October 10, 2016

Docket: PROJ0769

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

SUBJECT: NuScale Power, LLC Submittal of the Accepted Version of NuScale Licensing Topical Report: TR-0515-13952-NP-A "Risk Significance Determination," Revision 0, (TAC No. RN6110)

REFERENCES: 1. Letter from Frank Akstulewicz (NRC) to Thomas Bergman (NuScale), "Final Safety Evaluation for NuScale Power, LLC Licensing Topical Report: TR-0515-13952-NP, 'Risk-Significance Determination,' Revision 0, TAC No. RN6110" July 13, 2016.

By letter dated July 13, 2016 (Reference 1), the NRC issued a final safety evaluation report documenting the NRC staff conclusion that the licensing topical report TR-0515-13952-NP, "Risk-Significance Determination," Revision 0, is acceptable for referencing in licensing applications for the NuScale small modular reactor design. Reference 1 requested that NuScale publish the accepted version of TR-0515-13952-NP within three months of receipt thereof. Accordingly, the enclosure to this letter provides TR-0515-13952-NP-A. This accepted version of the subject topical report now incorporates the July 13, 2016 NRC letter and its enclosed final safety evaluation report after the title page and also provides historical review information in the form of letters to and from the NRC, including the NRC RAIs and NuScale responses.

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

Please feel free to contact Jennie Wike at 541-360-0539 or at jwike@nuscalepower.com if you have any questions.

Sincerely,

Thomas Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC

Distribution: Frank Akstulewicz, NRC, TWFN-6C20
Greg Cranston, NRC, TWFN-6E55
Omid Tabatabai, NRC, TWFN-6E7
Mark Tonacci, NRC, TWFN-6E55

Enclosure: NuScale Topical Report: TR-0515-13952-NP-A “Risk Significance Determination,” Revision 0

Enclosure:

NuScale Topical Report: TR-0515-13952-NP-A “Risk Significance Determination,” Revision 0

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

Risk Significance Determination

October 2016

Revision 0

Docket: PROJ0769

NuScale Nonproprietary

NuScale Power, LLC

1100 NE Circle Blvd., Suite 200

Corvallis, Oregon 97330

www.nuscalepower.com

© Copyright 2016 by NuScale Power, LLC

Contents

Section A

July 13, 2016

Mr. Thomas Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC
1100 NE Circle Boulevard, Suite 200
Corvallis, OR 97330

SUBJECT: FINAL SAFETY EVALUATION FOR NUSCALE POWER, LLC LICENSING TOPICAL REPORT: TR-0515-13952-NP, “RISK-SIGNIFICANCE DETERMINATION,” REVISION 0, TAC NO. RN6110

Dear Mr. Bergman:

On July 30, 2015, NuScale Power, LLC (NuScale) submitted Licensing Topical Report (TR) TR-0515-13952-NP, “Risk-Significance Determination,” Revision 0, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15211A469) to the NRC for staff review and approval. The NRC staff has found that the TR-0515-13952-NP, “Risk-Significance Determination,” Revision 0, is acceptable for referencing in licensing applications for NuScale small modular reactor design to the extent specified and under the conditions and limitations delineated in the enclosed safety evaluation report (SER). The SER defines the basis for acceptance of the TR.

Our acceptance applies only to matters approved in the subject TR. We do not intend to repeat our review of the acceptable matters described in the topical report. When the report appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. Regulatory licensing action requests that deviate from this TR will be subject to additional staff reviews in accordance with applicable review standards.

In accordance with the guidance provided on the NRC’s TR website (http://www.nrc.gov/about-nrc/regulatory/licensing/topical-reports.html), we request that NuScale publish an accepted version of this TR within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed safety evaluation between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain in appendices historical review information, such as questions and accepted responses, and original report pages that were replaced. The accepted version shall include an "-A" (designated accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion in this letter, that the TR is acceptable, is invalidated, the NuScale Power, LLC and/or the applicant referencing the TR will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the TR without revision of the respective documentation.

Sincerely,

/RA/ SLee for

Frank Akstulewicz, Director
Division of New Reactor Licensing
Office of New Reactors

Project No. PROJ0769

Enclosure: Safety Evaluation

cc w/encl: DC NuScale Power LLC Listserv

Staff Safety Evaluation Report for
NuScale Power, LLC Licensing Topical Report
TR-0515-13952-NP, "Risk-Significance Determination," Revision 0

1.1 Introduction

In a July 30, 2015, letter, NuScale Power, LLC (NuScale), submitted licensing topical report (TR) TR-0515-13952-NP, "Risk-Significance Determination," Revision 0, (Ref. 1) to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval. The subject TR describes the methods NuScale has elected to identify candidate risk-significant structures, systems, and components (SSC) using probabilistic risk assessment (PRA). This method involves using alternative metrics than those contained in Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 2), for defining the term "significant." NRC NUREG-0800, "Standard Review Plan" (SRP), Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluations for New Reactors" (Ref. 3), 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.

This safety evaluation report (SER) is based on the submitted letter and responses to requests for additional information (RAI). TR-0515-13952-NP, Revision 0, is designed to be used to support certification of the NuScale design and referenced in an initial combined license (COL) application that also references a certified NuScale design or in an amendment to a license issued to a COL applicant whose application referenced a certified NuScale design, as desired. This SER is divided into seven sections. Section 1 is the introduction. Section 2 presents a summary of applicable regulatory criteria and guidance. Section 3 contains a summary of the information presented in the TR. Section 4 contains the technical evaluation of the major components of TR-0515-13952-NP, Revision 0. Section 5 presents the conclusions of this review. Section 6 contains the restrictions and limitations on using the risk significance determination TR. Section 7 lists the references.

2.0 REGULATORY CRITERIA

2.1 Requirements

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(27) state that a design certification application must contain a final safety analysis report (FSAR) that includes a description of the design-specific PRA and its results. With respect to this regulation, the following items are noted:

- The Statement of Consideration (*Federal Register*, Vol. 72, No. 166, p. 49380 (72 FR 49380)) for the revised 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," states the understanding that the complete PRA (e.g., codes) will be available for NRC inspection at the applicant's offices, if needed.

- The regulations in 10 CFR 52.79(a)(46) state that a COL application must contain an FSAR that includes a description of the plant-specific PRA and its results. With respect to this regulation, the following item is noted: The Statement of Consideration (72 FR 49387) for the revised 10 CFR Part 52 states the understanding that the complete PRA (e.g., codes) would be available for NRC inspection at the applicant's offices, if needed.

2.2 <u>Relevant Guidance</u>

The NUREG-0800 (Ref. 4) supplies detailed review guidance that the staff finds acceptable in meeting the applicable regulatory requirements. In particular, NUREG-0800 sections that contain guidance relevant to this review are:

1. Section 19.0, Revision 3, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," January 2016

2. Section 19.2, "Review of Risk Information Used To Support Permanent Plant Specific Changes to the Licensing Basis: General Guidance," June 2007

3. Section 17.4, "Reliability Assurance Program," Revision 1, May 2014

The RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009, discusses criteria for determining risk-significance of SSCs.

The RG 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," March 2011, offers guidance on thresholds for risk significance of plant-specific changes to its licensing basis and using absolute risk measures in risk-informed applications that involve categorization of SSCs in the plant.

3.0 SUMMARY OF TECHNICAL INFORMATION

This licensing TR provides the NuScale methods for identifying SSCs as candidates for risk-significance in a NuScale-design PRA or the PRA of an applicant that references a certified NuScale design in a licensing application. It applies to a PRA that addresses internal hazards and external hazards, and all operating modes, including low-power and shutdown. It also applies to the analysis of core damage frequency (CDF, i.e., Level 1 PRA) and large release frequency (LRF, i.e., Level 2 PRA) for a single individual reactor module.

In the report, NuScale notes that the metrics for determining risk significance given in RG 1.200 are relative in nature and the specific values were established based on the collective results of PRAs performed for operating reactors in the 1990s and later (i.e., estimates of CDF and large early release frequency (LERF)). Based on design and analysis work performed to date, NuScale believes that, because of the simplicity and extensive use of passive design features in the NuScale design, its PRA will yield risk estimates that are several orders of magnitude lower than those of operating plants. Using the traditional metrics specified in RG 1.200 with a PRA that produces risk estimates several orders of magnitude lower than those of operating plants would likely result in identification of many components as risk significant that are not truly risk significant (i.e., components whose assumed failure would not increase CDF nor LRF significantly.) Such a result is counter to NRC policy (60 FR 42622) on use of PRA to help focus resources on the most truly safety-significant issues. Therefore, to reflect reduced risk in its determination of risk significance, NuScale has developed thresholds using absolute risk metrics.

The NuScale approach employs significance criteria consisting of the component-level core damage and large release frequency, conditional on complete component failure, greater than or equal to 3×10^{-6} per year and 3×10^{-7} per year, respectively. In other words, if an assumed failure of any component in the PRA model results in a conditional core damage frequency (CCDF) of 3×10^{-6} per year or higher, or conditional large release frequency (CLRF) of 3×10^{-7} per year or higher, it will be considered a risk-significant candidate. When system-level PRA events are evaluated, NuScale proposes a threshold of 1×10^{-5} per year for CDF. If the failure of any system results in a CCDF of 1×10^{-5} per year or higher, the system will be considered a risk-significant candidate.

These thresholds, applied at a single reactor module level, would be applicable to all initiating events collectively, aggregated across all hazards (i.e., internal events, low-power and shutdown conditions, internal flooding, internal fires, and external hazards).

NuScale's rationale for the component-level CCDF threshold is that the value provides an order of magnitude margin to the NRC goal of 1×10^{-4} per year for CDF, with an extra half-order of magnitude (on a log scale) [1] of margin to account for uncertainties in the PRA model. In addition, it notes that the value is consistent with the use of 1×10^{-5} per year as the threshold beyond which risk increases are too large to allow permanent changes to a plant's licensing basis given in RG 1.174 for a plant with a base CDF below 1×10^{-4} per year. The selection of the CLRF threshold an order of magnitude less than the CCDF follows the same approach taken in RG 1.174.

NuScale is employing an additional metric to identify those SSCs, human errors, or internal initiating event contributors that have the largest fractional contribution to risk, regardless of CDF or LRF. This metric is the Fussell-Vesely (FV) importance measure. The objective in applying this criterion is to ensure that any entity that has an unusually large contribution to risk is identified and the reasons for that contribution are examined, regardless of CDF or LRF. This metric is a contribution threshold and proposes that any SSC or other entity 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 threshold, applied at a single reactor module level, would be applied individually to each hazard group and mode of plant operation.

NuScale selected a significance threshold of 20 percent with the objective of maintaining consistency between the absolute value of risk associated with significant contributors for NuScale and the absolute value of risk associated with significant contributors in current operating plants. Current operating plants typically have CDFs 2 orders of magnitude greater

---

1 This corresponds to the fact that the midpoint between decades on a logarithmic scale is close to a value of 3.

than new passive designed plants such as NuScale. The traditional FV threshold value used by operating reactors is 0.5 percent. A threshold of 50 percent for NuScale would identify contributors that represent the same level of absolute risk as operating plants, given the expectation that the NuScale base CDF will be 2 orders of magnitude lower than that for current operating plants. NuScale, however, chose a threshold of 20 percent because it believed that some important contributors could be screened out by using a value as high as 50 percent.

4.0 TECHNICAL EVALUATION

In the absence of specific review procedures for evaluating methods for assessing risk significance, the staff identified the following three key areas of review for the licensing TR:

- use of absolute versus relative risk metrics for assessing risk significance;
- selection of threshold values for absolute risk metrics; and
- application of the risk metrics.

Use of Absolute versus Relative Risk Metrics for Assessing Risk Significance

In the licensing TR, NuScale argues that use of the traditional threshold values for risk achievement worth (RAW) (i.e., 2) and FV (i.e., 0.005) is not appropriate for the NuScale design, because they were selected for the fleet of operating reactors that have mean CDFs in the range of $1{\times}10^{-5}$ per year and the current estimate of CDF for NuScale is $1{\times}10^{-7}$ per year.[2] NuScale also argues that use of the traditional threshold value for RAW would result in identifying SSCs or systems as risk significant when the risk, conditional on their failure, is insignificant. It also argues that overall CDF and LRF for a design must be accounted for in the selection of threshold values of RAW and FV. The staff agrees that using RAW and FV metrics with the traditional thresholds would not be appropriate for NuScale if the base CDF and LRF are substantially less than those of operating reactors. This is because, as discussed in RG 1.174, importance measures do not directly relate to absolute changes in risk. Instead, the risk impact is indirectly reflected in the choice of the value of the measure used to determine whether an SSC should be classified as being high- or low-safety significance. The staff states, in RG 1.174, that the criteria for categorization into low- and high-safety significance should relate to the acceptance criteria for changes in CDF and LERF and that the criteria should be a function of the base case CDF and LERF rather than being fixed for all plants. Thus, the applicant or licensee should demonstrate how the chosen criteria are related to, and conform to, the acceptance guidelines described in RG 1.174. The staff believes that the concept of classifying SSCs in high- and low-significance categories is similar enough to classifying SSCs as risk significant or not, so that the concepts summarized above apply when considering determination of risk significance in a gross fashion (i.e., risk significant or not). The staff finds using risk metrics derived based on absolute measures in risk in conjunction with base CDF and

[2] In a closed presentation on June 25, 2015, to the NRC Advisory Committee on Reactor Safeguards (80 FR 32979) Subcommittee on Future Plant Design, NuScale presented data that indicate that the CDF for internal events at full power operation and for shutdown operations would be several orders of magnitude less than those of current operating reactors.

base large release frequency acceptable for use in assessing risk significance of SSCs because such an approach is consistent with guidance in RG 1.174.

Selection of Threshold Values for Absolute Risk Metrics

NuScale proposes to use the following numerical criteria to identify candidate risk-significant SSCs:

Table 1

ID	Metric	Criterion
1	CDF Conditional on Component-Level Basic Event Failure	≥ 3×10^{-6} per year
2	CDF Conditional on System-Level Basic Event Failure	≥ 1×10^{-5} per year
3	LRF Conditional on Component-Level Basic Event Failure	≥ 3×10^{-7} per year
4	LRF Conditional on System-Level Basic Event Failure	≥ 1×10^{-6} per year
5	Contribution to CDF Conditional on Basic Event Failure	≥ 20% of Base CDF

In the LTR, NuScale cites acceptance guidelines in RG 1.174 for changes in CDF as a function of base CDF as the basis for Metric 1 (component-level) shown above. It argues that, according to the guidelines in RG 1.174, the selection of a value between 1×10^{-5} and 1×10^{-6} per year represents an acceptably small change for the base CDF of 1×10^{-7} per year estimated for the NuScale design, but still borders on a region in which changes are not acceptable except under extraordinary conditions.

The staff considered the apparent inconsistency between the metrics proposed by NuScale, which are conditional hazard frequencies, and the risk-significance criteria in RG 1.174, which are stated in terms of increases in hazard frequency for a given base frequency. The staff observed that increases in hazard frequency associated with each of the criteria specified by NuScale will always be less than the NuScale threshold values proposed for CCDF and CLRF (in Table 1 above) because base frequencies are always greater than zero. In light of this, the staff finds it meaningful, and therefore acceptable, to use thresholds for increase in risk given in RG 1.174 as "benchmarks" for evaluating thresholds for the metrics proposed by NuScale.

The staff reviewed the guidelines in RG 1.174 and, based on its review, agrees that a change in CDF of 1×10^{-5} per year represents a threshold beyond which changes would be considered significant for plants with a base CDF of up to 1×10^{-4} per year. The staff further agrees that, after adjustment to account for uncertainty, it would also be an acceptable choice for NuScale with its estimated base CDF at 1×10^{-7} per year—which is much lower than 1×10^{-5} per year—and the structure of the guidelines in RG 1.174 tend to allow a larger increase in the metric for smaller baseline values.

NuScale has selected a component-level threshold value of 3×10^{-6} per year. It states that this value is approximately halfway between 10^{-6} per year and 10^{-5} per year, which allows half an order of magnitude (on a logarithmic scale) to account for uncertainty in the PRA. As discussed above, the staff considered the information in RG 1.174 regarding changes in risk in evaluating the acceptability of the threshold proposed by NuScale. In addition, the staff considered information provided in the NRC regulatory analysis guidelines, NUREG/BR-058, "Regulatory

Analysis Guidelines of the Nuclear Regulatory Commission" (Ref. 5). NUREG/BR-058 includes guidance for estimating reduction in risk from proposed changes in existing NRC requirements or adding new requirements and screening criteria for determining if changing requirements would lead to a substantial reduction in risk. The threshold included in these criteria (i.e., 10^{-5} per year) is an order of magnitude less than the CDF surrogate for the Commission's safety goals of 10^{-4} per year. The NuScale-proposed threshold of 3×10^{-6} per year is an additional factor of about 3 lower than the risk-significance threshold used in regulatory analyses.

NuScale indicated that it allows 3×10^{-6} per year in its threshold to account for uncertainties in the PRA, but did not provide a basis for the uncertainty value. Accordingly, the staff asked NuScale to provide a basis for the specified uncertainty in a RAI issued on November 12, 2015 (Ref. 6). NuScale responded to the RAI on December 11, 2015 (Ref. 7). NuScale stated in its response that mean values of the metrics used in its method are compared to the threshold values. In addition, the company states that it expects the span from a mean value to the 95-percent value will be less than half an order of magnitude based on observations of commercial nuclear power plant PRAs and uncertainty results from the NuScale PRA. The staff evaluated the response by reviewing uncertainty results for commercial nuclear power plants as documented in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (Ref. 8), and PRA results for two nuclear power plant designs certified by the NRC that include passive safety systems. In all cases, the margin between the mean value for CDF and the 95th percentile was less than a factor of 10. Based on its review of the response, the staff finds that margin incorporated to account for PRA uncertainties is reasonable and acceptable.

The staff compared the NuScale-proposed component level threshold on CDF of 3×10^{-6} per year with the implicit threshold for CDF increase associated with the use of a RAW of 2.0 for plants with a base CDF of 1×10^{-5} per year. This staff used the definition of the RAW importance measure to derive the threshold on allowable increase in CDF associated with a RAW of 2.0 and a base model CDF of 1×10^{-5} per year. The RAW for a basic event is defined as follows:

RAWi = Ri+/R0 where:

Ri+ is overall model risk metric (e.g., CDF) with the probability of basic event i set to 1.0
R0 is base overall risk metric

The implicit significance threshold for a RAW of 2.0 and R0 of 1×10^{-5} per year is then 2×10^{-5} per year, which is about an order of magnitude greater than the threshold proposed by NuScale.

System-level significance involves failure of multiple components in a system or multiple trains of a system. System-level failures will normally have a larger effect on the ability to maintain safety functions. Accordingly, the staff finds it reasonable to choose a threshold above the component-level value. The CCDF and CLRF thresholds for system-level events chosen by NuScale are larger than the component-level values and are consistent with the thresholds for significance in RG 1.174. For these reasons, the staff finds the thresholds to be acceptable.

NuScale indicated in the licensing TR that it has selected component-level and system-level thresholds for LRF to be an order of magnitude below the thresholds for CDF. This is consistent with the approach taken for the guidelines in RG 1.174 and is also consistent with the NRC's goal for conditional containment failure in advanced reactors to be less than 0.1 (in SRP 19.0, Ref. 3). For these reasons, the staff finds NuScale's proposed criteria for increase in large release frequency to be acceptable.

NuScale has not proposed specific significance threshold values for RAW. The staff notes that given specific significance thresholds for Ri+ as defined above, RAW thresholds are theoretically not necessary. However, should such thresholds for RAW be desired for purposes of implementation in the PRA model, they must be derived using the approved thresholds on risk discussed above and the appropriate risk metric (CDF of LRF) determined with a technically adequate PRA that addresses internal hazards and external hazards, and all operating modes, including low-power and shutdown.

NuScale is following current industry practice for judging risk significance by including a criterion on overall percent contribution of cut sets containing a basic event of interest to the total risk. This criterion makes use of the FV importance measure. The FV importance measure allows events to be ranked according to their contribution to overall risk. This criterion is used to identify SSCs that are a significant fraction of a hazard not previously identified as significant. The threshold value for this criterion was derived by scaling the threshold on FV used currently by licensees for operating reactors to account for the difference between plant-level CDFs reported by operating reactors and the CDF currently being estimated for the NuScale reactor design. This scaling process preserves the amount of risk associated with significant contributors between NuScale and operating reactors. Indeed, FV thresholds for operating reactors and NuScale are both based on allowing the absolute value of risk—in terms of CDF—accountable to a significant contributor to be 5×10^{-8} per year. The result of applying this process with the current expected NuScale CDF of 1×10^{-7} per year is an FV threshold of 0.5. NuScale proposes to reduce this value to a value of 0.2. This is done to assure that significant contributors are not screened out. The staff finds this process of deriving an FV criterion to be acceptable because it maintains parity between NuScale and operating reactors with regard to the absolute amount of CDF contributed by a significant contributor. NuScale has created a de facto upper bound on FV of 0.2 for very low base CDFs, which is acceptable to the staff. However, the actual base CDF and LRF information for the NuScale design will not be available to the staff until NuScale submits its application for design certification. Should the actual base CDF for any of the specific hazards treated in the certified NuScale design be substantially higher than the assumed value of 1×10^{-7} per year, then the associated FV threshold derived using the process described above will be smaller than 0.2.

<u>Application of Risk Metrics</u>

In the LTR, NuScale has proposed key high-level features of its method of assessing risk significance using PRA for early staff review and approval prior to submitting its application for design certification. The staff notes that important implementation details have not been addressed in the LTR. This includes, for example, the way in which a RAW or FV is assigned to a component or system based on the RAWs and FVs computed for basic events associated with failure of the component or system, and the specific techniques for assessing risk significance of component or system failures caused by specific hazards such as fires and floods. Such issues are normally considered by the staff in its review of a specific application that involves assessment of risk significance, such as identification of SSCs to be included in the design-reliability assurance program or categorization of SSCs for treatment under the requirements in 10 CFR 50.69. Such applications are submitted as part of an application for design certification or a combined license after the PRA required under 10 CFR Part 52 has been completed and is available for audit by the staff. Use of the LTR in specific risk-informed applications is reviewed on a case-by-case basis by the NRC when those risk-informed applications are submitted for review.

5.0 STAFF CONCLUSIONS

The staff has completed its review of the NuScale licensing TR (Ref. 1) and concludes that, subject to the conditions and limitations specified in Section 6.0 of this SER, the NuScale methods described in the licensing TR are acceptable for identifying SSCs as candidates for risk-significance in a NuScale design PRA or the PRA of an applicant that references a certified NuScale design in a licensing application.

The staff's conclusions for specific technical topics are found within the respective technical evaluation sections of this report.

The staff, therefore, approves the use of the NuScale licensing TR (Ref. 1), subject to the conditions and limitations specified in Section 6.0 of this SER, by NuScale in support of design certification and to be referenced by NuScale COL holders or COL applicants as desired in accordance with applicable license requirements such as 10 CFR Part 52, Appendix D.

6.0 CONDITIONS AND LIMITATIONS

1. The staff's approval of this TR is specific to the NuScale generic design. Any use in whole or in part for other designs would require additional applicability review by the staff.

2. The criteria proposed by NuScale may be used as discussed above in Section 4.0 to identify candidate risk-significant SSCs for risk-informed applications. Specific risk-informed applications and implementations of those applications are reviewed case-by-case by the NRC. In keeping with NRC policy on risk-informed regulation, the ultimate determination of risk significance shall 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 margin. As such, PRA risk insights shall be considered along with deterministic approaches and defense-in-depth concepts such that the user is implementing a "risk-informed" rather than a solely "risk-based" approach.

3. Implementation of the NuScale methodology for identifying SSCs as candidates for risk-significance shall use a technically adequate PRA that addresses internal hazards and external hazards, and all operating modes, including low-power and shutdown. This also applies to the analysis of CDF (i.e., Level 1 PRA) and LRF (i.e., Level 2 PRA) for a single, individual reactor module, which also considers the criteria noted in SRP 19.0, Revision 3, regarding the impact of other modules or shared SSCs on the reactor module under analysis.

4. Values for thresholds on importance measures for NuScale may be derived based on the absolute risk thresholds and base CDF or LRF, provided the core damage frequency is very low (i.e., approximately $1x10^{-7}$ per year or less).

7.0 REFERENCES

1. U.S. Nuclear Regulatory Commission, “Risk-Significance Determination,” TR-0515-13952-NP, Revision 0, July 2015, Agencywide Documents Access and Management System (ADAMS) Accession No. ML15211A470.

2. U.S. Nuclear Regulatory Commission, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Regulatory Guide (RG) 1.200, Revision 2, March 2009, ADAMS Accession No. ML090410014.

3. U.S. Nuclear Regulatory Commission, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” Section 19.0, Revision 3, December 2015.

4. U.S. Nuclear Regulatory Commission, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” NUREG-0800, June 2007.

5. U.S. Nuclear Regulatory Commission, “Regulatory Analysis Guidelines of the Nuclear Regulatory Commission,” NUREG/BR-058, Revision 4, September 2004, ADAMS Accession No. ML042820192.

6. Request for Additional Information Letter No. 1 for the Review of NuScale Topical Report, TR-0515-13952, “Risk Significance Determination,” Revision 0 (TAC No. RN6110), ADAMS Accession No. ML15316A674.

7. NuScale Power, LLC Submittal of RA-1215-19837, “Response to Request for Additional Information Letter No. 1 for the Review of NuScale Topical Report, TR-0515-13952, ‘Risk Significance Determination,’ Revision 0,” (TAC No. RN6110), ADAMS Accession No. ML15348A369.

8. U.S. Nuclear Regulatory Commission, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” NUREG-1150, Vol. 1, December 1990, ADAMS Accession No. ML040140729.

Section B

NUSCALE POWER™

December 11, 2015

Docket: PROJ0769

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

SUBJECT: NuScale Power, LLC Submittal of RA-1215-19837, "Response to Request for Additional Information Letter No. 1 for the Review of NuScale Topical Report, TR-0515-13952, 'Risk Significance Determination,' Revision 0," (TAC No. RN6110)

REFERENCES:

1. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "Risk Significance Determination," LO-0715-15841, Revision 0, dated July 30, 2015 (ML15211A469).
2. NuScale Topical Report "Risk Significance Determination," TR-0515-13952-NP, Revision 0, dated July, 2015.
3. Letter from U.S. Nuclear Regulatory Commission to NuScale Power, LLC, "Request For Additional Information Letter No. 1 For the Review of the NuScale Topical Report, TR-0515-13952, 'Risk Significance Determination,' Revision 0 (TAC No. RN6110)," dated November 12, 2015.

In a letter dated July 30, 2015 (Reference 1) NuScale Power, LLC (NuScale) submitted the topical report entitled "Risk Significance Determination," Revision 0, (Reference 2). In a letter dated November 12, 2015 (Reference 3), the NRC Staff provided the Requests for Additional Information (RAI) regarding the subject topical report.

The purpose of this letter is to provide NuScale's response to the NRC's RAI. The enclosure to this letter includes NuScale's "Response to Request for Additional Information Letter No. 1 for the Review of NuScale Topical Report, TR-0515-13952, 'Risk Significance Determination,' Revision 0".

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

Please feel free to contact Jennie Wike at 541-360-0539 or at jwike@nuscalepower.com if you have any questions.

Sincerely,

Zackary Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Frank Akstulewicz, NRC, TWFN-6C20
Greg Cranston, NRC, TWFN-6E7
Omid Tabatabai, NRC, TWFN-6E7
Mark Tonacci, NRC, TWFN-6E7

Enclosure: "Response to Request for Additional Information Letter No. 1 for the Review of NuScale Topical Report, TR-0515-13952, 'Risk Significance Determination,' Revision 0"

BCC: Chief Operation Officer/Chief Nuclear Officer
Chief Strategy Officer
Chief Technology Officer
Chief Commercial Officer
Chief Financial Officer
General Counsel
Executive Vice President, Program Development
Vice President, Supply Chain
Vice President, Regulatory Affairs
Vice President, Human Resources
Vice President, Operations & Engineering
All Managers, Licensing
Responsible Manager(s)

REMOVE THIS PAGE PRIOR TO SUBMITTAL

Enclosure:

"Response to Request for Additional Information Letter No. 1 for the Review of NuScale Topical Report, TR-0515-13952, 'Risk Significance Determination,' Revision 0"

NRC RAI Number: 01 NRC RAI Date: November 12, 2015

NRC Review of: Risk Significance Determination, TR-0515-13952-NP, Revision 0

NRC RAI Question Number: 17.04-1

NRC RAI Question

NRC Standard Review Plan section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors", contained in NUREG-0800 endorses the use of numerical criteria given in NRC Regulatory Guide 1.200 for establishing the significance of contributors to risk modeled in the PRA. This includes numerical thresholds for Risk Achievement Worth (RAW) and Fussell-Vesely (FV) importance. It is stated in section 2.4 of your report that these thresholds have been applied in a preliminary NuScale PRA model with the result being that a large number of systems, structures or components (SSCs) that do not control risk were identified as being risk significant. In Section 2.2 of your report you indicate that identifying dominant contributors to risk with relative importance measures in a plant with a very low predicted risk profile, such as NuScale, is difficult and should be met with skepticism. The staff wishes to confirm that application of the proposed numerical thresholds provides reasonable results regarding identification of significant contributors to risk. Accordingly, please explain how you confirmed that some SSCs identified as being risk significant did not play a role in controlling risk. In addition, provide a list of these SSCs that includes a brief description of their functional capabilities.

NuScale RAI Question Response

Below is the list of systems from the September 2015 version of the NuScale Power, LLC (NuScale) Probabilistic Risk Assessment (PRA) that would be classified as risk-significant under the traditional Regulatory Guide 1.200 thresholds, but when using the proposed NuScale thresholds from the subject topical report (ML15211A469), are not risk-significant. This list also includes an explanation on why these systems did not play a role in controlling risk.

Note: The following information is an illustrative example based on preliminary design and analysis information.

System	Function	Explanation of Low Risk Significance
AC power (medium and low voltage)	Powers non safety-related systems and components and is a backup to the batteries for powering DC buses.	Multiple safety-related system failures (i.e., decay heat removal, reactor safety valves, and emergency core cooling) are required before AC power support for nonsafety systems would be needed.
DC power	Powers the safety-related instrumentation and control and post-accident monitoring systems, and maintains the emergency core cooling system valves in the normal standby position.	Loss of DC power results in safety-related SSCs that function to provide heat transfer to the ultimate heat sink going to their fail-safe position, and fulfilling their safety function.
Demineralized water system	Provides one source of inventory for chemical volume and control system (CVCS) injection to the reactor pressure vessel.	Demineralized water is one of two sources of inventory for CVCS, and CVCS serves a backup function for safety-related systems. Note that the CVCS is captured as risk-significant under the FV criteria for large release frequency (LRF).
Containment isolation (PRA Level-2 function only)	Closes containment isolation valves to isolate containment in the event of a core damage accident.	There are multiple barriers to prevent core damage that result in a low core damage frequency (CDF) (lower than the LRF risk-significance threshold). It is noted here that one of the five regulatory treatment of nonsafety systems (RTNSS) criteria in NUREG-0800, Section 19.3, also provides a containment performance goal.

When assessed collectively in the context of the PRA, the systems listed in the table above are shown to contribute insignificantly to risk based on a sensitivity study whereby credit is given only to the risk-significant systems, safety-related systems, and RTNSS systems; all other systems are assumed to be failed (i.e., set to true in the PRA model). The results show an increase in CDF and LRF, however, both CDF and LRF remain below the Regulatory Guide 1.174 acceptance guidelines for baseline risk and the proposed risk-significant thresholds. Therefore, these systems that would have been identified as being risk significant under the traditional Regulatory Guide 1.200 thresholds do not play a role in controlling risk for the NuScale methodology.

NRC RAI Question Number: 17.04-2

NRC RAI Question

The results of using the NuScale preliminary PRA in conjunction with the traditional values for RAW and FV specified in Regulatory Guide 1.200 to assess risk significance of SSCs are described in Section 2.4 of the report. The staff wishes to confirm that application of the proposed numerical thresholds provides reasonable results regarding identification of significant contributors to risk. In this regard: Were any studies performed to determine the sensitivity of risk categorization results to the values selected for component-level CCDF, system-level CCDF, component-level CLRF, system-level CLRF or total FV? If so, please describe the results of these studies and how they informed the selection of values reported in Table 4-1 of the report; If not, why not?

NuScale RAI Question Response

NuScale did not perform a study to determine the sensitivity of risk categorization results to the values selected for component-level conditional core damage frequency (CCDF), system-level CCDF, component-level conditional large release frequency (CLRF), system-level CLRF, or total FV.

The NuScale criteria for risk significance described in the subject topical report were developed using NRC guidance that relied on absolute-value thresholds and were not designed based on a specific set of NuScale specific risk results. By developing the thresholds from the risk-acceptance guidelines in Regulatory Guide 1.174 Revision 2, NuScale identifies SSCs that are required to maintain CDF and LERF below the safety goals as risk-significant. Implementation of these criteria is independent of design and is consistent with meeting NRC safety goals described in References 6.1.20 and 6.1.28 of the subject topical report. Since the thresholds are below the NRC safety goals, reasonableness of the results is assured. Consequently, a sensitivity study as a function of different thresholds was not conducted.

NRC RAI Question Number: 17.04-3

NRC RAI Question

Uncertainties in the PRA model are accounted for by incorporating margin into the value selected for the component level CCDF as described in Section 3.1.1.1 of the report.

a) On what basis was it concluded that the amount of margin specified in the report was sufficient to account for uncertainties in the PRA model?

b) Does the amount of margin specified in the report apply only to the uncertainties in the NuScale PRA being used to support design certification of NuScale or will it apply to the uncertainties in the PRA holders of a combined license based on the certified NuScale design are required to develop in accordance with 10 CFR 50.71(h)(1)?

NuScale RAI Question Response

a) Preliminary uncertainty results from the NuScale PRA are consistent with PRAs for commercial nuclear power plants in that the uncertainty on CDF spans approximately one order of magnitude (i.e., the range from the 5 percent value for CDF to the 95 percent value for CDF is approximately a factor of ten). Given that the risk metrics being compared to the threshold values are mean values, the span from the mean value to the 95 percent value will be less than half an order of magnitude since the mean value will be approximately a factor of three greater than the median value. Based on the observation that the typical uncertainty on CDF, and subsequently on LRF, spans an order-of-magnitude, then the range between the best estimate value and the 95 percent value is approximately a half order-of-magnitude. This is the basis for the uncertainty margin in the threshold values.

b) NuScale intends to utilize the amount of margin specified in Section 3.1.1.1 of the subject topical report to account for uncertainties in the PRA model supporting the design certification application. Although NuScale has not made a determination on applicability, current industry PRA models do not readily support significant differences in uncertainty estimates, therefore the amount of margin may be used for uncertainties in the PRA for holders of a combined license based on the certified NuScale design.

Impact of NRC RAI Question Response on "Risk Significance Determination," TR-0515-13952-NP, Revision 0:

This RAI Response does not require a revision to the subject topical report.

Section C

LO-0715-15841

July 30, 2015

Docket: PROJ0769

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

SUBJECT: NuScale Power, LLC Submittal of Topical Report TR-0515-13952, "Risk Significance Determination," Revision 0 (NRC Project No. 0769)

REFERENCES

1. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission **"**Key Issue Resolution Prior to Design Certification Application," LO-0715-16060, dated July 22, 2015

2. Presentation Slides Entitled "Design Certification Pre-Application Submittal Licensing Topic Resolution," PM-0415-13405-NP, Revision 0, dated April 21, 2015 (ML15111A203)

In the referenced letter dated July 22, 2015, NuScale Power, LLC (NuScale) provided an updated schedule for topical report submittals. Consistent with that schedule and the referenced April 21, 2015 presentation to NRC, NuScale hereby submits Topical Report TR-0515-13952, "Risk Significance Determination," Revision 0.

Standard Review Plan (SRP) Chapter 19.0, Draft Revision 3, "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 Purpose of this report is to provide NuScale's alternative definition to the term "significant" as it relates to risk significance determination. The Topical Report describes the methodology NuScale has elected to identify candidate risk-significant structures, systems and components (SSC) using probabilistic risk assessment (PRA). This methodology involves the use of alternative metrics to Regulatory Guide 1.200 for defining the term "significant".

NuScale requests NRC review and approval, under the NRC's licensing topical report program, of Enclosure 1 entitled TR-0515-13952, "Risk Significance Determination," Revision 0.

This letter and its enclosure make no regulatory commitments and no revisions to any existing regulatory commitments.

Please feel free to contact Steven Mirsky at (301) 770-0472 or at smirsky@nuscalepower.com if you have any questions.

Sincerely,

Steven Mirsky
NuScale Power, LLC
Licensing Manager, Washington DC

Distribution: Frank Akstulewicz, NRC, TWFN-6C20
Greg Cranston, NRC, TWFN-6E7
Omid Tabatabai, NRC, TWFN-6E7
Mark Tonacci, NRC, TWFN-6E7

Enclosure 1: TR-0515-13952-NP, “Risk Significance Determination,” Revision 0

Section D

Department of Energy Acknowledgement and Disclaimer

This material is based upon work supported by the Department of Energy under Award Number DE-NE0000633.

CONTENTS

TABLES

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 1×10^{-5} per year. This is in contrast to NuScale where the full-power internal events CDF is on the order of 1×10^{-7} per year.

With a baseline CDF of 1×10^{-5} per year, using the relative risk metric of risk achievement worth (RAW) of 2 implies a change in CDF of 1×10^{-5} per year. This is in contrast to NuScale, where a CDF on the order of 1×10^{-7} per year and a RAW of 2 means the CDF increases by only 1×10^{-7} per year. Consequently, when using a relative risk metric such as RAW, a plant with a baseline CDF of 1×10^{-5} per year would allow an increase in CDF of 1×10^{-5} per year, whereas a much safer plant with a baseline CDF of 1×10^{-7} would only allow an increase of 1×10^{-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 1×10^{-6} per year is considered very small or not risk-significant for a plant with a CDF on the order of 1×10^{-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-

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

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

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

Term	Definition
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

Term	Definition
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

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.

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 < 1×10^{-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 < 1×10^{-4} per reactor year).

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 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:

$$RAW = RIR = R1/Rb$$

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 1×10^{-5} per reactor year. With a baseline CDF of 1×10^{-5} per reactor year, using the *relative* risk metric of a RAW of 2 implies a change in CDF of 1×10^{-5} per year; a component is risk-significant if its CCDF is greater than 2×10^{-5} per year.

This is in contrast to NuScale, where a CDF on the order of 1×10^{-7} per year and a RAW of 2 means the CDF increases by only 1×10^{-7} per year. In this case, a component would be considered risk-significant if its CCDF is only 2 x 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 x 10^{-5} per year would allow an increase in CDF of 1 x 10^{-5} per year, whereas a much safer plant with a baseline CDF of 1 x 10^{-7} would only allow an increase of 1 x 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, conservatisms 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:

$$RAW\ interval = RII = R1 - Rb$$

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.

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 < 1×10^{-4} per reactor year and LRF < 1×10^{-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).

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.

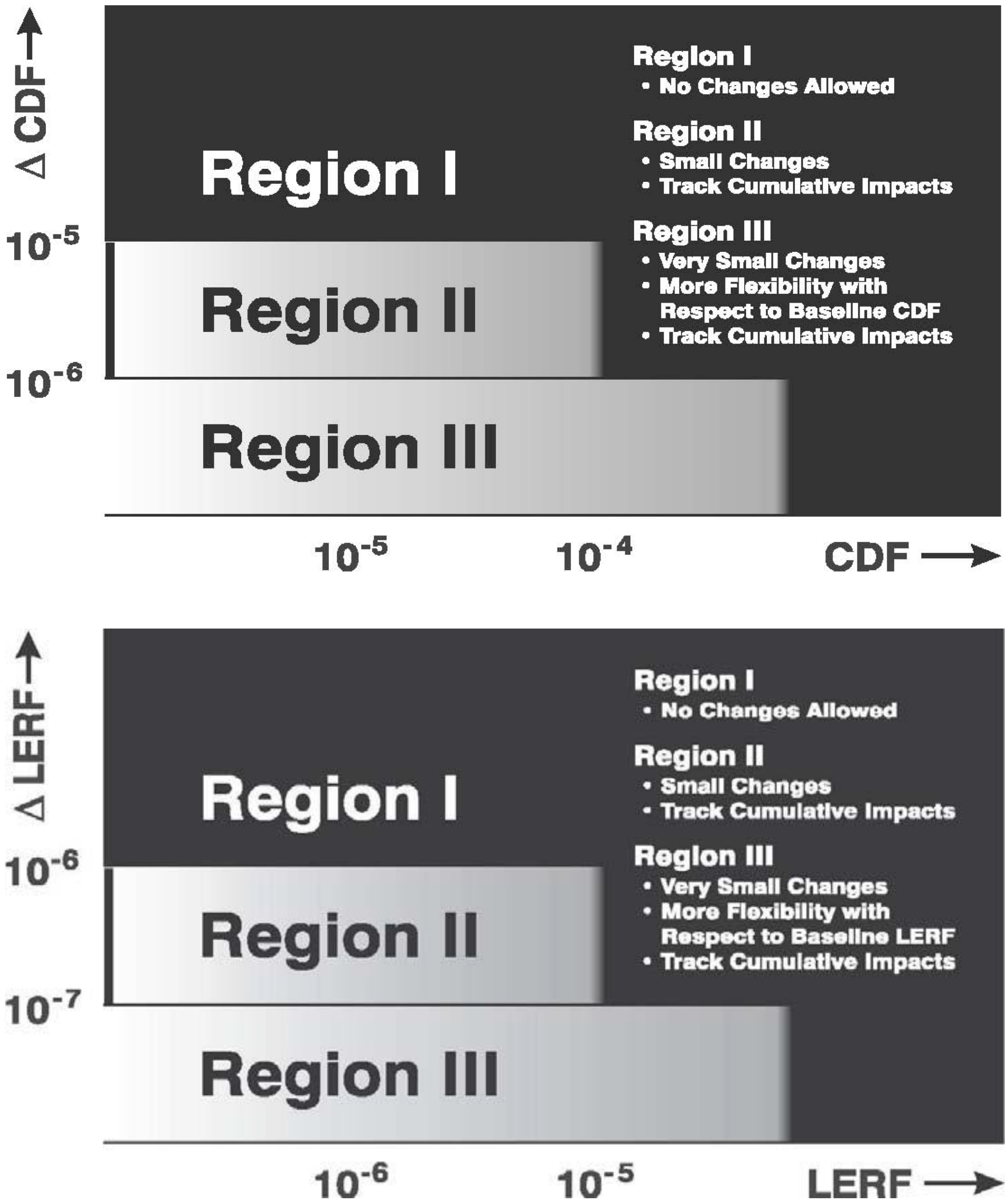

Figure 3-1. NRC Regulatory Guide 1.174 acceptance guidelines

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 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 x 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 x 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 x 10^{-4} per reactor year. However, if the baseline CDF is on the order of 1 x 10^{-7} per reactor year, an increase in CDF by a factor of 2 is not significant in relation to the 1 x 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 x 10^{-6} per year. Basically, if the failure of any component results in a CCDF of 3 x 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 [1]; 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 x 10^{-5} per year (i.e., about a half order of magnitude above the component-level threshold of 3 x 10^{-6} but within Region II of RG 1.174). If the failure of any system results in a CCDF of 1 x 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

[1] 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 x 10^{-6} was chosen by virtue of it being approximately the midpoint (on a log scale) between 1 x 10^{-5} and 1 x 10^{-6}. This value provides an order of magnitude margin to the NRC safety goal of 1 x 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 x 10^{-5} and 1 x 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 1E-6 per reactor year to 1E-5 per reactor year, applications will be considered only if it can be reasonably shown that the total CDF is less than 1E-4 per reactor year (Region II)."

The component-level threshold of 3 x 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 x 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 x 10^{-4} per year for CDF. It is also an order of magnitude lower than 1 x 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 x 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 x 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.

3.1.2.1 Basis/Justification for Large Release Frequency Criteria

The component-level value of the 3×10^{-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 1×10^{-6} per year for LRF for new reactor designs, and below the NRC safety goal of less than 1×10^{-6} for a large release.

The value 3×10^{-7} is also approximately an order of magnitude below the LERF guideline of 1×10^{-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 1×10^{-6} and 1×10^{-7} addresses making permanent changes to a plant's licensing basis:

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

The threshold of 3×10^{-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 1×10^{-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 1×10^{-6} per year for large release. It is also an order of magnitude lower than the guideline of 1×10^{-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 1×10^{-5} per year, the current FV criterion (i.e., 0.005 or 0.5 percent) translates to a CDF of 5×10^{-8} per year. For a plant like NuScale, with an expected CDF on the order of 1×10^{-7} per year, a CDF of 5×10^{-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 1×10^{-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, 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 $\geq 3 \times 10^{-6}$/yr	• Order of magnitude below NRC safety goal of 1×10^{-4}/ yr for CDF. • Order of magnitude below CDF guideline of 10^{-4}/yr used as surrogate for the cancer fatality QHO. • Consistent with RG 1.174 criteria for Region II permanent licensing basis changes: ΔCDF between 10^{-5} and 10^{-6} considered if CDF can reasonably be shown to be < 1×10^{-4}/yr. • Component-level vs. system-level in line with NEI 00-04 and RG 1.201 categorization guidelines.
System-level	CCDF $\geq 1 \times 10^{-5}$/yr	
Component-level	CLRF $\geq 3 \times 10^{-7}$/yr	• Meets NRC safety goal of 1×10^{-6}/yr for large release, meets Commission-established goal of <1×10^{-6}/yr for LRF for new reactor designs, and consistent with Commission-established goal of 0.1 for CCFP for new reactors. • Order of magnitude below LERF guideline of 10^{-5}/yr used as surrogate for the prompt fatality QHO. • Consistent with RG 1.174 criteria for Region II permanent licensing basis changes: ΔLERF between 10^{-6} and 10^{-7} considered if LERF can reasonably be shown to be < 1×10^{-5}/yr. • Component-level vs. system-level in line with NEI 00-04 and RG 1.201 categorization guidelines.
System-level	CLRF $\geq 1 \times 10^{-6}$/yr	
SSCs	Total FV $\geq$ 0.20	• Threshold consistent with criteria for operating plants (i.e., CDF of 1×10^{-5}/yr & FV of 0.005 = 5×10^{-8}/yr is approximately equal to the NuScale CDF of 1×10^{-7}/yr & FV of 0.2).

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 1×10^{-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 ≥ 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

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.

6.0 References

6.1 Referenced Documents

6.1.1 American Society of Mechanical Engineers/American Nuclear Society, *Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications*, ASME/ANS RA-Sa-2009, ASME/ANS, New York, NY, February 2, 2009.

6.1.2 Bradley, Biff, Director, Risk Assessment, Nuclear Generation Division, Nuclear Energy Institute, memorandum to Mr. Girija S. Shukla, Advisory Committee on Reactor Safeguards, "Transmission of Industry White Paper, Risk Metrics for Operating New Reactors," March 27, 2009.

6.1.3 Cheok, M. C., Parry, G. W., and Richard R. Sherry, "Use of Importance Measures in Risk-informed Regulatory Applications," *Reliability Engineering and System Safety*, 60 (1998) 213-226, July 11, 1997.

6.1.4 Electric Power Research Institute, "Advanced Nuclear Technology: Advanced Light Water Reactor Utility Requirements Document," Rev. 13, Produce ID: 3002003129, EPRI, December 2014.

6.1.5 Nuclear Energy Institute, "10 CFR 50.69 SSC Categorization Guideline," NEI 00-04, Rev. 0, July 2005.

6.1.6 Nuclear Energy Institute, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," NUMARC 93-01, Rev. 4A, April 2011.

6.1.7 Powers, Dana A., Chairman, Advisory Committee on Reactor Safeguards, memorandum to The Honorable Greta Joy Davis, Chairman, U.S. Nuclear Regulatory Commission, "Proposed Plans for Developing Risk-Informed Revisions to 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities," October 12, 1999.

6.1.8 Powers, Dana A., Chairman, Advisory Committee on Reactor Safeguards, memorandum to The Honorable Richard A. Meserve, Chairman, U.S. Nuclear Regulatory Commission, "Importance Measures Derived from Probabilistic Risk Assessments," February 11, 2000.

6.1.9 U.S. Nuclear Regulatory Commission, "Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certifications Application and Combined License Application," Interim Staff Guidance, DC/COL-ISG-028, November 2014.

6.1.10 *U.S. Code of Federal Regulations*, "Definitions," Section 50.2, Part 50, Chapter I, Title 10, "Energy," (10 CFR 50.2), 2014.

6.1.11 *U.S. Code of Federal Regulations*, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Section 50.65, Part 50, Chapter I, Title 10, "Energy," (10 CFR 50.65), 2007.

6.1.12 U.S. Nuclear Regulatory Commission, "An Approach for Plant-Specific Risk-Informed Decision making for Inservice Inspection of Piping," Regulatory Guide 1.178, Rev. 1, September 2003.

6.1.13 U.S. Nuclear Regulatory Commission, "An Approach for Plant-Specific, Risk-Informed Decision making: Inservice Testing, Regulatory Guide 1.175, August 1998.

6.1.14 U.S. Nuclear Regulatory Commission, "An Approach for Plant-Specific, Risk-Informed Decision-Making: Technical Specifications," Regulatory Guide 1.177, Rev. 1, May 2011.

6.1.15 U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Rev. 2, May 2011.

6.1.16 U.S. Nuclear Regulatory Commission, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Rev. 2, March 2009.

6.1.17 U.S. Nuclear Regulatory Commission, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationships to Current Regulatory Requirements," SRM to SECY-90-016, June 26, 1990.

6.1.18 U.S. Nuclear Regulatory Commission, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Regulatory Guide 1.201 (for trial use), Revision 1, May 2006.

6.1.19 U.S. Nuclear Regulatory Commission, "History of the Use and Consideration of the Large Release Frequency Metric by the U.S. Nuclear Regulatory Commission," SECY-13-0029, March 22, 2013.

6.1.20 U.S. Nuclear Regulatory Commission, "Implementation of the Safety Goals," SRM on SECY-89-102, June 15, 1990.

6.1.21 U.S. Nuclear Regulatory Commission, "Modifying the Risk-Informed Regulatory Guidance for New Reactors," SECY-10-0121, September 14, 2010.

6.1.22 U.S. Nuclear Regulatory Commission, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," Policy Statement, Federal Register Vol 50 FR 32138, August 8, 1985.

6.1.23 U.S. Nuclear Regulatory Commission, "Policy Statement on the Regulation of Advanced Reactors," Final Policy Statement, Federal Register Vol 73 FR 60612, October 14, 2008.

6.1.24 U.S. Nuclear Regulatory Commission, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors,” NUREG-0800, Chapter 19, Section 19.0, Draft Rev. 3, November 2014.

6.1.25 U.S. Nuclear Regulatory Commission, “Reliability Assurance Program,” NUREG-0800, Chapter 17, Section 17.4, Draft Rev. 1, May 2013.

6.1.26 U.S. Nuclear Regulatory Commission, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance,” NUREG-0800, Chapter 19, Section 19.2, June 2007.

6.1.27 U.S. Nuclear Regulatory Commission, “Risk-Informed Regulatory Framework for New Reactors,” SECY-12-0081,” June, 2012.

6.1.28 U.S. Nuclear Regulatory Commission, “Safety Goals for the Operation of Nuclear Power Plants; Policy Statement,” Correction and Republication, Federal Register Vol 51 FR 30028, August 21, 1986.

6.1.29 U.S. Nuclear Regulatory Commission, “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement,” Federal Register Vol. 60 No. 158, August 16, 1995.

6.1.30 U.S. Nuclear Regulatory Commission, “Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design,” NUREG-1966, Volume 4, April 2014.

6.1.31 U.S. Nuclear Regulatory Commission, “Measures of Risk Importance and Their Applications,” NUREG/CR-3385, May 1986.

6.1.32 U.S. Nuclear Regulatory Commission, “Regulatory Treatment of Non-Safety Systems for Passive Advanced Light Water Reactors,” Chapter 19, Section 19.3, NUREG-0800, Draft Rev. 0, October 2012.

6.1.33 U.S. Nuclear Regulatory Commission, “Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (WASH-1400),” NUREG-75/014, October 1975.

6.1.34 Vesely, W.E., “The Use of Risk Importances for Risk-Based Applications and Risk-Based Regulations,” *Proceedings of PRA ’96*, Park City, Utah, September 29–October 3, 1996.

6.1.35 Stetkar, John W., Chairman, Advisory Committee on Reactor Safeguards, memorandum to Mr. Mark A Satorious, Executive Director for Operations, U.S. Nuclear Regulatory Commission, “Standard Review Plan Chapter 19 and Section 17.4,” July 16, 2014.