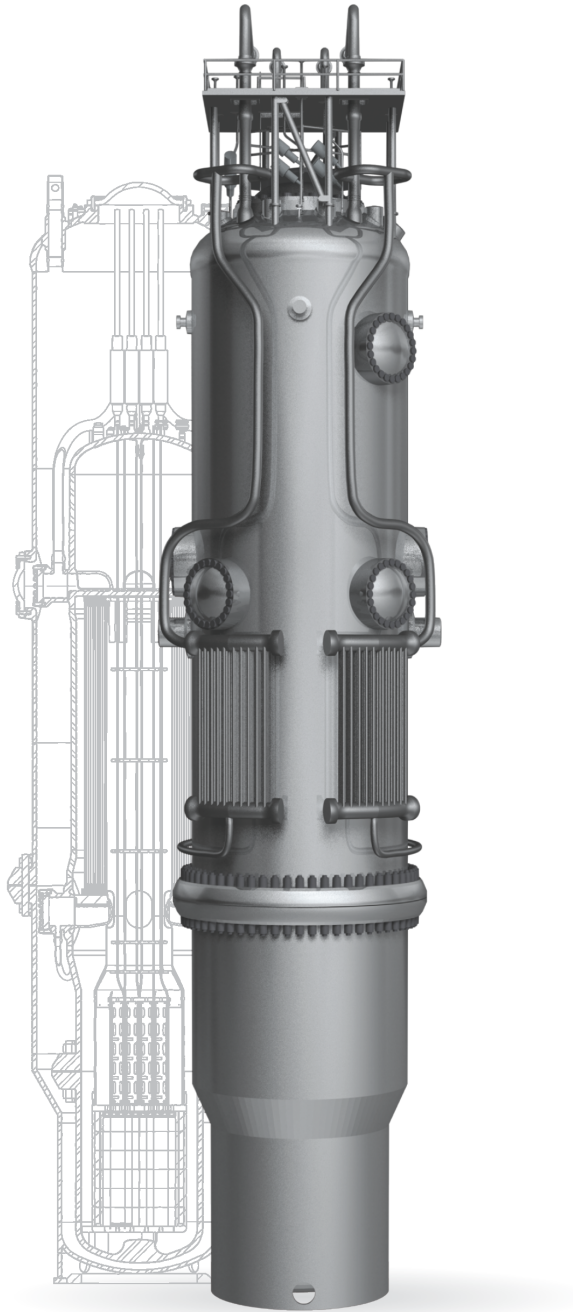




NuScale Standard Plant
Design Certification Application

Chapter Seventeen
**Quality Assurance and
Reliability Assurance**

PART 2 - TIER 2

Revision 1
March 2018

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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 (QAPD) 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.

This section describes the design reliability assurance program (D-RAP) that applies to activities conducted during the NuScale Power Plant design and construction phases before the initial fuel load.

17.4.1 Design Reliability Assurance Program Description

The implementation of the RAP provides reasonable assurance that the

- plant is designed, constructed, and operated in a manner that is consistent with the assumptions and risk insights for the risk-significant SSC.
- risk-significant SSC do not degrade to an unacceptable level during plant operations.
- frequency of transients that challenge 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 or D-RAP, encompasses the reliability assurance activities that occur during detailed design of the plant before initial fuel load. This includes implementation of those portions of D-RAP activities that apply to the standard design and implementation of those portions of the D-RAP activities that apply to site-specific activities.

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

COL Item 17.4-1: A COL applicant that references the NuScale Power Plant design certification will describe the reliability assurance program conducted during the operations phases of the plant's life.

The D-RAP is implemented in two phases during the design stage. The first phase occurs during development of the standard plant design. The second phase occurs during development of the site-specific design. D-RAP implementation during development of the standard plant design includes the following activities:

- develop D-RAP details (i.e., scope, purpose, objectives, framework, and phases for implementation) that are implemented during design certification and site-specific implementation phases
- perform a preliminary D-RAP process on functions and systems
- establish and implement programmatic controls of the D-RAP
- develop a list of RAP SSC (within the scope of the standard plant design) using a combination of probabilistic, deterministic, and other methods of analysis used to identify and quantify risk

- establish and implement the appropriate quality assurance (QA) controls for standard plant design activities for the nonsafety-related RAP SSC in accordance with nonsafety-related SSC quality controls

COL Item 17.4-2: A COL applicant that references the NuScale Power Plant design certification 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 RAP 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 audits. 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 of Engineering has the overall responsibility for the management of the programmatic controls during the standard plant design phase of the D-RAP. The D-RAP Coordinator implements the D-RAP procedures, and is responsible for the formulation, implementation, execution, and output of the D-RAP process as well as the selection of the D-RAP Expert Panel members. The D-RAP Coordinator is the chairman of the expert panel.

The D-RAP expert panel is a select team of personnel with collective experience in safety, licensing, engineering, operations, and maintenance processes to review, evaluate, and confirm the list of risk-significant SSC, i.e., the D-RAP list. The expert panel composition and responsibilities are discussed in Section 17.4.4.

The SMEs identify, classify, and evaluate SSC before the review by the expert panel for inclusion into the D-RAP list.

17.4.2.2 Design Control

NuScale 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 Audit 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 SSC

The SSC are evaluated and classified as to their risk-significance to determine whether they are part of the D-RAP. The scope of the D-RAP program includes those SSC that are determined to be safety-related and risk-significant, and nonsafety-related and risk-significant.

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 subsections.

17.4.3.1 SSC 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 plant 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 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.

Standard Review Plan Section 19.0 states that the term "significant," in the context of PRA results and insights, is intended to be consistent with the definition provided in Regulatory Guide (RG) 1.200. In RG 1.200, "significant" is defined in terms of relative risk criteria, and defines a basic event or contributor as significant if its risk achievement worth (RAW) is greater than 2.0 or its Fussell-Vesely (FV) importance is greater than 0.005.

Use of the aforementioned relative risk criteria of RG 1.200 would artificially raise the relative importance of SSC that do not drive risk in the NuScale design. Therefore, an alternative approach to determining risk-significance is needed for the NuScale design because the traditional relative importance measures are insensitive to the global improvements in safety associated with the lower risk profile of the NuScale design. NuScale is implementing an approach which employs an absolute evaluation of the RAW importance measure. The alternative approach for determining risk-significance is described in NuScale Topical Report TR-0515-13952, Risk Significance Determination (Reference 17.4-1).

This approach is implemented in a manner that is consistent with the guidelines in RG 1.174 that provide the risk-informed integrated decision-making framework for making changes to a licensee's approved licensing basis. This method relies on a combination of absolute and relative risk-significant thresholds that are based on the concepts provided by the NRC in RG 1.174 and consider the much reduced risk metrics associated with the NuScale design. The NuScale approach directly addresses the ratio limitations of traditional importance measures. It also includes a backstop using the FV importance measure; this metric will capture measurable contributors to risk regardless of the overall level of core damage frequency or large release frequency. The NuScale criteria balance measures to ensure maintaining margins to NRC safety goals with those of maintaining the current risk profile.

17.4.3.2 Identification of D-RAP SSC

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 in a standard format 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) or B2 (nonsafety-related, not risk-significant). The D-RAP SSC are those that are required to perform the system functions that are risk-significant, i.e., the functions categorized as A1 and 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 specifically determined that the SSC functionality is not risk-significant.

Table 17.4-1 provides a listing of the SSC determined by this process to be D-RAP SSC. The table also provides the basis for the determination. The classification process is subject to the design control process (see Section 17.4.2.2).

17.4.3.3 Classification of Risk-Related and Regulatory Treatment of Non-Safety Systems SSC

The process for evaluating SSC with respect to the regulatory treatment of nonsafety systems (RTNSS) program is described in Section 19.3.

SSC determined to meet the RTNSS criteria are deterministically considered risk-significant in accordance with NRC guidance, and are categorized as B1.

17.4.4 Expert Panels

The D-RAP expert panel is a select team with collective experience in safety, licensing, engineering, operations, and maintenance processes. The expert panel members review, evaluate, and confirm the list of risk-significant SSC, i.e., the D-RAP list, 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
- licensing
- power plant operations, maintenance, previous commercial senior reactor operator license
- design integration or system engineering
- design engineering (mechanical, electrical, I&C, structural, civil)

The D-RAP coordinator serves as the chairman of the expert panel. The expert panel members are trained on the applicable D-RAP program procedures. A minimum of 5 individuals (the SME and panel members from operations, safety analysis, PRA, and design engineering) are required for a quorum to validate decisions made at the expert panel meetings.

Conclusions of the expert panel meetings are documented.

The roles of the expert panel in the review, identification, and categorization of risk-significant SSC is illustrated in Figure 17.4-1.

17.4.5 Reliability Assurance Program SSC List

The output of the D-RAP process during the design phase is a list of risk-significant SSC (D-RAP list). Maintenance of the list is performed by evaluating SSC design changes and changes in PRA results, defense-in-depth principles, and pertinent operating experience. Changes to the D-RAP list are reviewed by the expert panel.

Table 17.4-1 provides a list of NuScale Power Plant D-RAP SSC that were determined to be risk-significant using the process described in Section 17.4.3.2. A complete list of NuScale Power Plant SSC, including designation of their classification, seismic category, and quality group is provided in Table 3.2-1.

Changes to the D-RAP SSC list are controlled by D-RAP procedures.

17.4.6 Determination of Dominant Failure Modes

The relevant component failure databases are used by the PRA in addressing the equipment failure modes. Refer to Section 19.1.4.1 for information on the use of component failure rate data.

To enhance the reliability of the D-RAP SSC, the following elements are considered during the design process:

- system functions and their risk-significance categorization
- dominant failure mode for risk-significant system functions
- SSC dominant failure mode that contributes to the system function failure
- activities (equipment performance goals and condition monitoring) that ensure that the SSC failure modes are reduced or kept to an acceptably low probability

The determination of dominant failure modes of risk-significant SSC considers historical information, analytical models, and existing requirements. Many components have a substantial operating history available that define the significant failure modes and their likely causes. For SSC that have insufficient operating history to identify critical failure modes, analytical methods are utilized.

Analytical methods for identifying dominant failure modes include PRA importance analysis, root cause analysis, fault trees, and failure modes and effects analysis (FMEA) in accordance with applicable provisions of NUREG-0492. These methods are considered during the detailed design phase to identify dominant failure modes, such as single latent failures that are not detected by routine monitoring, and common cause failures or failures that could cascade into more significant functional failures.

17.4.7 Quality Assurance Applicable to RAP Activities

The QA controls applicable to the D-RAP process during the standard plant design phase are included within the QAPD (Section 17.5). D-RAP SSC that are both safety-related and risk-significant are subject to the full 10 CFR 50 Appendix B QA program. QA controls for nonsafety-related risk-significant SSC are consistent with the controls described for nonsafety-related RAP SSC in NUREG-0800, Section 17.5.

COL Item 17.4-3: A COL applicant that references the NuScale Power Plant design certification 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 ITAAC for Design Reliability Assurance Program

The determination and selection of inspections, tests, analyses, and acceptance criteria (ITAAC) for SSC is described in Section 14.3.

17.4.9 Reference

- 17.4-1 NuScale Power Licensing Topical Report, LLC "Risk Significance Determination," TR-0515-13952, Revision 0.

Table 17.4-1: D-RAP SSC Functions, Categorization, and Categorization Basis

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
Containment System (CNTS)			
- Supports reactor building by providing a barrier to contain mass, energy, and fission product release from a degradation of the reactor coolant pressure boundary (RCPB) - Supports reactor building by providing a barrier to contain mass, energy, and fission product release by closure of the containment isolation valves (CIVs) upon containment isolation signal - Supports emergency core cooling system (ECCS) operations by providing a sealed containment and thermal conduction for the condensation of steam that provides makeup water to the reactor coolant system (RCS) - Supports control rod drive system (CRDS) by providing structural support for the control rod drive mechanisms - Supports RCS by providing structural support for the reactor pressure vessel (RPV) - Supports RCS by transferring core heat from reactor coolant in containment to the ultimate heat sink (UHS) - Supports ECCS by providing structural support of the trip and reset valves for the ECCS reactor vent and recirculation valves - Supports neutron monitoring system (NMS) by providing structural support for the ex-core detectors - Supports ECCS by providing electrical penetration assemblies for reactor instrumentation cables through containment vessel (CNV) - Supports RCS by closing the CIVs for pressurizer spray, chemical and volume control (CVC) makeup, CVC letdown, and RPV high point degas when actuated by module protection system (MPS) for RCS Isolation - Supports MPS by providing MPS actuation instrument information signals through CNV	A1	All CNTS SSC <u>with the exception</u> of the following: - CIV close and open position sensors: - Containment evacuation system (CES), inboard and outboard - Containment flooding and drain system (CFDS), inboard and outboard - Chemical and volume control system (CVCS) inboard and outboard pressurizer spray line - CVCS, inboard and outboard RCS discharge - CVCS, inboard and outboard RCS injection - CVCS, inboard and outboard reactor pressure vessel (RPV) high-point degasification - Feedwater system (FWS), supply to steam generators and decay heat removal (DHR) heat exchangers feedwater isolation valves - Reactor component cooling water system (RCCWS), inboard and outboard return and supply - Steam generator system (SGS), steam supply CIV/ main steam isolation valves (MSIVs) and CIV/MSIV bypasses - CFDS piping inside containment - Containment air temperature detectors (RTDs) - Piping from systems (CES, CFDS, CVCS, FWS, MSS, RCCWS) CIVs to disconnect flange (outside containment) - Containment pressure transducers (wide range) - Feedwater temperature transducers (RTDs) - Hydraulic skid for valve reset	Determination by probabilistic risk assessment (PRA) and concurrence by the expert panel as being needed for maintaining containment and RCPB integrity, removing fuel assembly heat, reactivity control, and emergency response
Supports reactor building crane (RBC) by providing lifting attachment points that the RBC can connect to, so that the module can be lifted	B1	- NuScale Power Module lifting lugs and top auxiliary mechanical access structure diagonal lifting braces - Top auxiliary mechanical access structure	Determination by PRA and concurrence by the expert panel as being needed for maintaining containment integrity

Table 17.4-1: D-RAP SSC Functions, Categorization, and Categorization Basis (Continued)

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
Steam Generator System (SGS)			
Supports RCS by supplying part of the RCPB	A1	- Steam generator tubes - Steam generator tube supports - Feed plenums - Steam plenums	Determination by PRA and concurrence by the expert panel as being needed for maintaining RCPB integrity
Reactor Core System (RXC)			
- Supports control rod assembly (CRA) by providing control rod guide tubes to receive and align the CRA - Supports RCS by containing fission products and transuranics within the fuel rods to minimize contamination of the reactor coolant - Supports RCS by maintaining a coolable geometry	A1	Fuel assembly	Determination by PRA and concurrence by the expert panel as being needed for reactivity control, radioactivity control, and removing fuel assembly heat
Control Rod Drive System (CRDS)			
Supports CRA by releasing control rod during a reactor trip	A1	- Control rod drive shafts - Control rod drive latch mechanism	Determination by PRA and concurrence by the expert panel as being needed for reactivity control
Reactor Coolant System (RCS)			
- Supports RXC by removing heat to ensure core thermal design limits are not exceeded - Supports CNT by supplying the RCPB and a fission product boundary via the RPV and other appurtenances - Supports MPS by providing instrument information signals for MPS actuation - Supports CRDS by the RPV and the reactor vessel internals supporting and aligning the control rods - Supports ECCS by providing mechanical support for the ECCS valves - Supports in-core instrumentation (ICI) by providing structural support of the ICI guide tubes - Supports RXC by the reactor vessel internals providing mechanical support to orient, position, and seat the fuel assemblies - Supports SGS by providing physical support for the steam generator tube supports and for the integral steam and feed plenums - Supports RXC by containing soluble neutron poison	A1	All RCS SSC <u>with the exception</u> of the following: - Wide range RCS pressure element - Wide range RCS cold leg temperature element - Reactor safety valve position indicator - Pressurizer vapor temperature element - Pressurizer control cabinet - Pressurizer heater power cabling from MPS breaker to pressurizer heaters - Pressurizer liquid temperature element - Narrow range RCS cold leg temperature element - Pressurizer heater power cabling from low voltage AC electrical distribution system breaker to MPS breaker	Determination by PRA and concurrence by the expert panel as being needed for removing fuel assembly heat, maintaining containment and RCPB integrity, radioactivity control, and reactivity control

Table 17.4-1: D-RAP SSC Functions, Categorization, and Categorization Basis (Continued)

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
Emergency Core Cooling System (ECCS)			
- Supports RCS by providing a portion of the RCPB for maintaining RCPB integrity - Supports RCS by providing recirculated coolant from containment to RPV for the removal of core heat - Supports CNTS by providing a portion of the containment boundary for maintaining containment integrity - Supports RCS by opening ECCS reactor vent valves and reactor recirculation valves when their respective trip valve is actuated by MPS	A1	- Reactor vent valve (RVV) - Reactor recirculation valve (RRV) - RVV trip valve - RRV trip valve - Reset valve	Determination by PRA and concurrence by the expert panel as being needed for maintaining containment and RCPB integrity, and removing fuel assembly heat
Decay Heat Removal System (DHRS)			
Supports by providing 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, and reactivity control
Ultimate Heat Sink (UHS)			
- Supports CNTS by providing heat removal via direct water contact with the containment vessel for the removal of core heat - Supports spent fuel storage system (SFSS) by providing the removal of decay heat from spent fuel via direct water contact with spent fuel assemblies - Supports SFSS by providing radiation shielding and iodine scrubbing for spent fuel via water surrounding the components - Supports DHRS by accepting heat from DHR heat exchanger - Supports RXC by direct water contact with fuel assemblies during refueling to remove decay heat - Supports RXC by providing borated water for reactivity control during refueling - Supports RXC by providing radiation shielding and iodine scrubbing for fuel assemblies via the water surrounding the components	A1	UHS Pool	Determination by PRA and concurrence by the expert panel as being needed for removing fuel assembly heat, radioactivity control, and reactivity control

Table 17.4-1: D-RAP SSC Functions, Categorization, and Categorization Basis (Continued)

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
Module Protection System (MPS)			
- Supports low voltage AC electrical distribution system (ELVS) by removing electrical power to the pressurizer on a pressurizer heater trip actuation signal - Supports ECCS by removing electrical power to the trip solenoids of the RVVs on an ECCS actuation signal - Supports ECCS by removing electrical power to the trip solenoids of the RRVs on an ECCS actuation signal - Supports DHRS by removing electrical power to the trip solenoids of the DHR actuation valves on a DHRS actuation signal - Supports CNT by removing electrical power to the trip solenoids of the main steam isolation valves, main steam bypass isolation valves and feedwater isolation valves on a DHRS actuation signal	A1	All MPS SSC <u>with the exception</u> of the following: - Division I and Division II engineered safety features actuation system (ESFAS) - Equipment interface modules for secondary MSIVs, secondary MSIV bypass isolation valves and feedwater regulating valves for containment isolation and DHRS actuation - Division I and Division II manual low temperature overpressure protection (LTOP) actuation switch - Separation Groups A, B, C, and D: - Safety function module and associated maintenance switch for LTOP function	Determination by PRA and concurrence by the expert panel as being needed for maintaining containment and RCPB integrity, removing fuel assembly heat, reactivity control, and emergency response

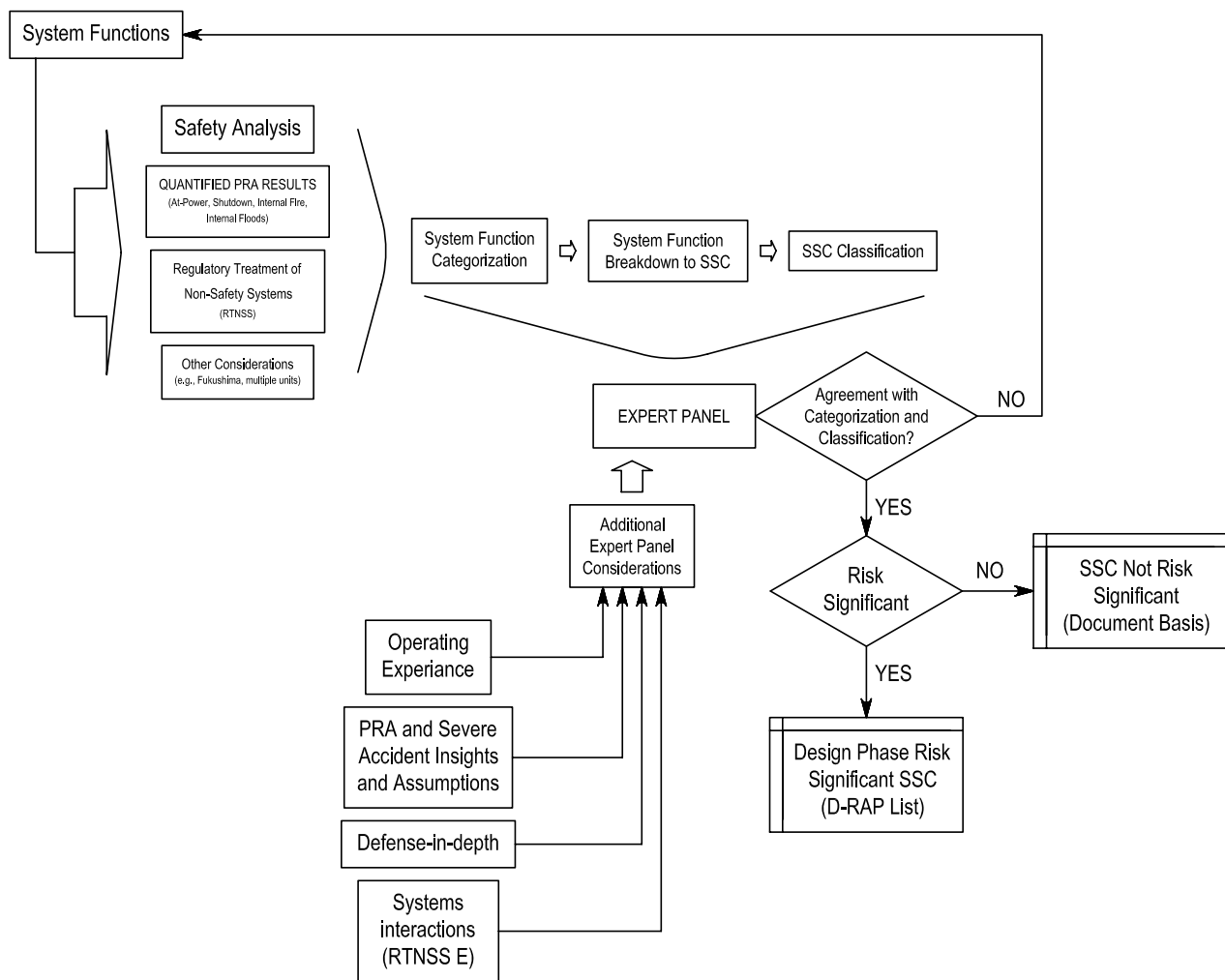
Table 17.4-1: D-RAP SSC Functions, Categorization, and Categorization Basis (Continued)

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
- Supports CNT by removing electrical power to the trip solenoids of the following CIVs on a containment system isolation actuation signal: - RCS injection CIVs - RCS discharge CIVs - Pressurizer spray CIVs - RPV high point degasification CIVs - Feedwater isolation valves - Main steam isolation valves - Main steam bypass isolation valves - Containment evacuation isolation valves - Reactor component cooling water inlet and outlet CIVs - Containment flooding and drain CIVs - Supports CNT by removing electrical power to the trip solenoids of the following CIVs on a CVC isolation actuation signal: - RCS injection CIVs - RCS discharge CIVs - Pressurizer spray CIVs - RPV high point degasification CIVs - Supports CVC by removing electrical power to the trip solenoids of the demineralized water system isolation valves on a DWS isolation actuation signal - Supports non-safety DC electrical and AC distribution system by removing electrical power to the CRDS for a reactor trip - Supports CNT by providing power to sensors - Supports DHR by providing power to main steam pressure sensors - Supports RCS by providing power to sensors		24-hour timers for post-accident monitoring-only mode - Division I and Division II: - ESFAS - equipment interface module for loss of AC voltage function - ESFAS monitoring and indication bus - communication module - MPS gateway - Reactor trip system monitoring and indication bus - communication module - Separation Groups A, B, C, and D: - Monitoring and indication bus - communication module - Separation Groups B and C - safety function modules for post-accident monitoring (PAM) indication functions - Separation Group A - safety function module: - Feedwater indication and control - Leak detection into containment - Separation Groups B and C - safety function module for PAM indication functions - Separation Group D - safety function module: - Leak detection into containment - Division I and II maintenance workstations	
Neutron Monitoring System (NMS)			
Supports MPS by providing neutron flux data for various reactor trips, operating bypasses, and actuations	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	Determination by PRA and concurrence by the expert panel as being needed for reactivity control and emergency response

Table 17.4-1: D-RAP SSC Functions, Categorization, and Categorization Basis (Continued)

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
Reactor Building (RXB)			
• Supports CNTS by housing and providing structural support • Supports CVCS by housing, allowing access, and providing structural support • Supports UHS by housing and providing structural support • Supports MPS by housing and providing structural support • Supports NMS by housing and providing structural support	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, and emergency response
Reactor Building Crane (RBC)			
• Supports NuScale Power Module by providing structural support and mobility while moving from refueling, inspection and operating bay	B1	• Reactor building crane • Module lifting adapter	Determination by PRA and concurrence by the expert panel as being needed for containment integrity
Control Building (CRB)			
• Supports the MPS by housing and providing structural support	A1	• CRB structure at elevation 120'-0" and below	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

Figure 17.4-1: NuScale D-RAP Process for SSC Risk Significance Determination



17.5 Quality Assurance Program Description

The Quality Assurance Program Description (QAPD) for the standard design of the NuScale Power Plant is provided in the topical report, "NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant" (Reference 17.5-1).

COL Item 17.5-1: A COL applicant that references the NuScale Power Plant design certification will describe the quality assurance program applicable to site-specific design activities and to the construction and operations phases.

17.5.1 References

- 17.5-1 NuScale Power, LLC, "NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant," NP-TR-1010-859-NP-A, Revision 3.

17.6 Maintenance Rule

COL Item 17.6-1: A COL applicant that references the NuScale Power Plant design certification will describe the program for monitoring the effectiveness of maintenance required by 10 CFR 50.65.