Preliminary Structure, System, and Component Categorization Report

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Nonproprietary

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1.0 Introduction

The U.S. Nuclear Regulatory Commission (NRC) has established a framework for performing a risk-informed, integrated review for integral pressurized-water reactor (iPWR) applications. This framework is provided in SECY-11-0024 (Reference 5.1) and more recently in draft guidance for new applicants in the Introduction to the Standard Review Plan (SRP), Part 2 (Reference 5.2).

As described in the draft SRP Introduction Part 2, design-specific review plans are developed to schedule pre-application and post-application activities including development of a design specific review standard (DSRS). Each DSRS section will reflect differences in function or acceptance criteria for structures, systems, and components (SSCs) that differ from the SRP. Development of DSRS sections will take into account the categorization of SSCs in terms of safety and risk. Applicable programmatic requirements (e.g., quality, administrative control and testing) will be included in a DSRS section based on the categorization of SSCs. The NRC will assess and verify the applicant's categorization of SSCs based on design information and probabilistic risk assessment (PRA) information made available during pre-application.

Consistent with the SRP Introduction, Part 2, NuScale SSCs are categorized into one of four categories depending on whether the SSC is safety-related and/or risk-significant (see Figure 2-1).

An SSC is classified as safety-related if it is relied on to remain functional during and following design-basis events to assure that the three criteria defined in 10 CFR 50.2 (Reference 5.3) are met. An SSC is categorized as risk-significant if it meets risk criteria defined by NRC guidance (References 5.4 and 5.5). Risk-significant SSCs may be safety-related or nonsafety-related. An SSC that is determined to be risk-significant is included in the NuScale design reliability assurance program (D-RAP) list.

Regulatory treatment of nonsafety systems (RTNSS) applies to SSCs that are nonsafety-related and are selected in accordance with the RTNSS criteria defined in the recently issued draft SRP, Section 19.3 (Reference 5.6).

1.1 Purpose

This report provides a preliminary list of NuScale systems and functions categorized in terms of safety and risk. The report is preliminary in that it reflects the current stage of the NuScale design. It will be updated periodically as the NuScale design progresses. Final categorization will be presented in the design certification application (DCA).

1.2 Scope

The methodology used in categorizing SSCs is summarized in Section 2.0. Although this methodology is applicable to all NuScale SSCs, a subset of systems and functions were selected, based on the scope of the SRP, and categorized for this stage of the design. Quantitative risk metrics used as input for categorizing systems are described in Section 3.0. Categorization results for systems and functions are listed in Section 4.0.

1.3 Abbreviations

Table 1-1. Abbreviations

Term	Definition
ATWS	anticipated transient without scram
CCF	common-cause failure
CDF	core damage frequency
CFR	U.S. Code of Federal Regulations
CVCS	chemical and volume control system
DCA	design certification application
D-RAP	design reliability assurance program
DSRS	design-specific review standard
ESFAS	engineered safety feature actuation system
FV	Fussell-Vesely
iPWR	integral pressurized-water reactor
LRF	large release frequency
mcyr	module critical year
PRA	probabilistic risk assessment
RAW	risk achievement worth
RCS	reactor coolant system
RTNSS	regulatory treatment of nonsafety systems
RTS	reactor trip system
SBO	station blackout
SRP	Standard Review Plan
SSC	structure, system, and component

2.0 Methodology

The current stage of the NuScale design development does not allow SSC categorization on a component level for all systems. Thus, as an initial step in the categorization process, systems and functions are identified, and categorization is performed on a system and/or functional level.

The methodology has two fundamental elements:

1. identification of safety-related systems and functions
2. identification of risk-significant systems and functions

Systems and functions are binned into the four categories as illustrated in Figure 2-1.

The four categories are

S = Safety-related, risk-significant

X = Safety-related, not risk-significant

R = Risk-significant, nonsafety-related

N = Not risk-significant, nonsafety-related

RTNSS SSCs are a subcategory of R = Risk-significant, nonsafety-related.

Risk-significant functions become the bases for the D-RAP list; specifically, functions identified as Categories “S” and “R” are assigned to the D-RAP list. As the design progresses, the D-RAP list will be refined to identify specific SSCs associated with risk-significant functions.

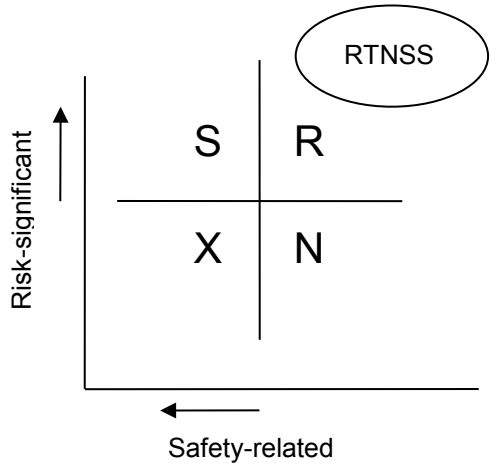

Figure 2-1. Structure, system, and component categories

2.1 Safety-Related Structures, Systems, and Components

An SSC is classified as safety-related, as defined in 10 CFR 50.2, if it is relied on to remain functional during and after a design basis event to assure:

(1) The integrity of the reactor coolant pressure boundary

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

Plant-level safety functions are defined which, if successfully executed, accomplish the three functional goals in 10 CFR 50.2. For the NuScale design, a preliminary set of plant-level safety functions is to

- control core reactivity
- maintain primary coolant inventory
- maintain reactor pressure control
- remove core heat
- maintain containment integrity

These plant-level safety functions are characteristic of a light-water reactor with multiple barriers to a radionuclide release, and are also applicable to the NuScale iPWR. However, the small NuScale size and passive features result in unique system-level characteristics needed to perform the plant-level safety functions.

NuScale is currently revising the design-basis event spectrum that is associated with the plant-level safety functions and the unique NuScale design features. For the NuScale DCA, it is expected that specific design-basis events will be grouped in the following six types of events, consistent with the categories in Chapter 15.0 of the SRP (Reference 5.7):

(1) Increase in heat removal by the secondary system
(2) Decrease in heat removal by the secondary system
(3) Reactivity and power distribution anomalies
(4) Increase in reactor coolant inventory
(5) Decrease in reactor coolant inventory
(6) Radioactive release from a subsystem or component

Note that the SRP grouping *Decrease in RCS flow rate* is not applicable, because the NuScale design does not use reactor coolant pumps. Decrease in reactor coolant system (RCS) flow rate is reflected within the phenomenology of other event types.

After design-basis events are defined, SSCs used to mitigate those events are classified as safety-related. When making this determination, the effects of single active failures in those systems are considered.

2.2 Nonsafety-Related Structures, Systems, and Components

As indicated by NRC guidance, nonsafety-related SSCs may be risk-significant. Risk-significance in this context fundamentally addresses SSCs that are not encompassed by the plant's "design basis," but have been determined to be significant enough to warrant special treatment with regard to quality or administrative control considerations. Examples include SSCs needed to

mitigate events defined by regulation, such as anticipated transient without scram (ATWS) (Reference 5.8) and station blackout (SBO) (Reference 5.9), insights from a PRA, and ongoing industry initiatives such as are associated with the Fukushima event response.

Many of the criteria used to define such SSCs are defined by RTNSS as specified in draft SRP Section 19.3 and SECY-95-132 (References 5.6 and 5.10):

A. *SSC functions relied upon to meet beyond design basis deterministic NRC performance requirements such as those set forth in Title 10 of the Code of Federal Regulations (10 CFR) 50.62 for mitigating Anticipated Transient Without Scram (ATWS) and in 10 CFR 50.63 for station blackout (SBO).*

B. *SSC functions relied on to ensure long-term safety (beyond 72 hours) and to address seismic events.*

C. *SSC functions relied on under power-operating and shutdown conditions to meet the Commission's safety goal guidelines of a core damage frequency (CDF) of less than* $1x10^{-4}$ *each reactor year and a large release frequency (LRF) of less than* $1x10^{-6}$ *each reactor year.*

D. *SSC functions needed to meet the containment performance goal (SECY-93-087, Issue I.J), including containment bypass (SECY-93-087, Issue II.G), during severe accidents.*

E. *SSC functions relied on to prevent significant adverse systems interactions between passive safety systems and active non-safety SSCs.*

Additional criteria to define nonsafety-related SSCs as risk significant include risk-importance measures, special events, operating experience, and expert panel judgment.

2.3 Categorization of Structures, Systems, and Components

The NuScale methodology requires that the appropriate engineering and safety analysis organizations develop most of the information necessary to categorize SSCs according to the criteria discussed in Sections 2.1 and 2.2. Both safety- and nonsafety-related SSCs are assessed based on criteria needed to established risk significance, such as importance measures.

Because the NuScale design is not complete, categorization is initially performed on a functional level. For example, the functions of the chemical and volume control system (CVCS) include RCS makeup, primary coolant chemistry control, and containment isolation. Each function is identified so that it may be categorized in one of four categories as shown in Figure 2-1. The categories are consistent with that suggested in the NRC's draft SRP Introduction, Part 2:

1. Safety-related, risk-significant (S): This category includes SSCs relied on to mitigate a design-basis event and found to be risk significant.
2. Safety-related, not risk-significant (X): This category includes SSCs relied on to mitigate a design-basis event but not found to be risk-significant.
3. Risk-significant, nonsafety-related (R): This category includes SSCs not relied on to mitigate a design-basis event but found to be risk-significant. Components associated with RTNSS criteria would be assigned to this category.
4. Nonsafety-related, not risk-significant (N): This category includes SSCs not relied on to mitigate a design-basis event and not found to be risk-significant.

The categorization information with regards to risk significance is reviewed by an expert panel whose membership includes personnel knowledgeable of nuclear plant design and operation.

Panel collective expertise includes PRA, safety analysis, plant operations, maintenance, design engineering, and system engineering. Conclusions of the expert panel are documented in panel meeting minutes and will be used to establish the NuScale D-RAP list, which will be provided in the NuScale DCA.

In addition to the formalized criteria provided as input to the expert panel, additional considerations are made when categorizing SSCs as risk-significant, e.g., multi-module risk, operating experience (beyond that considered in establishing basic event and initiating event frequencies), and recognized safety principles such as defense-in-depth.

SSCs are evaluated as to whether they reduce the likelihood or mitigate the consequences of severe accidents. Insights from the PRA provide both plant risk metrics (e.g., CDF) and individual component risk metrics (e.g., risk achievement worth [RAW]).

For this stage of the NuScale design development, risk metrics are evaluated at the system and/or functional level.

3.0 Risk Analysis

3.1 Approach

One of the inputs to the SSC categorization methodology is the use of the NuScale PRA to assess the importance of each system with respect to CDF. Currently, only the full-power internal-events PRA is developed sufficiently to produce quantitative risk results. The approach taken in estimating the risk impact relies on the execution of a series of sensitivity calculations to generate effective risk importance measures (specifically, RAW and Fussell-Vesely (FV)). These sensitivity calculations are performed assuming that a particular system of interest is always failed (i.e., perfectly unreliable), or always successful (i.e., perfectly reliable). The CDF is then calculated for each of these two cases and compared to the base case CDF to calculate each of the two risk importance measures.

The NuScale PRA is maintained in keeping with the evolving design in order to provide risk insight for various design considerations (i.e. risk-inform the design). As part of the design evolution it is expected that the contribution of different systems in terms of risk significance will change.

3.2 Metrics and Thresholds

Common practice for judging risk significance based on importance measures is to use threshold values for RAW and FV (NEI 00-04 [Reference 5.11]). The thresholds values typically used to identify candidate risk-significant SSCs are

- sum of FV for all basic events modeling the SSC of interest, including common-cause events > 0.005
- maximum of component basic event RAW values > 2
- maximum of applicable common-cause basic events RAW values > 20

If any of these criteria are exceeded, the SSC is typically considered a candidate for risk-significant categorization. However, these criteria were developed for operating nuclear power plants that typically have a CDF for internal events at-power, in the range of 1E-4 to 1E-6 per year. A RAW value of two for a particular SSC implies that if that particular SSC were to fail with a 100% probability, then the CDF would increase by a factor of two. For a baseline CDF of 1E-5/year, an increase by a factor of two (ΔCDF of 1E-5/year) represents a 10% increase relative to the commonly accepted safety goal of 1E-4/year.

However, if the baseline CDF is on the order of 1E-8/year, then an increase by a factor of two is much less significant (0.01% increase relative to the 1E-4/year CDF safety goal). In addition, uncertainties typical of nuclear power plant PRAs make reliance on numbers as small as 1E-8/year unstable and imprecise. Clearly, using thresholds of FV = 0.005 and RAW = 2 for risk-significance would be extremely conservative for plants with a CDF of less than 1E-7/year. As described in Appendix A of Regulatory Guide 1.174 (Reference 5.12), relative importance measures do not directly relate to changes in risk (ΔCDF and ΔLRF). Therefore, the thresholds chosen for making determinations of risk significance should take into account the baseline CDF.

Additionally, the thresholds listed above have typically been applied on an individual component basis (use of the term SSC notwithstanding). System-level importance measures are not commonly reported. Given the state of the NuScale design development, the more robust approach is to focus on systems rather than individual components. Use of thresholds for evaluating component contributions would add additional conservatism when evaluations are made at the system level.

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3.3 Risk Results

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4.0 Structure, System, and Component Categorization - Preliminary Results

{{ }}3(a-c) Systems were selected for initial categorization based on the systems and functions that are addressed in the NRC's Standard Review Plan (NUREG-0800). Table 4-1 presents SSC categorization results on a system functional level.

Table 4-1. System functions and categories

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