UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

May 18, 2016

Mr. Victor M. McCree
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: NUSCALE POWER, LLC LICENSING TOPICAL REPORT, "RISK SIGNIFICANCE DETERMINATION"

Dear Mr. McCree:

During the 634th meeting of the Advisory Committee on Reactor Safeguards (ACRS), May 5-6th, 2016, we reviewed the NuScale Power, LLC (NuScale) licensing topical report, "Risk Significance Determination." Our Future Plant Design Subcommittee reviewed this matter on March 1, 2016. During these reviews, we had the benefit of discussions with the NRC staff and representatives of NuScale. We also had the benefit of the referenced documents.

CONCLUSION AND RECOMMENDATIONS

1. The staff continues to deal with criteria for determining risk significance in a case-by-case manner and this can lead to inconsistencies in regulatory positions.

2. The staff should develop a consistent approach by adopting a continuous scale to determine quantitative risk significance criteria, with more margin allowed for plants with lower risk.

3. The approach proposed by NuScale is reasonable provided that the core damage frequency (CDF) or large release frequency (LRF) remains consistent with their current estimates in the licensing topical report.

4. The staff will need to address the multi-module aspects of the NuScale design that could alter the CDF and LRF risk estimates and associated structures, systems, and components (SSCs) classification.

BACKGROUND

NuScale has elected to identify candidate risk significant SSCs using their probabilistic risk assessment (PRA). This candidate list of SSCs would be used as input for the NuScale Design Reliability Assurance Program (DRAP). The proposed methodology involves the use of alternative metrics to Regulatory Guide 1.200 metrics for defining the term "significant" and is subject to staff review and approval.

The traditional Regulatory Guide 1.200 criteria, including those referenced in the Standard Review Plan, Section 19.0, are based on relative risk metrics and the risk profiles of operating nuclear power plants. Because of the NuScale passive design, the risk is expected to be significantly lower than that of current light water reactors. The traditional relative criteria are insensitive to such global improvements in safety. While NuScale design features reduce the frequency of accident sequence types that dominate the risk for current plants, continued use of the same relative risk criteria in reactor designs such as NuScale would artificially raise the relative importance of SSCs that do not contribute meaningfully to risk.

DISCUSSION

For current light water reactors, the full-power internal events mean CDF is on the order of $1x10^{-05}$ per year. While the complete NuScale PRA results are not available and have not been reviewed, the NuScale topical report expects the total CDF to be on the order of $1x10^{-07}$ per year and estimates the full power internal events CDF to be less than $1x10^{-07}$ per year.

Given a nominal CDF of $1x10^{-05}$ per year and using the Regulatory Guide 1.200 relative risk metric of risk achievement worth (RAW) of 2, this implies that a change in CDF of $1x10^{-05}$ per year is the criterion for determining risk significance. This is in contrast to NuScale, where a CDF on the order of $1x10^{-07}$ per year and using a RAW of 2 means the allowable change in CDF can only increase by $1x10^{-07}$ per year. Consequently, when using a relative risk metric such as RAW, a plant with a CDF of $1x10^{-05}$ per year would allow an increase in CDF of $1x10^{-05}$ per year, whereas a plant design with a CDF of $1x10^{-07}$ would only allow an increase of $1x10^{-07}$ per year before a particular SSC becomes risk-significant. Therefore, using the Regulatory Guide 1.200 relative risk criteria to identify risk significant SSCs would result in categorizing a majority of NuScale equipment modeled in the PRA as risk-significant. This approach does not account for the lower risk of the NuScale design, and the significant reduction in risk compared to current light water reactors.

In contrast to the Regulatory Guide 1.200 relative risk criteria, the NuScale proposed approach is justified based on Regulatory Guide 1.174 concepts. The basis of the implementation is predicated on the assumption that the CDF for a single NuScale module for all modes of operation is quite low with a value of $1x10^{-07}$ per year used as the estimate in this topical report. In principle, the NuScale proposed approach is in line with the ACRS letter recommendations of July 16, 2014. It is NuScale specific, however, and it does not adopt the continuous scale approach that was suggested in our letter of April 26, 2012.

The NuScale approach proposes that a component is considered significant to risk if its failure results in a CDF that exceeds $3x10^{-06}$ per year. Similarly the component would be considered significant to risk if its failure results in a LRF that exceeds $3x10^{-07}$ per year. It should be noted that the NuScale rationale for setting the LRF criterion is not conceptually consistent with the "margins" that are used for the CDF criterion, i.e., a factor of 10 below the regulatory acceptance value, plus an additional factor of 3 for "model uncertainty."

The staff found the NuScale use of absolute risk metrics to determine risk significance to be an acceptable approach. The staff did not approve these specific values for importance measures, as part of their review of the topical report. Rather, the staff expects to review the specific candidate list of SSCs for the DRAP later and render their judgment at that time.

We disagree with the staff's approach. The staff continues to deal with criteria for determining risk significance in a case-by-case manner and this can lead to inconsistencies in regulatory positions. NuScale's fundamental purpose for seeking approval of their proposed construct through this topical report is to gain assurance of regulatory stability and consistency when they submit their design for review. If the staff will re-examine the NuScale risk significance determination process for applications such as the DRAP and 10 CFR 50.69, what is the purpose for accepting the specific construct that NuScale proposes in the topical report?

The approach proposed by NuScale is reasonable provided that the CDF and LRF after completion of a comprehensive probabilistic risk assessment remain consistent with current estimates. However, if the CDF and LRF are found to be significantly higher than currently estimated and used in the topical report, NuScale and the staff do not have a logical and consistent framework to adjust the quantitative risk significance criteria.

The staff should adopt a continuous scale to determine quantitative risk significance criteria, with more margin allowed for plants with lower risk. This approach should be consistent with the logic outlined in our April 26, 2012 letter and our July 16, 2014 letter on risk-informed regulatory framework for new reactors. Such a framework is a logical extension of the quantitative guidance found in Regulatory Guide 1.174. It also preserves a consistent, objective, and reproducible basis for regulatory oversight decisions for current reactors, as well as new reactor designs like NuScale. This approach would use a continuous quantitative scale of increasing flexibility and a method to measure consistently whether a system or component is deemed 'risk significant'. An example of such a continuous scale is provided in our April 26, 2012 letter.

NuScale also proposes to use the Fussell-Vesely (FV) importance metric. This is a measure of the fractional contribution to risk. Any SSC or other entity modeled in the PRA that contributes 20% or more to risk is considered a risk-significant candidate (i.e., FV greater than or equal to 0.20). It is important to note that this FV criterion, as described in the NuScale topical report, is applied on a hazard-by-hazard basis for each mode of plant operation (e.g., internal events at power, internal events at shutdown, fires at power, seismic events at shutdown, etc.). However, one can identify applications of this criterion that result in inappropriate conclusions. For example, there may be instances where SSC X satisfies the 20% criterion for internal events at power, and is thereby included in the candidate DRAP list. However, SSC Y does not satisfy that criterion as applied on a hazard-by-hazard basis, and is excluded from the candidate DRAP list. This may occur despite the fact that SSC Y FV importance to overall CDF is higher than that for SSC X. This has led some risk analysts to consider a two-tier approach for this type of metric, e.g., a 20% screen at the hazard-by-hazard level and a 10% screen at the total CDF or LRF level.

The staff found the NuScale process of deriving a FV criterion to be acceptable because it is consistent with that used for current operating reactors. However, the staff established a limitation that if the actual CDF for any of the specific hazards treated in the certified NuScale design is substantially different from that assumed, then the associated FV threshold needs to be reexamined. We concur with this limitation.

The NuScale conceptual design envisions six to twelve reactor modules at a site. At any time, some of the modules could be operating, while others might be experiencing outages, refueling, or installation. The modules share common support systems, as well as a common water pool that serves as a radiological shield and ultimate heat sink. Failure contributions of SSCs and human actions to the risk of this complete plant complex, may be different from those identified by examining the risk arising from a single module. This is an important topic and must be addressed before the DRAP list can be considered complete.

Complete plant site risk (i.e., "multi-module risk") is not addressed in NuScale's topical report or the staff's safety evaluation report. We focus here on the subject of employing a consistent approach for evaluating risk significance, i.e., a continuous scale, because adopting such an approach now can facilitate the review of the NuScale design and will provide a consistent approach for review of other reactor designs. The staff has acknowledged the potential importance of complete plant risk. We will address it later in our review process.

Dr. Rempe and Dr. Riccardella did not participate in the discussions or deliberations.

Sincerely,

/RA/

Dennis C. Bley
Chairman

REFERENCES

1. NuScale Power, LLC, Licensing Topical Report TR-0515-13952-NP, "Risk Significance Determination," July 2015 (ML15211A470).

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009 (ML090410014).

3. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.0, December 2015 (ML15089A068).

4. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, May 2011 (ML100910006).

5. Advisory Committee on Reactor Safeguards, "Standard Review Plan Chapter 19 and Section 17.4," July 16, 2014 (ML14196A119).

6. Advisory Committee on Reactor Safeguards, "Draft Commission Paper, 'Risk Informed Regulatory Framework for New Reactors'," April 26, 2012 (ML12107A199).

7. U.S. Nuclear Regulatory Commission, Draft Safety Evaluation Report, “Advanced SER with ACRS Subcommittee Comments Incorporated for NuScale Power Licensing Topical Report: TR-0515-13952-NP, Revision 0, ‘Risk-Significance Determination’,” April 21, 2015 (ML16111A758).

8. NuScale Power, LLC, “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’,” December 11, 2015 (ML15348A369).

7. U.S. Nuclear Regulatory Commission, Draft Safety Evaluation Report, “Advanced SER with ACRS Subcommittee Comments Incorporated for NuScale Power Licensing Topical Report: TR-0515-13952-NP, Revision 0, ‘Risk-Significance Determination’,” April 21, 2015 (ML16111A758).

8. NuScale Power, LLC, “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’,” December 11, 2015 (ML15348A369).

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