

## Licensing Topical Report

# Risk Significance Determination

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## **Abstract**

Standard Review Plan (SRP) Section 19.0 (Reference 6.1.24) states that the term ‘significant,’ in the context of probabilistic risk assessment (PRA) results and insights, is intended to be consistent with the definition provided in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.200 (Reference 6.1.16). 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.

An alternative approach to determining risk significance is needed because the traditional relative importance measures are insensitive to the global improvements in safety associated with the lower risk profile of the NuScale Power, LLC (NuScale) design. Continued use of the aforementioned relative risk criteria of RG 1.200 would artificially raise the relative importance of structures, systems, and components (SSC) that do not drive risk in the NuScale design, eventually leading to an unnecessary expenditure of resources for both the licensee and regulatory staff.

Therefore, NuScale is implementing an approach which employs an absolute evaluation of the RAW importance measure. The alternative approach is implemented in a manner that is consistent with the guidelines in RG 1.174 (Reference 6.1.15) that provide the risk-informed integrated decision-making framework for making changes to a licensee’s approved licensing basis. 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 (CDF) or large release frequency (LRF). The NuScale criteria balance measures to ensure maintaining margins to NRC safety goals with those of maintaining the current risk profile.

## Executive Summary

This report describes the methodology NuScale Power, LLC (NuScale) has elected to identify candidate risk-significant structures, systems, and components (SSCs) using probabilistic risk assessment (PRA). This methodology involves the use of alternative metrics to Regulatory Guide (RG) 1.200 metrics for defining the term “significant.” Standard Review Plan (SRP) Section 19.0, Probabilistic Risk Assessment and Severe Accident Evaluations for New Reactors, states that the term “significant” is intended to be consistent with the definition provided in RG 1.200, and any other definition shall be subject to additional staff review and approval.

The traditional RG 1.200 criteria, including those referenced in SRP Section 19.0, are based on *relative* risk metrics and the risk profiles of operating nuclear power plants (NPPs). Because of the NuScale passive design, the risk profile is significantly lower compared to the current fleet of NPPs. The traditional relative criteria, however, are insensitive to global improvements in safety. While NuScale design features significantly reduce the frequency of accident sequence types that dominate the risk profile for current plants, continued use of the same relative risk criteria would artificially raise the relative importance of SSCs that do not drive risk, eventually leading to an unnecessary expenditure of resources for both the applicant, licensee, and NRC staff.

For the current fleet of operating NPPs, the full-power internal events mean core damage frequency (CDF) is on the order of  $1 \times 10^{-5}$  per year. This is in contrast to NuScale where the full-power internal events CDF is on the order of  $1 \times 10^{-7}$  per year.

With a baseline CDF of  $1 \times 10^{-5}$  per year, using the relative risk metric of risk achievement worth (RAW) of 2 implies a change in CDF of  $1 \times 10^{-5}$  per year. This is in contrast to NuScale, where a CDF on the order of  $1 \times 10^{-7}$  per year and a RAW of 2 means the CDF increases by only  $1 \times 10^{-7}$  per year. Consequently, when using a relative risk metric such as RAW, a plant with a baseline CDF of  $1 \times 10^{-5}$  per year would allow an increase in CDF of  $1 \times 10^{-5}$  per year, whereas a much safer plant with a baseline CDF of  $1 \times 10^{-7}$  would only allow an increase of  $1 \times 10^{-7}$  per year before an item becomes risk-significant. Using the current relative risk criteria would result in categorizing a majority of NuScale PRA systems as risk-significant, which does not reflect NuScale’s low risk, and the significant reduction in risk compared to the current fleet of NPPs. For identical reasons, the Economic Simplified Boiling Water Reactor (ESBWR) design application proposed alternative risk significance thresholds, which were approved by the staff.

Using the relative risk criteria would also be overly conservative with respect to U.S. Nuclear Regulatory Commission (NRC) guidelines for making plant changes. As outlined in NRC RG 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis (Reference 6.1.15), when the total CDF is small, larger increases in risk may be accepted by the NRC when making permanent changes to a plant’s licensing basis. For example, a change in CDF of less than  $1 \times 10^{-6}$  per year is considered very small or not risk-significant for a plant with a CDF on the order of  $1 \times 10^{-5}$  per year. The RG 1.174 guidance supports making permanent plant changes based on small changes in risk, while meeting regulations, maintaining defense-in-depth, maintaining sufficient safety margins, monitoring performance, and meeting NRC safety goals.

As summarized in Table 1, the NuScale criteria for determining SSC risk significance includes an *absolute* evaluation of the RAW importance measure, and a backstop using the Fussell-

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Vesely (FV) importance measure. The NuScale approach directly addresses the ratio limitations of traditional importance measures and can be implemented in a manner consistent with meeting NRC safety goals (Reference 6.1.28), goals for new reactors (Reference 6.1.4), and the guidelines in RG 1.174 for making permanent licensing basis changes (Reference 6.1.15). It is also consistent with the Advisory Committee on Reactor Safeguards (ACRS) recommendation that risk significance criteria should be consistent for a broad spectrum of designs and absolute levels of overall plant risk (Reference 6.1.35). The NuScale methodology includes evaluating all SSCs considered in the PRA, against CDF and large release frequency (LRF), internal and external hazards, and all operating modes. This methodology will be used to identify candidate risk-significant SSCs in the PRA in support of risk-informed applications, including the design reliability assurance program (DRAP) to support the design certification (DC) application.

Table 1. NuScale criteria for risk significance

<b>Parameter</b>	<b>Criteria for Risk Significance</b>
Component-level basic event	Conditional CDF $\geq 3 \times 10^{-6}/\text{yr}$
System-level basic event	Conditional CDF $\geq 1 \times 10^{-5}/\text{yr}$
Component-level basic event	Conditional LRF $\geq 3 \times 10^{-7}/\text{yr}$
System-level basic event	Conditional LRF $\geq 1 \times 10^{-6}/\text{yr}$
Basic event/contributor	Total FV $\geq 0.20$

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## 1.0 Introduction

### 1.1 Purpose

The purpose of this report is to describe NuScale’s approach for determining risk significance in the context of the PRA and provide the basis for the approach and relevant criteria. It includes:

- a discussion of the current risk significance criteria and why they are inappropriate for NuScale.
- an example of how the criteria will be used in DC, specifically for the DRAP discussed in SRP Section 17.4.
- NuScale criteria based on an absolute evaluation of the RAW importance measure.
- NuScale criteria based on the FV importance measure.
- the basis for the NuScale criteria, including a comparison to NRC safety goals and the goals for new reactors.

### 1.2 Scope

This report provides the methodology for identifying SSCs in the PRA as candidates for risk-significance. It applies to the PRA for internal hazards and external hazards, and all operating modes, including low-power and shutdown. It also applies to the analysis of CDF (i.e., Level 1 PRA) and LRF (i.e., Level 2 PRA) for a single, individual module.

The SSCs not considered in the PRA are outside of the scope of this report. The SSCs typically not modeled in the PRA include those that do not result in a reactor trip, do not perform a safety-related function as defined in 10 CFR 50.2 (Reference 6.1.10) (or support or complement a safety function), do not support operator actions credited in the PRA (including recovery actions), and are not part of a system that acts as a barrier to fission product release during a severe accident.

**Note:** All numbers in this report associated with CDF, conditional core damage frequency (CCDF), LRF, large early release frequency (LERF), and conditional large release frequency (CLRF), are on a “per reactor critical year” basis.

### 1.3 Abbreviations and Definitions

Table 1-1. Abbreviations

Term	Definition
ACRS	Advisory Committee on Reactor Safeguards
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
CCDF	conditional core damage frequency
CCFP	conditional containment failure probability
CDF	core damage frequency

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<b>Term</b>	<b>Definition</b>
CLRF	conditional large release frequency
COL	combined operating license
DC	design certification
DRAP	design reliability assurance program
ESBWR	Economic Simplified Boiling Water Reactor
FV	Fussell-Vesely
LERF	large early release frequency
LRF	large release frequency
LWR	light-water reactor
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
PRA	probabilistic risk assessment
QHO	quantitative health objectives
RAP	reliability assurance program
RAW	risk achievement worth
RG	regulatory guide
RII	risk increase interval
RIR	risk increase ratio
RRW	risk reduction worth
RTNSS	regulatory treatment of nonsafety systems
SRP	standard review plan
SSC	structure, system, and component

Table 1-2. Definitions

<b>Term</b>	<b>Definition</b>
baseline PRA	For a nuclear power plant at the DC or COL stage, where the plant is not built or operated, the baseline PRA model reflects the as-designed plant.
basic event	An element of the PRA model for which no further decomposition is performed, because it is at the limit of resolution consistent with available information. There are typically two types of basic events: equipment unavailabilities and human errors.
conditional probability	In PRA, a conditional probability can be calculated for containment failure, core damage, and large release given the knowledge that a prior event has occurred.
core damage frequency	The sum of the accident sequence frequencies of those accident sequences whose end state is core damage.
cutset (minimal cutset)	A combination of failures that result in a particular outcome, such as core damage or large release.
external hazard	An event or a natural phenomenon that poses some risk to a facility. External hazards include events such as flooding and fires external to the plant, tornadoes, earthquakes, and aircraft crashes.
Fussell-Vesely	A PRA importance measure that provides the relative contribution of an event to the calculated risk. This relative or fractional contribution is

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Term	Definition
	obtained by determining the reduction in risk if the probability of the event is set to false (or zero) (i.e., the component is assumed to be always functioning properly).
internal hazard	An event that poses some risk to a facility. Internal hazards include events such as equipment failures, human failures, and flooding and fires internal to the plant.
large release frequency	The frequency of an unmitigated release of airborne fission products from the containment to the environment such that there is a potential for health effects.
licensing basis	The collection of licensee commitments that provides the basis upon which the NRC issues a license to construct and operate a nuclear facility.
risk achievement worth	A PRA importance measure that provides the increase in risk if an event is assumed to be always unavailable or is assumed to be failed (i.e., true or set to 1.0).
risk increase interval	An indication of how much the minimal cutset upper bound would increase if the basic event probability were increased (to a probability of 1.0).
risk reduction worth	A PRA importance measure that provides the decrease in risk if an event is assumed to be perfectly reliable (i.e., false or set to 0).
risk-based	A characteristic of decision-making in which a decision is solely based on the numerical results of a risk assessment.
risk-informed	A characteristic of decision-making in which risk results or insights are used together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to health and safety.

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## 2.0 Background

In 1986, the Commission published the Safety Goal Policy Statement for the Operation of Nuclear Power Plants (Reference 6.1.28). In defining an acceptable level of risk from NPP operation, the Commission indicated that it believed that current regulatory practice ensured compliance with the basic statutory standard of adequate protection. The Commission also believed that current practices could be improved to provide better means for testing the adequacy of current requirements. In establishing this policy, the Commission established two qualitative safety goals that are supported by two quantitative health objectives (QHO) for use in the regulatory decision-making process. The qualitative safety goals are:

- Individual members of the public should be provided a level of protection from the consequences of NPP operation such that individuals bear no significant additional risk to life and health.
- Societal risk to life and health from NPP operation should be comparable to or less than the risk of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The quantitative objectives used in determining achievement of the safety goals are:

- The risk to an average individual in the vicinity of an NPP or prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
- The risk to the population in the area near an NPP of cancer fatalities that might result from NPP operation should not exceed one-tenth of one percent of the sum of cancer fatality risks resulting from all other causes.

Following completion of the 1975 Reactor Safety Study (Reference 6.1.33), the Commission recognized that it was feasible to use quantitative safety objectives. The Commission proposed the following general performance guideline for the staff to use as a basis for determining whether a level of safety ascribed to a plant is consistent with the safety goal policy (Reference 6.1.28):

Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation (i.e.,  $LRF < 1 \times 10^{-6}$  per reactor year).

In addition, the Commission identified the following subsidiary objective (Reference 6.1.20):

A core damage probability of less than 1 in 10,000 per year of reactor operation (i.e.,  $CDF < 1 \times 10^{-4}$  per reactor year).

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In 1995, in its Final Policy Statement on the Use of PRA Methods in Nuclear Regulatory Activities (Reference 6.1.29), the Commission determined that the use of PRA technology should be increased in all regulatory matters, to the extent supported by state-of-the-art PRA methods and data. This policy statement recommended increasing the use of PRA in all regulatory matters in a manner that complements the NRC's deterministic approaches and the traditional defense-in-depth philosophy. The PRA methods have been applied successfully in several regulatory activities and have proven to be a valuable complement to the traditional engineering approaches.

Reactor risk metrics refer to the quantitative measures of risk to the public from reactor operations up to and including severe core damage accidents. The two primary risk metrics used in evaluating operating reactors are CDF and large early release frequency (LERF). These metrics have been used as surrogates for the QHOs. CDF is generally regarded as the surrogate for the individual cancer fatality risk QHO, and LERF has been shown to be an adequate surrogate for the individual early fatality risk QHO (Reference 6.1.19). Note that while the use of CDF and LERF/LRF are surrogates for risk, use of these surrogates for the NuScale design is conservative compared to the way they have been used in the past because the size of the NuScale reactor core is much smaller.

In SECY-12-0081, Risk-Informed Regulatory Framework for New Reactors (Reference 6.1.27), the Commission reaffirmed that existing safety goals, subsidiary risk goals and associated risk guidance, and quantitative metrics for implementing risk-informed decision making are sufficient for new plants. Today, the NRC's risk-informed and performance-based plan identifies the initiatives that help the agency achieve the Commission's goal of a holistic, risk-informed, and performance-based regulatory structure. There have been a number of guidance documents published by the NRC that discuss NRC expectations in cases when a licensee chooses to employ risk arguments to address a licensing issue (References 6.1.12, 6.1.13, 6.1.14, 6.1.15).

PRAs can be used to evaluate the risk profile of a plant to ensure that the design and operating practices satisfy the NRC's safety goals. In addition, PRAs can be used to identify cases where failing SSCs or exceeding design values could potentially lead to core damage and a release to the environment. Providing risk-informed insights helps operators and regulators ensure that the risks resulting from changes in reliability or availability are maintained acceptably low.

## **2.1 Regulatory Requirements**

Through the mid-1980s and mid-1990s, the Commission issued a series of policy statements regarding safety goals for operating reactors and expectations for advanced reactors (Reference 6.1.20). In its policy statement, Severe Reactor Accidents Regarding Future Designs and Existing Plants (Reference 6.1.22), the Commission stated that it, "fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe-accident safety performance than their prior designs." In its Policy Statement on Regulation of Advanced Nuclear Power Plants (Reference 6.1.23), the Commission further stated that, "the Commission expects that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety and security

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functions.” The following summarize the safety goals and subsidiary objectives for new reactors (Reference 6.1.17 and 6.1.21):

- CDF <  $10^{-4}$  per reactor year
- LRF <  $10^{-6}$  per reactor year
- Conditional containment failure probability (CCFP) less than approximately 0.1

Regulatory interactions in the areas of licensing and oversight rely upon a number of regulatory processes and guidance, some of which are risk-informed. In the current framework, RG 1.174 provides the risk-informed integrated decision-making framework to support permanent changes to a licensee’s approved licensing basis (Reference 6.1.15). The RG 1.174 acceptance guidelines are based on subsidiary objectives derived from the safety goals and their QHOs. A key principle in risk-informed regulations is that proposed changes in CDF and risk are small and consistent with the intent of the Commission’s Safety Goal Policy Statement (Reference 6.1.28).

These quantitative safety goals are identified as acceptance criteria for risk in SRP Section 19.0. Section 19.0 of the SRP covers the design-specific PRA for a DC application and plant-specific PRA for a combined operating license (COL) application. It covers severe accidents and the use of PRA in the design and operation of facilities under review. Section 19.0 of the SRP includes the staff’s acceptance criteria that apply to the PRA and severe accident evaluation (Reference 6.1.24). Acceptance Criterion No. 17 in Section 19.0 of the SRP states:

In the context of the PRA results and insights, the term “significant” is intended to be consistent with its definition provided in RG 1.200. The definitions of “significant accident sequence” and “significant contributor” are suitable for both LERF and LRF. Using any other definition of “significant” inconsistent with the definitions provided by RG 1.200 shall be subject to additional staff review and approval.

In RG 1.200, the following numerical criteria are recommended for defining significance:

- Basic events/contributors that have a RAW > 2
- Basic events/contributors that have a FV importance > 0.005
- Set of sequences (defined at the functional or systemic level) that compose 95 percent of the CDF or LERF/LRF, or that individually contribute more than ~one percent to CDF or LERF/LRF (Reference 6.1.16)

Significance is measured with respect to the contribution to the total CDF or LERF, or with respect to the contribution to the CDF or LERF for a specific hazard group, or plant operating state. While there is no published basis for the current risk thresholds, they have been used successfully in a variety of applications (References 6.1.5, 6.1.6, 6.1.13) and are endorsed by the NRC in RG 1.174.

The following equations provide details on how RAW (or the risk increase ratio [RIR]) and FV importance measures are calculated:

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$$RAW = RIR = R1/Rb$$

Where:

- R1 = increased risk with basic event set to true (i.e., 1.0, failed) = conditional CDF (CCDF), or conditional LRF (CLRF)
- Rb = baseline PRA risk metric (i.e., CDF or LRF)

$$FV = 1 - R0/Rb$$

Where:

- R0 = decreased risk with basic or initiating event set to false (i.e., 0.0, perfectly reliable)
- Rb = baseline PRA risk metric (i.e., CDF or LRF)

These criteria are consistent with the definition of significance in the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) standard, Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications (Reference 6.1.1). They are also consistent with the interim staff guidance developed to convey the staff position on use of the PRA standard for advanced light water reactor DC or COL applications until other documents are updated (Reference 6.1.9). They were developed in the context of regulating currently operating NPPs that have PRAs, and were licensed under the traditional deterministic regulations, and designed and built before the widespread use of PRA.

## 2.2 Need for NuScale Thresholds

The relatively simple, passive NuScale design results in risk estimates that are significantly lower than those of operating plants. In order to reflect the benefits of global improvements in safety in risk significance determination, NuScale is employing thresholds using an *absolute* risk metric.

The existing thresholds for identifying risk significance, measured as a ratio to the total CDF or LERF, while appropriate for the current fleet of large operating NPPs, are not appropriate for NuScale. For the current fleet of operating NPPs, CDFs are typically on the order of  $1 \times 10^{-5}$  per reactor year. With a baseline CDF of  $1 \times 10^{-5}$  per reactor year, using the *relative* risk metric of a RAW of 2 implies a change in CDF of  $1 \times 10^{-5}$  per year; a component is risk-significant if its CCDF is greater than  $2 \times 10^{-5}$  per year.

This is in contrast to NuScale, where a CDF on the order of  $1 \times 10^{-7}$  per year and a RAW of 2 means the CDF increases by only  $1 \times 10^{-7}$  per year. In this case, a component would

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be considered risk-significant if its CCDF is only  $2 \times 10^{-7}$  per year, a value that is much lower than that of the operating fleet. Consequently, when using a relative risk metric such as RAW, a plant with a baseline CDF of  $1 \times 10^{-5}$  per year would allow an increase in CDF of  $1 \times 10^{-5}$  per year, whereas a much safer plant with a baseline CDF of  $1 \times 10^{-7}$  would only allow an increase of  $1 \times 10^{-7}$  per year before an item becomes risk-significant.

As the frequencies of core damage (and large release) become smaller and smaller, the uncertainties associated with those frequency estimates become relatively larger. It might appear counter-intuitive, but assuring completeness in the PRA becomes more challenging when dealing with such low frequency events. At the relatively high frequencies associated with the current generation of plants, there can be reasonable confidence that the *dominant* contributors and the drivers of the risk profile (i.e., those events that most influence the risk) have been identified and accounted for. For the low frequencies associated with new passive designs, assuring complete accounting of all the significant contributors is more problematic. At such low frequencies (i.e., less than one-in-a-million year events), relevant hazards are much more difficult to identify (including both natural hazards and man-made hazards). At these low frequencies natural disasters that have potentially global consequences such as asteroid impact and super-volcanoes, and inherently unpredictable human errors of commission (both intentional and unintentional), could very well become the dominant risk contributors. The consequence of this is, while there can be reasonable confidence that the risk is very low, there should be more skepticism in identifying the dominant contributors to that very low risk since subtle changes in assumptions, conservatism or including/excluding selected very rare events can significantly change the risk profile. Relying on this risk profile by using relative importance measures to guide the expenditure of resources would not be a particularly effective or efficient strategy.

With the uncertainties associated with such small risk estimates, using absolute metrics will provide assurance that the risk (albeit uncertain with respect to the dominant contributors) will remain low. Specifically, when assessing changes to an already very low CDF the use of absolute metrics provides confidence that any “non-significant” change in risk will result in the risk remaining very low. This is one reason why NuScale’s absolute metrics for establishing risk significance are more appropriate than RG 1.200’s relative metrics.

Although the RG 1.200 criteria are based on relative metrics, the use of absolute metrics is not new. For example, in NUREG/CR-3385, “Measures of Risk Importance and their Applications” (Reference 6.1.31), RAW intervals and risk reduction worth (RRW) intervals were used to evaluate the importance of safety functions and containment for four nuclear plants. The Birnbaum importance measure is also an absolute risk measure. The following provides details of how the RAW interval, also called the risk increase interval (RII), is calculated:

$$RAW \text{ interval} = RII = R1 - Rb$$

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Where:

- R1 = increased risk with basic event set to true (i.e., 1.0, failed) = CCDF, or CLRF
- Rb = baseline PRA risk metric (i.e., CDF or LRF)

While the ratio and interval definitions generally give the same rankings for a plant, the interval definition is more appropriate for cost-benefit evaluations, or when different plants are compared. The use of an interval measure of importance will also address some of the limitations of ratio forms of importance measures. The ACRS raised the issue of traditional importance measures being insensitive to global improvements in safety in its letter to the Commission on plans for developing risk-informed revisions to 10 CFR Part 50 (Reference 6.1.7). Results using universally fixed FV and RAW criteria were compared; even though several protection systems were added, thereby reducing reliance on the original system and reducing overall risk, the importance measures did not change. In addition, the ACRS noted in implementation of 10 CFR 50.69, that it is the absolute value of risk that is critical while importance measures provide only relative risk.

In reducing the frequency of accident sequences that dominate current plant designs, the traditional FV and RAW criteria artificially raise the importance of traditionally less important SSCs. Therefore, it is appropriate to use the RAW interval (or RII) as an absolute measure in the determination of risk significance.

The ACRS also recommended that criteria used to determine risk significance be consistent for a broad spectrum of designs and absolute levels of overall plant risk (Reference 6.1.35). In the new plant designs, with very low estimated frequencies of core damage and large releases, the ACRS noted that a large number of SSCs may be identified as risk-significant using the RG 1.200 criteria; universal application of the RG 1.200 criteria may produce an inappropriately large population of SSCs that are subject to enhanced availability and reliability controls, with commensurate undue burden for both the licensee and regulatory staff.

### 2.3 New Light Water Reactor Design Thresholds

Alternative thresholds to those found in RG 1.200 have been proposed and approved for other new light water reactor designs.

As discussed in NUREG-1966, Volume 4 (Reference 6.1.30), the Staff approved the following thresholds for identifying potentially risk-significant basic events in the Economic Simplified Boiling-Water Reactor (ESBWR) design:

- $RAW \geq 5$  for individual events
- $FV \geq 0.01$  for individual events
- $RAW \geq 50$  for common-cause failure events

Basic events that do not meet the threshold values are considered not risk-significant in the context of the ESBWR PRA.

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## 2.4 NuScale Reliability Assurance Program

The reliability assurance program (RAP) is one area where risk significance will be used to support the NuScale DC application. SRP Section 17.4 provides staff guidance on how to perform safety reviews of the RAP in DC and COL applications (Reference 6.1.25). The RAP is implemented in two stages. The first stage, the DRAP, encompasses reliability assurance activities that occur before initial fuel load. The second stage is comprised of the reliability assurance activities conducted during the operations phase of the plant's life cycle. The RAP applies to those SSCs, both safety-related and nonsafety-related, identified as risk-significant. As discussed in SRP Section 17.4, the SSCs within the scope of the RAP are identified by using a combination of probabilistic, deterministic, and other methods to identify and quantify risk, including information obtained from sources such as the PRA, severe accident evaluations, industry operating experience, and expert panels. The DRAP provides assurance that risk-significant component reliability assumptions made in the DC are retained during the detailed design process, and are considered in ongoing operational reliability assurance activities, such as the Maintenance Rule (Reference 6.1.11).

In addition, the NRC has established requirements for nonsafety-related SSCs that perform risk-significant functions (i.e., regulatory treatment of nonsafety systems [RTNSS], Reference 6.1.32). These nonsafety-related SSCs are relied upon under power-operating and shutdown conditions to meet the NRC safety goal guidelines of CDF <  $1 \times 10^{-4}$  per reactor year and LRF <  $1 \times 10^{-6}$  per reactor year. Nonsafety-related SSCs that are identified as subject to RTNSS are classified as risk-significant in the DRAP program.

The traditional RG 1.200 criteria artificially raise the relative importance of SSCs in the NuScale plant. As an illustrative example, roughly half of the systems modeled in the PRA were flagged as risk significant utilizing RG 1.200 criteria in a preliminary NuScale PRA model. Including roughly half of the PRA systems in DRAP would direct resources on SSCs that do not control risk at the expense of SSCs that are important in maintaining safety. It should be noted that this illustrative example focused on system-level importance. To maximize the benefit of the DRAP, SSCs should be identified that focus resources on design and operational issues commensurate with providing reasonable assurance of adequate protection of the environment and public health and safety.

### 3.0 Analysis/Methodology

#### 3.1 Selection of Risk Significance Thresholds

The NuScale approach focuses on identifying SSCs whose reliability and availability are important in ensuring that margins to the NRC safety goals are maintained, while also maintaining the low risk profile. The NuScale approach can be implemented in a manner consistent with the risk-acceptance guidelines established in RG 1.174, and when appropriate, SRP Section 19.2 (Reference 6.1.26). These guidelines are based on the principles and expectations for risk-informed regulation and support licensing basis changes to a NPP. Figure 3-1 shows the guidelines from RG 1.174 (Reference 6.1.15). It depicts the changes in CDF and LERF that the NRC considers acceptable when making permanent changes to a plant's licensing basis. As this figure illustrates, for those cases where the baseline CDF and LERF are small, larger risk increases may be accepted by the NRC.

Although the RG 1.174 guidance addresses permanent plant changes, the guidelines are intended to provide assurance that proposed increases in CDF and LERF are small and are consistent with the intent of the Commission's safety goals. While importance measures do not directly relate to changes in risk, the risk impact is indirectly reflected in the value used to determine whether or not the SSC is risk-significant. As such, the NuScale criteria for identifying candidates for risk significance are based on the acceptance guidelines for changes in CDF and LRF that are small, while total CDF and LRF remain well below the safety goals.

**Note:** This topical report establishes metrics against LRF; LRF and CCFP are used during DC and COL application reviews. As discussed in SECY-12-0081, "Risk-Informed Regulatory Framework for New Reactors" (Reference 6.1.27), the staff recommends transitioning at or before initial fuel load from LRF and CCFP to LERF. Also, as discussed in SECY-13-0029, "the staff's view is that the objective of using LRF as a basis for determining whether a level of safety ascribed to a plant is consistent with the safety goal policy statement is fulfilled today by the use of LERF and CDF guidelines for operating reactors (Reference 6.1.19). As such, the NuScale criteria for identifying candidates for risk significance based on LERF would be the same as those proposed for LRF (i.e., "CLRF" could be replaced with "CLERF" in the NuScale criteria for identifying candidates for risk significance without needing to adjust the thresholds). (This is conservative based on the LRF goal of  $< 10^{-6}$  per year being more restrictive than the LERF goal of  $< 10^{-5}$  per year.)

The guidance in RG 1.174 supports decisions for making plant changes based on small changes in risk while meeting current regulations and safety goals, and maintaining sufficient defense-in-depth and safety margins. These guidelines are also consistent with the Commission's view that "reaffirms that the existing safety goals, safety performance expectations, subsidiary risk goals and associated risk guidance (such as the Commission's 2008 Advanced Reactor Policy Statement and Regulatory Guide 1.174), key principles and quantitative metrics for implementing risk-informed decision making, are sufficient for new plants" (Reference 6.1.27).

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Sections 3.1.1 through 3.1.3 describe the NuScale criteria for identifying candidate risk-significant SSCs using the PRA. Following each section is the basis and justification for the criteria.

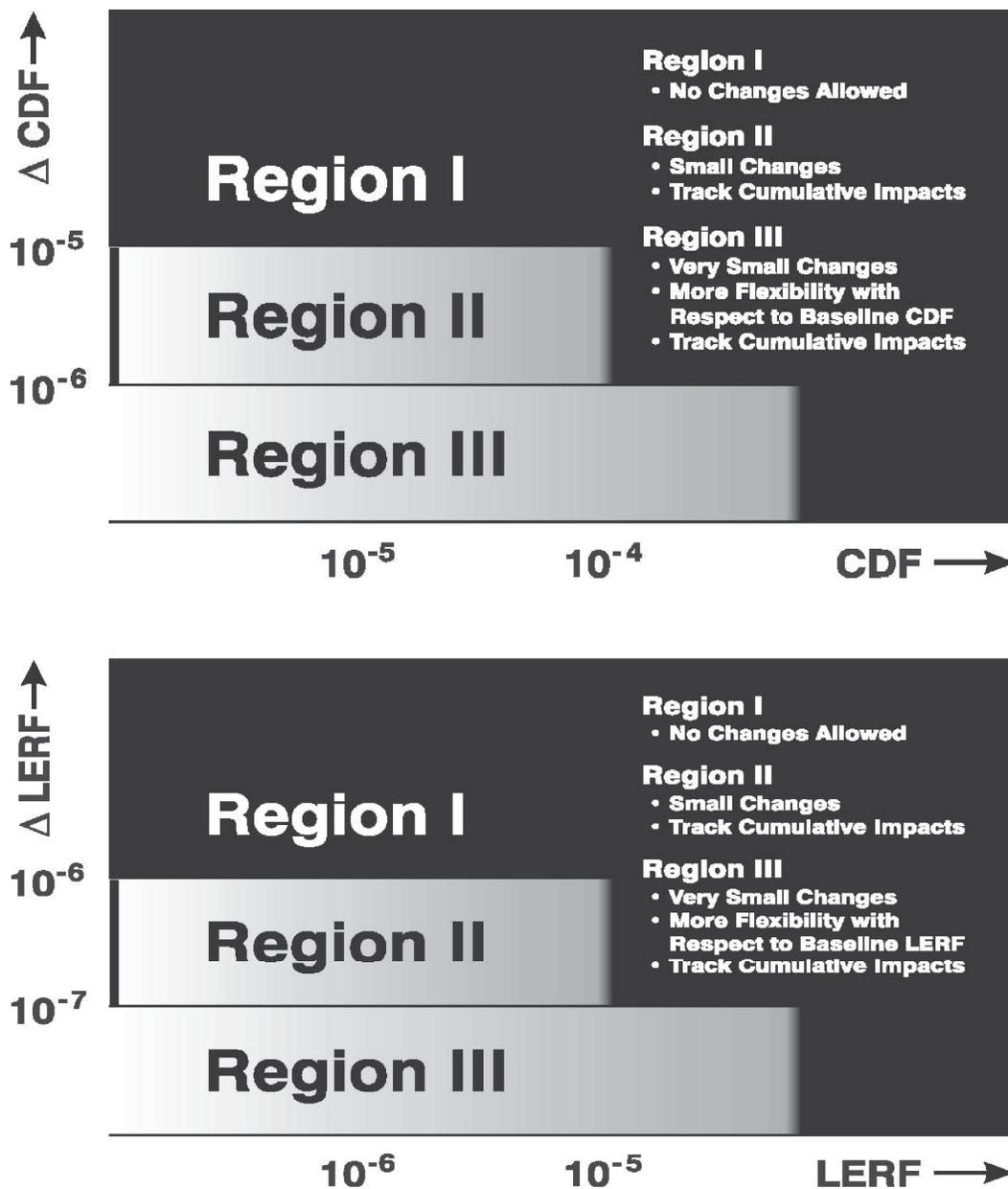


Figure 3-1. NRC Regulatory Guide 1.174 acceptance guidelines

### 3.1.1 Criteria for Core Damage Frequency

The current industry practice for judging component-level risk significance associated with equipment failure uses the RAW importance measure. The RAW estimates the increase in risk that would result without the component (i.e., if the component failed 100

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percent of the time). The RAW is especially informative for components with high reliability because it shows increases in risk when the reliability of the component is considerably reduced.

The current criteria uses a threshold value for RAW of greater than 2 for components (Reference 6.1.16). If, when failing a component-level basic event, risk increases by a factor of 2, the component is considered risk-significant. However, because the current criteria were developed for operating plants that typically have a CDF for internal events on the order of  $1 \times 10^{-5}$  per year, a RAW of 2 for a failed component would result in an increase in CDF by a factor of 2. For a baseline CDF of  $1 \times 10^{-5}$  per reactor year, an increase in CDF by a factor of 2 represents a significant loss in safety margin with respect to the NRC safety goal of  $1 \times 10^{-4}$  per reactor year. However, if the baseline CDF is on the order of  $1 \times 10^{-7}$  per reactor year, an increase in CDF by a factor of 2 is not significant in relation to the  $1 \times 10^{-4}$  per reactor year safety goal. Maintaining an expectation for risk significance based on a RAW of 2 effectively results in a new safety goal for designs that are demonstrably safer than the existing fleet.

Therefore, the NuScale approach employs an absolute evaluation of the RAW importance measure; a criteria consisting of the component-level CCDF value being of greater than or equal to  $3 \times 10^{-6}$  per year. Basically, if the failure of any component results in a CCDF of  $3 \times 10^{-6}$  per year or higher, it will be considered a risk-significant candidate.

For a system-level threshold, criteria from NEI 00-04 (Reference 6.1.5), as endorsed in Regulatory Guide 1.201, (Reference 6.1.18), was considered<sup>1</sup>; specifically the importance measure criteria for common cause events is based on a RAW value of 20. This value reflects that common cause is measuring the failure of two or more redundant trains. As such, this common cause criterion applies to system-level basic events. Most systems expected to provide important safety missions in the NuScale design typically include some intrasystem redundancy. As such, current industry practice uses about an order of magnitude increase for system-level metrics compared to component-level metrics.

Therefore, when system-level PRA events are evaluated, the NuScale criteria will use a threshold of  $1 \times 10^{-5}$  per year (i.e., about a half order of magnitude above the component-level threshold of  $3 \times 10^{-6}$  but within Region II of RG 1.174). If the failure of any system results in a CCDF of  $1 \times 10^{-5}$  per year or higher, the system will be considered a risk-significant candidate.

These thresholds, applied at a single module level, would be applicable to all initiating events collectively, that is aggregated across all hazards (i.e., internal events, low-power and shutdown conditions, internal flooding, internal fires, and external hazards). The risk significance thresholds could be partitioned based on expectations regarding the contribution to risk from each hazard. For example, if only an internal events PRA model is available, and internal events are expected to contribute roughly ten percent of total

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<sup>1</sup> Note: NuScale does not presently intend to implement 10 CFR 50.69 within the scope of design certification and is simply including reference to guidance associated with 10 CFR 50.69 (NEI 00-04) for the purposes of this report.

plant risk, the risk significance thresholds could be reduced to ten percent of the component-level and system-level thresholds discussed above.

### 3.1.1.1 Basis/Justification for Core Damage Frequency Criteria

The component-level CCDF value of  $3 \times 10^{-6}$  was chosen by virtue of it being approximately the midpoint (on a log scale) between  $1 \times 10^{-5}$  and  $1 \times 10^{-6}$ . This value provides an order of magnitude margin to the NRC safety goal of  $1 \times 10^{-4}$  per year for CDF, with an extra half-order of magnitude (on a log scale) of margin to account for uncertainties in the PRA model.

Using the range between  $1 \times 10^{-5}$  and  $1 \times 10^{-6}$  as the threshold for determining risk significance is further supported by guidance from RG 1.174, which addresses making permanent changes to a plant's licensing basis:

“When the calculated increase in CDF is in the range of  $1\text{E-}6$  per reactor year to  $1\text{E-}5$  per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than  $1\text{E-}4$  per reactor year (Region II).”

The component-level threshold of  $3 \times 10^{-6}$  represents approximately the midpoint (on a log scale) of the Region II range identified in RG 1.174, and the total CDF will remain significantly below the  $10^{-4}$  per year value. Furthermore, NuScale is proposing using this threshold to simply identify risk significance on a component level, and is not implying these evaluations constitute permanent changes to the design.

The system-level value of  $1 \times 10^{-5}$  represents the upper end of the Region II range for CDF identified in RG 1.174 for making permanent changes to a plant's licensing basis. It provides an order of magnitude margin to the NRC safety goal of  $1 \times 10^{-4}$  per year for CDF. It is also an order of magnitude lower than  $1 \times 10^{-4}$  per year for CDF, the surrogate for the latent cancer fatality QHO.

### 3.1.2 Criteria for Large Release Frequency

In addition to core damage, SSCs are evaluated for risk significance against LRF, the Level 2 criteria. The NuScale approach is similar to that for CDF. The criteria will consider an absolute evaluation of RAW, and consider both component and system-level thresholds. The thresholds for LRF are an order of magnitude below those for CDF.

If the failure of any component (i.e., setting the failure probability to true) results in a CLRF of  $3 \times 10^{-7}$  per year or higher, it will be considered a risk-significant candidate. If the failure of any system results in a CLRF of  $1 \times 10^{-6}$  per year or higher, it will be considered a risk-significant candidate.

These thresholds, applied at a single module level, would be applicable to all initiating events collectively, that is aggregated across all hazards (i.e., internal events, low-power and shutdown conditions, internal flooding, internal fires, and external hazards). Similar to the CDF metrics, the LRF risk significance thresholds could be partitioned based on expectations regarding the contribution to risk from each hazard.

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### 3.1.2.1 Basis/Justification for Large Release Frequency Criteria

The component-level value of the  $3 \times 10^{-7}$  was chosen for several reasons. First, it is an order of magnitude below the criteria for CCDF, which is consistent with the Commission CCFP goal of 0.1 for new reactors. In addition, it is below the Commission-established goal of less than  $1 \times 10^{-6}$  per year for LRF for new reactor designs, and below the NRC safety goal of less than  $1 \times 10^{-6}$  for a large release.

The value  $3 \times 10^{-7}$  is also approximately an order of magnitude below the LERF guideline of  $1 \times 10^{-5}$  per year, which is used as the surrogate for the prompt fatality QHO, with an extra half-order of magnitude (on a log scale) margin to account for uncertainties. Finally, this value is within the RG 1.174 Region II criteria; the range between  $1 \times 10^{-6}$  and  $1 \times 10^{-7}$  addresses making permanent changes to a plant's licensing basis:

“When the calculated increase in LERF is in the range of  $1\text{E-}7$  per reactor year to  $1\text{E-}6$  per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than  $1\text{E-}5$  per reactor year (Region II).”

The threshold of  $3 \times 10^{-7}$  represents the midpoint of the Region II range identified in RG 1.174, and the total LRF will remain significantly below the  $10^{-5}$  per year value.

The system-level value of  $1 \times 10^{-6}$  represents the upper end of the LERF Region II range identified in RG 1.174 for making permanent changes to a plant's licensing basis. This value meets the NRC safety goal of  $1 \times 10^{-6}$  per year for large release. It is also an order of magnitude lower than the guideline of  $1 \times 10^{-5}$  per year for LERF, the surrogate for the prompt fatality QHO.

As discussed in Section 3.1, NuScale's criteria are based on LRF; LRF and CCFP are used during DC and COL application reviews. Because the objective of using LRF is fulfilled today by the use of LERF for operating plants (Reference 6.1.19); the criteria for LERF would be the same as those proposed for LRF.

### 3.1.3 Criteria Based on Contribution

To supplement the absolute RAW metrics, NuScale is employing an additional metric to identify those SSCs that have the largest fractional contribution to risk, regardless of CDF or LRF. The focus of this criteria is on identifying SSCs for which the reliability and availability have the greatest influence on the risk profile.

This metric is a contribution threshold and proposes that any SSC modeled in the PRA that contributes 20 percent or more to risk be considered a risk-significant candidate (i.e., FV greater than or equal to 0.20). This criteria ensures that any SSC that has an unusually large contribution to risk is identified and the reasons for that contribution are examined, regardless of CDF or LRF.

Current industry practice for judging risk significance based on contribution also makes use of the FV importance measure. The FV importance measure allows events to be ranked according to their contribution to overall risk. It measures the overall percent

contribution of cut sets containing a basic event of interest to the total risk. This criteria is used to identify SSCs that are a significant fraction of a hazard with very low risk.

In addition to equipment unavailabilities and human failures, this metric also includes internal initiating event contributors. Internal initiating events are included because they play an important role in a PRA. External initiating events are excluded because they are not initiated by plant components.

This threshold, applied at a single module level, would be applied individually to each hazard group, and mode of plant operation. For example, SSCs will be identified as candidates for risk significance if they contribute 20 percent or more (i.e.,  $FV \geq 0.20$ ) to internal events risk, or seismic risk, or external flood risk, etc. It will also be applied individually to CDF and LRF; SSCs will be identified as candidates for risk significance if they contribute 20 percent or more to internal events CDF or internal events LRF. It is being applied individually because the focus is on identifying SSCs for which the reliability and availability have the greatest influence on risk.

### **3.1.3.1 Basis/Justification of Criteria Based on Contribution**

Based on the risk of operating NPPs and the current FV importance threshold, a reasonable measure of significance can be rationalized based on the significantly lower CDF associated with the NuScale design. For a typical operating plant with a CDF of  $1 \times 10^{-5}$  per year, the current FV criterion (i.e., 0.005 or 0.5 percent) translates to a CDF of  $5 \times 10^{-8}$  per year. For a plant like NuScale, with an expected CDF on the order of  $1 \times 10^{-7}$  per year, a CDF of  $5 \times 10^{-8}$  per year corresponds to an FV of 0.5, or 50 percent. However, setting a threshold for FV at 0.5 does not reflect the intent behind the use of FV for identifying those components that contribute a significant portion of the risk. To be more consistent with this intention, a FV of 0.2 (i.e., or 20 percent) is proposed here.

While the absolute RAW metrics identify SSCs in accordance with the RG 1.174 guidelines, the FV metric identifies additional events, including internal initiating events, that are a significant fraction of a hazard with very low risk.

## **3.2 Implementation/Use**

NuScale has elected to utilize risk information in many of the decision-making processes employed to develop a safe, economical, and efficient design.

As discussed in Section 2.4, applying the RG 1.200 criteria to the preliminary NuScale PRA model, with an internal events CDF on the order of  $1 \times 10^{-7}$  per year, identified roughly half of the systems modeled as risk-significant. When the NuScale absolute RAW and FV thresholds are applied to the preliminary NuScale PRA model, the number of risk-significant systems is reduced by approximately 25 percent. Identifying less than half of the PRA modeled systems as candidates for risk significance will better focus resources on design and operational issues commensurate with their importance to health and safety.

Consistent with the principles of risk-informed decision-making, PRA results and importance measures will be used to identify candidate risk-significant SSCs. However,

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the ultimate determination of risk significance will be based on the specific application, with appropriate consideration of uncertainties, sensitivities, traditional engineering evaluations and regulations, and maintaining sufficient defense-in-depth and safety margins. As such, PRA risk insights will be considered along with deterministic approaches and defense-in-depth concepts such that NuScale is utilizing a “risk-informed” rather than a solely “risk-based” approach. In short, these new criteria will be implemented in the same way the traditional RG 1.200 relative risk criteria would have been.

## 4.0 Summary of Justification

As discussed in Section 3.1, justification of each criteria is included in the subsection immediately following the proposed criteria.

The NuScale criteria focus on identifying SSCs whose availability and reliability are important in maintaining margins to NRC safety goals with those of maintaining the current risk profile. Table 4-1 shows how the NuScale criteria for risk significance meet NRC goals and guidelines. An SSC is a candidate for risk significance if it meets the criteria in Table 4-1.

Table 4-1. Basis for NuScale criteria for risk significance

Criteria for risk significance		Basis
Component-level	$CCDF \geq 3 \times 10^{-6}/\text{yr}$	<ul style="list-style-type: none"> <li>Order of magnitude below NRC safety goal of <math>1 \times 10^{-4}/\text{yr}</math> for CDF.</li> <li>Order of magnitude below CDF guideline of <math>10^{-4}/\text{yr}</math> used as surrogate for the cancer fatality QHO.</li> <li>Consistent with RG 1.174 criteria for Region II permanent licensing basis changes: <math>\Delta CDF</math> between <math>10^{-5}</math> and <math>10^{-6}</math> considered if CDF can reasonably be shown to be <math>&lt; 1 \times 10^{-4}/\text{yr}</math>.</li> <li>Component-level vs. system-level in line with NEI 00-04 and RG 1.201 categorization guidelines.</li> </ul>
System-level	$CCDF \geq 1 \times 10^{-5}/\text{yr}$	
Component-level	$CLRF \geq 3 \times 10^{-7}/\text{yr}$	<ul style="list-style-type: none"> <li>Meets NRC safety goal of <math>1 \times 10^{-6}/\text{yr}</math> for large release, meets Commission-established goal of <math>&lt; 1 \times 10^{-6}/\text{yr}</math> for LRF for new reactor designs, and consistent with Commission-established goal of 0.1 for CCFP for new reactors.</li> <li>Order of magnitude below LERF guideline of <math>10^{-5}/\text{yr}</math> used as surrogate for the prompt fatality QHO.</li> <li>Consistent with RG 1.174 criteria for Region II permanent licensing basis changes: <math>\Delta LERF</math> between <math>10^{-6}</math> and <math>10^{-7}</math> considered if LERF can reasonably be shown to be <math>&lt; 1 \times 10^{-5}/\text{yr}</math>.</li> <li>Component-level vs. system-level in line with NEI 00-04 and RG 1.201 categorization guidelines.</li> </ul>
System-level	$CLRF \geq 1 \times 10^{-6}/\text{yr}$	
SSCs	Total FV $\geq 0.20$	<ul style="list-style-type: none"> <li>Threshold consistent with criteria for operating plants (i.e., CDF of <math>1 \times 10^{-5}/\text{yr}</math> &amp; FV of 0.005 = <math>5 \times 10^{-8}/\text{yr}</math> is approximately equal to the NuScale CDF of <math>1 \times 10^{-7}/\text{yr}</math> &amp; FV of 0.2).</li> </ul>

## 5.0 Summary and Conclusions

Consistent with Commission policy, new reactor designs have an enhanced level of severe accident prevention and mitigation capability. As a result, new designs have achieved a higher standard of severe accident safety performance than prior designs and provide an enhanced margin of safety in preventing core damage, containment failure, and a release. For example, NuScale’s full-power internal events PRA CDF is on the order of  $1 \times 10^{-7}$  per year.

The traditional risk criteria utilized by the operating fleet are overly conservative for newer plant designs with very low CDF values, like NuScale. Therefore, NuScale has developed alternative risk criteria to identify candidate SSCs as risk-significant. The NuScale criteria includes an absolute evaluation of the RAW importance measure, and a backstop using the FV importance measure. The criteria directly address the ratio limitations of traditional importance measures thereby identifying SSCs whose reliability and availability are important in maintaining safety, margins to NRC safety goals, and the low risk profile. This focuses resources on the SSCs whose design and operation provides reasonable assurance of maintaining adequate protection of the environment and public health and safety.

The NuScale approach for identifying candidate risk-significant SSCs using the PRA is implemented in a manner consistent with the guidelines established in RG 1.174. While importance measures do not directly relate to changes in risk, the risk impact is indirectly reflected in the value used to determine whether or not the SSC is risk-significant. These guidelines are based on the principles and expectations for risk-informed regulation and support licensing basis changes to an NPP.

The methodology involves calculating importance measures for events in the PRA and identifying those that meet the criteria outlined in Table 5-1. The criteria apply to the full-scope PRA, including all hazards and operating modes, and both CDF and LRF. The criteria apply to all PRA basic events, including those implicitly evaluated in the PRA through operator action, as well as internal initiating events. The thresholds are applied at a single module level; the absolute RAW thresholds apply to the aggregated risk across all hazards, and the FV thresholds apply individually to each hazard group and mode of plant operation, and individually to CDF and LRF.

Table 5-1. NuScale criteria for risk significance

Parameter	Criteria for Risk Significance
Component-level basic event	CCDF $\geq 3 \times 10^{-6}$ /yr
System-level basic event	CCDF $\geq 1 \times 10^{-5}$ /yr
Component-level basic event	CLRF $\geq 3 \times 10^{-7}$ /yr
System-level basic event	CLRF $\geq 1 \times 10^{-6}$ /yr
Basic event/contributor	Total FV $\geq 0.20$

These criteria will be used to examine events considered in the PRA and identify candidates for risk significance. Results will be used to support the DRAP, and other applications that consider risk significance. Applications, such as DRAP, will use the

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PRA risk-significant insights; however, the process will be risk-informed rather than risk-based with appropriate consideration of uncertainties, sensitivity analyses, traditional engineering evaluations and regulations, and maintaining sufficient defense-in-depth and safety margins.

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## **NuScale Nonproprietary**