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 Relevant Guidance

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

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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 1×10^{-5} per year and the current estimate of CDF for NuScale is 1×10^{-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.