor Regulation dated November 15, 2022 (ML22305A518)
4. NuScale letter to NRC, “NuScale Power, LLC Staged Submittal of Planned Standard Design Approval Application,” dated November 21, 2022 (ML22325A349)

NuScale Power, LLC (NuScale) is pleased to submit Chapter 17 of the Standard Design Approval Application, “Quality Assurance and Reliability Assurance,” Revision 0. This chapter supports Part 2, “Final Safety Analysis Report,” (FSAR) of the NuScale Standard Design Approval Application (SDAA) (Reference 1). NuScale submits the chapter in accordance with requirements of 10 CFR 52 Subpart E, Standard Design Approvals. As described in Reference 4, the enclosure is part of a staged SDAA submittal. NuScale requests NRC review, approval, and granting of standard design approval for the US460 standard plant design.

From July 25, 2022 to October 26, 2022, the NRC performed a pre-application readiness assessment of available portions of the draft NuScale FSAR to determine the FSAR’s readiness for submittal and for subsequent review by NRC staff (References 2 and 3). The NRC staff reviewed draft Chapter 17. NuScale is enclosing information in this submittal that: 1) closes gaps identified between the draft SDAA Chapter 17 and technical content generally expected by the NRC; and 2) resolves identified technical issues that may have adversely impacted acceptance, docketing, or technical review of the application. Enclosure 2 provides NuScale’s responses to Reference 3 for Chapter 17 observations.

Enclosure 1 contains SDAA Part 2 Chapter 17, “Quality Assurance and Reliability Assurance,” Revision 0. Enclosure 2 contains the Readiness Assessment Report for this chapter.

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Mark Shaver at 541-360-0630 or at mshaver@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 27, 2022.

Sincerely,

Carrie Fosaaen
Senior Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Brian Smith, NRC
Michael Dudek, NRC
Getachew Tesfaye, NRC
Bruce Bavol, NRC
David Drucker, NRC

Enclosure 1: SDAA Part 2 Chapter 17, “Quality Assurance and Reliability Assurance,” Revision 0
Enclosure 2: Readiness Assessment Report responses for Chapter 17

Enclosure 1:

SDAA Part 2 Chapter 17, “Quality Assurance and Reliability Assurance,” Revision 0

NuScale US460 Plant
Standard Design Approval Application

Chapter Seventeen
Quality Assurance and Reliability Assurance

Final Safety Analysis Report

Revision 0

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CHAPTER 17 QUALITY ASSURANCE AND RELIABILITY ASSURANCE

17.1 Quality Assurance During the Design Phase

The Quality Assurance Program is addressed in Section 17.5.

17.2 Quality Assurance During the Construction and Operation Phases

This section is not applicable to new plant designs. The Quality Assurance Program is described in Section 17.5.

17.3 Quality Assurance Program Description

The Quality Assurance Program Description is addressed in Section 17.5.

17.4 Reliability Assurance Program

The Reliability Assurance Program (RAP) applies to safety-related and nonsafety-related structures, systems, and components (SSC) that are identified as being risk-significant. The risk significance is determined by Probabilistic Risk Assessment (PRA), deterministic, or other methods of analysis.

The implementation of the RAP provides reasonable assurance that the

- plant is designed, constructed, and operated in a manner that is consistent with the risk insights and key assumptions (e.g., SSC design, reliability, and availability) from the probabilistic, deterministic, and other methods of analysis used to identify and quantify risk.
- risk-significant SSC do not degrade to an unacceptable level of reliability, availability, or condition during plant operations.
- frequency of transients that challenge these SSC is minimized.
- risk-significant SSC function reliably when challenged.

The RAP is implemented in two stages. The first stage, the Design Reliability Assurance Program (D-RAP), encompasses the reliability assurance activities conducted during the plant design and construction phases before initial fuel load. This stage includes implementation of those portions of D-RAP activities that apply to the standard design and to site-specific activities.

The second stage of the RAP is conducted during the operations phases of the plant's 60-year operating life to ensure that the reliability of the SSC within the scope of the RAP is maintained during operations.

COL Item 17.4-1: An applicant that references the NuScale Power Plant US460 standard design will describe the Reliability Assurance Program conducted during the operations phases of the plant's 60-year design life.

17.4.1 Design Reliability Assurance Program Description

Design Reliability Assurance Program implementation includes the following activities:

- develop D-RAP details (i.e., scope, purpose, objectives, framework, and phases for implementation) that are implemented during design activity phases
- perform a preliminary D-RAP process on functions and systems
- establish and apply programmatic controls of the D-RAP
- develop a list of RAP SSC (within the scope of the design activities) using a combination of probabilistic, deterministic, and other methods of analysis used to identify and quantify risk
- implement the appropriate quality assurance controls for design activities for the nonsafety-related Reliability Assurance Program SSC in accordance with NUREG-0800 Section 17.5, Revision 1, Part U, and nonsafety-related SSC quality controls

COL Item 17.4-2: An applicant that references the NuScale Power Plant US460 standard design will identify site-specific structures, systems, and components within the scope of the Reliability Assurance Program.

17.4.2 Programmatic Controls of Design Reliability Assurance Program

Programmatic controls are established and applied to ensure that the risk insights and key assumptions used to identify and quantify risk are consistent with the plant design, and to ensure that the Reliability Assurance Program SSC are identified, maintained, and communicated to the appropriate organizations.

The D-RAP program includes controls associated with D-RAP organization, design control, procedures and instructions, corrective action process, records, and assessment plans. These controls are addressed in Section 17.4.2.1 through Section 17.4.2.6.

17.4.2.1 Organization

The D-RAP organization consists of the vice president of Engineering, D-RAP coordinator, subject matter experts (SMEs), and expert panel members.

The vice president, Engineering has overall responsibility for establishing and maintaining the D-RAP program.

The D-RAP program coordinator assists with preparation for expert panel meetings, works with SMEs to develop system function reports, and prepares the D-RAP summary report and D-RAP list.

System SMEs, with the assistance of the PRA and the Safety Analysis groups, identify, categorize, and evaluate system functions and equipment SSC classifications for review by the expert panel.

The D-RAP expert panel is a select team of personnel with collective experience in safety analysis, licensing, PRA, design engineering, operations, and maintenance processes to review the recommendation for categorization of system functions, the D-RAP summary report, and the D-RAP equipment list. The expert panel composition and responsibilities are discussed in Section 17.4.4.

17.4.2.2 Design Control

Procedures define the process for evaluating design changes to controlled engineering documents to ensure that the impact is considered before a change is approved, and that the affected documents are identified and updated as appropriate. The D-RAP processes the change through the expert panel as applicable.

The design control process also ensures that the list of SSC within the scope of the D-RAP is maintained.

17.4.2.3 Controls for Procedures and Instructions

The procedures and instructions applicable to the D-RAP are developed and controlled in accordance with the applicable provisions of the Quality Assurance Program Description (QAPD) (Section 17.5).

17.4.2.4 Corrective Action Process

The Corrective Action Program is applied to D-RAP activities to identify and resolve issues. The identified issues are entered into the Corrective Action Program for resolution. The Corrective Action Program is included within the QAPD (Section 17.5).

17.4.2.5 Controls for Records

The D-RAP activities are subject to the records requirements described in the QAPD (Section 17.5).

17.4.2.6 Assessment Plans

The D-RAP process is subject to the audit requirements described in the QAPD (Section 17.5).

17.4.3 Methodology for Risk-Informed Categorization of Structures, Systems, and Components

The objective of the SSC Classification Program is to classify the SSCs that comprise the plant in terms of safety, risk, augmenting requirements, seismic class, and quality group. System-level functions are evaluated and classified as to their risk-significance to determine whether associated SSC are part of the D-RAP.

The D-RAP process for SSC risk-significance determination is depicted in Figure 17.4-1.

The methodology for the classification of the SSC as to their risk-significance is discussed in the following sections.

17.4.3.1 Structures, Systems, and Components Classification and Categorization Process

The SSC classification process is described in Section 3.2 and considers both safety and risk. Risk significance is determined by the identification and review of each system function. Each system-level function is evaluated to determine the SSC required to fulfill the function. System functions and the SSC that perform those functions are evaluated for risk-significance based on a consideration of probabilistic, deterministic, and other methods of analysis, including industry operating experience, expert panel reviews, and severe accident evaluations. The SSC risk categorization is determined by the SME and confirmed by expert panel review.

Risk evaluations cover the spectrum of potential events and the range of plant operating modes considered in the PRA (Section 19.1). This evaluation ranges from full power operation to shutdown and anticipated maintenance conditions. Beyond-design-basis accidents resulting in core damage and large releases of radioactivity into containment and the environment are also considered. The evaluation of severe accidents is described in Section 19.2.

NuScale uses an alternative approach to Regulatory Guide 1.200 that is described in Topical Report TR-0515-13952-NP-A, “Risk Significance Determination” (Reference 17.4-1), and Section 19.1 demonstrates applicability to the US460 standard design.

17.4.3.2 Identification of Design Reliability Assurance Program Structures, Systems, and Components

The SSC classification process uses a functional hierarchy concept in which system functions are broken down into components that are required to fulfill the function. The process begins by defining system functions and categorizing them in accordance with their contribution to safety and risk-significance.

The defined standard functions are categorized as:

- A1 (safety-related and risk-significant)
- A2 (safety-related, not risk-significant)
- B1 (nonsafety-related, risk-significant)
- B2 (nonsafety-related, not risk-significant)

The D-RAP structures, systems, and components are those that are required to perform the system functions that are risk-significant (functions categorized as either A1 or B1). As noted in Section 17.4.3.1, the evaluation for risk-significance is based on probabilistic, deterministic, and other methods of analysis, industry operating experience, expert panel reviews, and severe accident evaluations.

Concurrence by the expert panel constitutes the final classification of the SSC. If a downgrade in the safety-significance classification is not deemed necessary due to change in the original PRA information, the original classification is retained for the SSC. The risk-significance classification for safety-related equipment is the default classification unless the PRA determined that the SSC functionalities are not risk-significant.

Table 17.4-1 lists the system functions and associated SSC determined by this process to be risk-significant. The table also provides the basis for the determination.

17.4.3.3 Classification of Regulatory Treatment of Nonsafety Systems Structures, Systems, and Components

The process for evaluating SSC with respect to the Regulatory Treatment of Nonsafety-Systems (RTNSS) Program is described in Section 19.3.

Structures, systems, and components determined to meet the RTNSS criteria are deterministically considered risk-significant in accordance with Nuclear Regulatory Commission guidance, and are categorized as B1.

17.4.4 Expert Panels

The D-RAP expert panel is a select team of experts with collective experience in safety analysis, licensing, PRA, design engineering, operations, and maintenance processes. The expert panel reviews the recommendation for categorization of system functions and the list of SSC that may be risk-significant, RTNSS, or nonsafety-related with augmented requirements, as determined by the SMEs. The expert panel members possess an accredited four-year degree in engineering, science, or other related field with a minimum of 5 years of experience in one or more of the following areas:

- PRA or risk and reliability analysis, including 3 years PRA experience on small modular reactor design
- safety analysis expertise with U.S. nuclear regulatory guidelines
- licensing
- power plant operations, maintenance, previous commercial senior reactor operator license
- design integration or systems engineering
- design engineering (mechanical, electrical, instrumentation and controls, structural, civil)

The chairperson of the expert panel is assigned by the vice president, Engineering. There are seven review team roles within the expert panel (expert panel chair, Operations, Safety Analysis, PRA, Licensing, and two representatives from Design Engineering). The expert panel chair may also serve one other review role simultaneously. A quorum of at least five individuals (including the chair) is required to validate decisions made at the expert panel meetings.

Conclusions of the expert panel meeting are summarized and documented in the system function reports.

The roles of the expert panel in the SSC classification process is illustrated in Figure 17.4-1.

17.4.5 Reliability Assurance Program Structures, Systems, and Components List

The output of the SSC classification process consists of system function reports and equipment lists. The risk-significant SSC from the equipment lists are compiled into the D-RAP list. As the design progresses, the equipment list grows to the final state in which the D-RAP list includes all risk-significant SSC. A D-RAP summary report is prepared to summarize the results and bases of the D-RAP process, including the results of expert panel decisions pertaining to system function categorization.

Table 17.4-1 provides a list of NuScale Power Plant D-RAP system functions and associated SSC that are determined to be risk-significant using the process described in Section 17.4.3.2.

The system function categorization and equipment classifications are reviewed and updated as necessary when a system design is changed, additional components are added to a system, updated PRA data are obtained, an error is found, or further expert panel input is provided.

17.4.6 Determination of Dominant Failure Modes

The PRA addresses dominant failure modes of equipment.

The design process considers the following elements for enhancing the reliability of D-RAP structures, systems, and components:

- Identification of system functions and their risk-significance categorization
- Identification of dominant failure modes for risk-significant system functions
  - The PRA provides dominant failure modes to the expert panel as part of the basis for their determination of risk significant functions.
  - The detailed design phase confirms the dominant failure modes as part of the site-specific PRA analysis after addition of site-specific SSC detail to the PRA models.
- Identification of SSC dominant failure modes that contribute to system function failure
  - The detailed design phase includes site-specific PRA analysis after addition of site-specific SSC detail to the PRA models.
- Identification of activities (equipment performance goals and condition monitoring) that ensure that the SSC failure modes are reduced or kept to an acceptably low probability.
  - The detailed design process includes this task. Implementation of the Maintenance Rule program in 10 CFR 50.65 includes condition monitoring.

For some components, operating history is available that defines the dominant failure modes and their likely causes for consideration during the detailed design process. For those SSC with insufficient operating history to identify critical failure modes, an analytical approach is necessary.

Analytical methods include PRA importance analysis, root cause analysis, fault trees, and failure modes and effects analysis. The detailed design phase includes these methods to identify dominant failure modes, such as single latent failures not detected by routine monitoring, common cause failures, or failures that could cascade into significant functional failures.

17.4.7 Quality Assurance Applicable to Reliability Assurance Program Activities

The quality assurance controls applicable to the D-RAP process during the design activity phase are included within the QAPD (Section 17.5) and are consistent with the requirements in NUREG-0800, Section 17.5, Quality Assurance Program Description.

COL Item 17.4-3: An applicant that references the NuScale Power Plant US460 standard design will identify the quality assurance controls for the Reliability Assurance Program structures, systems, and components during site-specific design, procurement, fabrication, construction, and preoperational testing activities.

17.4.8 Inspections, Tests, Analyses, and Acceptance Criteria for Design Reliability Assurance Program

The methodology for selection of Inspections, Tests, Analyses, and Acceptance Criteria for SSC is described in Section 14.3.

17.4.9 Reference

17.4-1 NuScale Power, LLC, “Risk Significance Determination,” TR-0515-13952-NP-A, Revision 0.

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 <u>with the exception</u> 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 PRA and concurrence by the expert panel as being needed for maintaining containment and reactor coolant pressure boundary (RCPB) integrity, removing fuel assembly heat, and reactivity control

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	
• Provides backup isolation capability for containment isolation lines that may result in a loss-of-coolant event.	B1	• 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 PRA and concurrence by the expert panel as being needed for maintaining containment integrity
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 PRA and concurrence by the expert panel as being needed for radioactivity control and removing fuel assembly heat

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
Reactor Coolant System			
• Removes heat to ensure core and fuel thermal design limits are not exceeded • Provides instrument information signals for module protection system (MPS) actuation	A1	RCS SSC with the exception of the following: • Reactor safety valves • Reactor safety valve position indicator • PZR vapor temperature element • PZR control cabinet • PZR heater power cabling from MPS breaker to PZR heaters • PZR liquid temperature element • PZR heater power cabling from low voltage AC electrical distribution system breaker to MPS breaker • PZR heaters • RCS injection and discharge lines • PZR spray supply line • RPV PZR high point degasification line	Determination by PRA and concurrence by the expert panel as being needed for removing fuel assembly heat, maintaining containment and RCPB integrity, and reactivity control
Emergency Core Cooling System			
• Depressurizes the RPV to allow recirculated coolant from the containment to enter the RPV for the removal of core heat • Provides recirculated coolant from containment to the RPV for the removal of core heat • Opens the reactor vent valves or reactor recirculation valves when power is removed from their respective trip valves	A1	• Reactor vent valve (RVV) • Reactor recirculation valve (RRV) • RVV trip valve • RRV trip valve • RVV venturi • RRV venturi	Determination by PRA and concurrence by the expert panel as being needed for removing fuel assembly heat
Decay Heat Removal System (DHRS)			
• Provides MPS actuation instrument information signals.	A1	• Steam generator steam pressure instrumentation (4 per side)	Determination by PRA and concurrence by the expert panel as being needed for maintaining containment and RCPB integrity, removing fuel assembly heat, and reactivity control

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
Ultimate Heat Sink (UHS)			
• Acts as heat sink to remove heat from the NuScale Power Modules (NPM) and fuel assemblies • Removes decay heat from the spent fuel via direct water contact with the spent fuel assemblies • Provides iodine scrubbing for contents of the reactor pool, refueling pool, and spent fuel pool via the surrounding water • Provides borated water for reactivity control during refueling	A1	• UHS Pool	Determination by PRA and concurrence by the expert panel as being needed for removing fuel assembly heat, power generation, human habitability, radioactivity control, and reactivity control
Module Protection System			
• Removes electrical power from the PZR heaters on a PZR heat trip actuation signal • Removes electrical power from the trip solenoids of the RVV on an emergency core cooling system actuation signal • Removes electrical power from the trip solenoids of the RRV on an emergency core cooling system actuation signal • Removes electrical power from the trip solenoids of the DHRS actuation valves on a DHRS actuation signal • Removes electrical power from the trip solenoids of the MSS isolation valves, MSS isolation bypass valves and feedwater isolation valves on a DHRS actuation signal • Removes electrical power from the trip solenoids of the following containment isolation valves on a containment system isolation actuation signal - Condensate and feedwater system isolation valves - MSS isolation valves - MSS isolation bypass valves - Containment evacuation system containment isolation valves - Reactor component cooling water system inlet and outlet containment isolation valves - CFDS containment isolation valves - CVCS containment isolation valves	A1	Module protection system SSC with the exception of the following: • Separation Groups A, B, C, D: - Monitoring and indication bus - communication modules • Separation Groups B, C: - Safety function modules for post-accident monitoring • Division I and Division II: - Reactor trip system monitoring and indication bus - communication modules - Engineered safety features actuation system monitoring and indication bus - communication modules - Gateway communication modules - Safety function modules for gateways - Hardwired modules for gateways - Direct current (DC) converters for gateways - Equipment interface modules for de-energizing cabinets and neutron monitoring system-flood - Maintenance workstations • Actuation and priority logic for de-energizing cabinets and neutron monitoring system-flood	Determination by PRA and concurrence by the expert panel as being needed for maintaining containment and RCPB integrity, removing fuel assembly heat, power generation, reactivity control, and emergency response

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
MPS (Continued)			
• Removes electrical power from the trip solenoids of the following containment isolation valves on a chemical and volume control isolation actuation signal: - RCS injection containment isolation valves - RCS discharge containment isolation valves - PZR spray containment isolation valves - RPV high point degasification containment isolation valves • Removes electrical power from the trip solenoids of the demineralized water supply isolation valves on a demineralized water system isolation actuation signal • Removes electrical power to the control rod drive system for a reactor trip • Provides power to the CNTS sensors • Provides power to the MSS pressure sensors • Provides power to the RCS sensors • Removes electrical power from the trip solenoids of the MSS isolation valves, MSS isolation bypass valves, and feedwater isolation valves on a secondary system isolation actuation signal • Removes electrical power to the trip solenoids of the secondary MSS isolation valves and secondary MSS isolation valve bypass valves on a secondary system isolation actuation signal • Removes electrical power from the trip solenoids of the secondary MSS isolation valves and secondary MSS isolation valve bypass valves on a DHRS actuation signal • Removes electrical power from the trip solenoids of the feedwater regulating valves on a DHRS actuation signal • Removes electrical power from the trip solenoids of the feedwater regulating valves on a secondary system isolation actuation signal			

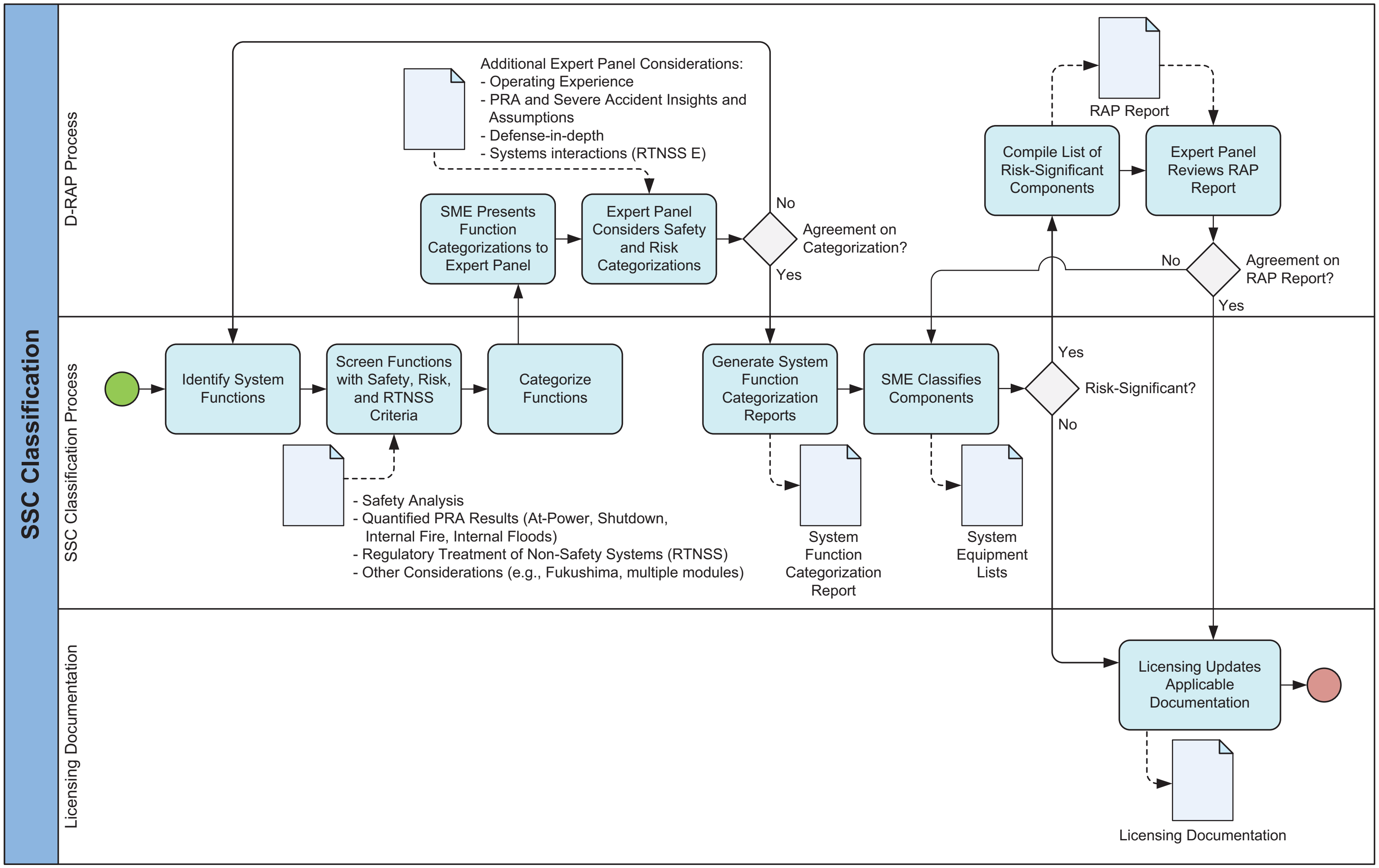
Table 17.4-1: Design Reliability Assurance Program Structures, Systems, and Components Functions, Categorization, and Categorization Basis (Continued)

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
Neutron Monitoring System			
• Provides neutron flux measurement information for startup, normal operations, shutdown, reactor trips, operating bypasses, and actuations • Provides a positioning mechanism to maintain the neutron detector assemblies in a fixed position relative to the containment vessel during normal operations, anticipated operational occurrences, and design-basis events	A1	• Ex-core neutron detectors • Ex-core signal conditioning and processing equipment • Ex-core Separation Groups A, B, C, and D - power isolation, conversion and monitoring devices • Ex-core DC and high voltage power supplies Separation Groups A, B, C, and D • Instrumentation cabinet Separation Groups A, B, C, and D • Positioning support mechanism Separation Groups A, B, C, and D • Ex-core DC-DC converter #1 and #2 Separation Groups A, B, C, and D	Determination by PRA and concurrence by the expert panel as being needed for reactivity control and emergency response
Reactor Building			
• Houses safety-related, risk significant equipment and facilities pertinent to the operation and support of the reactor module(s) • Provides anchorage and support for safety-related, risk significant equipment and facilities pertinent to the operation and support of the reactor module(s) • Protects safety-related, risk significant equipment and facilities from natural phenomena and externally generated missiles • Protects safety-related, risk significant equipment and facilities from internal events and internal generated missiles • Houses and protects spent fuel	A1	• Reactor Building	Determination by PRA and concurrence by the expert panel as being needed for removing fuel assembly heat, maintaining containment and RCPB integrity, reactivity control, radioactivity control, protection of plant assets, plant security, human habitability, and emergency response
• Houses nonsafety-related risk significant equipment and facilities pertinent to the operation and support of the reactor module(s) • Provides anchorage and support for nonsafety-related risk significant equipment and facilities pertinent to the operation and support of the reactor module(s) • Protects nonsafety-related risk significant equipment and facilities from natural phenomena	B1	• Reactor Building	Determination by PRA and concurrence by the expert panel as being needed for protection of plant assets

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
Reactor Building Components			
• Vents air pressures internal to the Reactor Building that result from a high energy line break • Provides safety-related anchorage and structural support to the NPM	A1	• Over-pressurization vents • NPM supports • Steam gallery blow off panels • CVCS high energy line break blow off panels	Determination by PRA and concurrence by the expert panel as being needed for human habitability, protection of plant assets, and power generation
• Provides nonsafety-related anchorage and structural support to handling equipment	B1	• Reactor Building crane runway rail support	Determination by PRA and concurrence by the expert panel as being needed for power generation
Reactor Building Crane			
• Provides structural support and movement to the NPM while moving from refueling, inspection and operating bay • Limits motion of an NPM or spent fuel cask containing nuclear fuel, to within predefined safe load paths and outside of exclusion zones	B1	• Reactor Building crane • Reactor Building control cabinet • Reactor Building power cabinet	Determination by PRA and concurrence by the expert panel as being needed for power generation
Control Building			
• Houses safety-related, risk significant equipment and facilities pertinent to the operation and support of the reactor module(s) • Provides anchorage and support for safety-related, risk significant equipment and facilities pertinent to the operation and support of the reactor module(s) • Protects safety-related, risk significant equipment and facilities from natural phenomena and externally generated missiles	A1	• Control Building	Determination by PRA and concurrence by the expert panel as being needed for power generation, removing fuel assembly heat, maintaining containment and RCPB integrity, and reactivity control
• Protects operators from natural phenomena and externally generated missiles	B1	• Control Building	Determination by PRA and concurrence by the expert panel as being needed for human habitability

Figure 17.4-1: Structures, Systems, and Components Classification Process Flow Chart



17.5 Quality Assurance Program Description

The Quality Assurance Program Description (QAPD) for the NuScale Power Plant US460 standard design is provided in the topical report, “NuScale Power, LLC Quality Assurance Program Description” (Reference 17.5-1).

17.5.1 Reference

17.5-1 NuScale Power, LLC, “NuScale Power, LLC Quality Assurance Program Description,” MN-122626, Revision 0.

17.6 Maintenance Rule

This section addresses an operational program that is the responsibility of an applicant and is not applicable to new plant designs.

Enclosure 2:

Readiness Assessment Report responses for Chapter 17

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

The table below provides the NuScale responses to each of the Nuclear Regulatory Commission readiness assessment observations on draft Chapter 17, “Quality and Reliability Assurance” of the Standard Design Approval Application.

Section	Observation	Response
17.4	As discussed in SRP Section 17.4 “Reliability Assurance Program (RAP),” Revision 1, the RAP applies to those SSCs, both safety-related and nonsafety-related, identified as risk significant (or significant contributors to plant safety). Section 17.4 of the SDA references the staff approved TR-0515-13952-NP-A “Risk Significance Determination,” Revision 0. To use a TR, an applicant needs to demonstrate that all conditions of use are either met or justify deviations. Therefore, the SDA needs to include a discussion on how each condition of use for the TR is met, including Condition 2, which identifies the consideration of PRA uncertainties and sensitivity assessment in risk significance determination. This discussion can be either in SDA Section 17.4 or elsewhere with a pointer to the location in SDA Section 17.4. The staff provided a similar comment during the June 2022 public meeting on the use of this TR for the SDA application (ML22195A049).	Final Safety Analysis Report Section 19.1.4 discusses risk significance determination and conditions of applicability for topical report TR-0515-13952-NP-A. A pointer to Section 19.1 was added in Section 17.4.