Response to SDAA Audit Question

Question Number: A-19.2.6-1

Receipt Date: 03/27/2023

Question:

Based on the acceptance review of Section 19.2.6, the staff has the following concerns which have been communicated previously to NuScale:

1. 10 CFR 51.22(c)(22) makes issuing a SDA a categorical exclusion and does not require a SAMDA cost-benefit analysis to meet the agency's NEPA obligation. However, material in Section 19.2 of the US460 Safety Analysis Report (SAR) appears indicative of aspects of an environmental review under 10 CFR Part 51.

2. Section 19.2.6 could be incorporated by reference in a NuScale COLA submittal. Based on the concern in item #1 above, the staff will not have the opportunity to review the SAMDA submittal under 10CFR52.79(c)(1).

The staff proposes the following to resolve its concerns:

1. NuScale explicitly states in a docketed correspondence that: (i) NuScale is requesting a finding for the Section 19.2.6 material only against 50.34(f), specifically 50.34(f)(1)(i), and (ii) that a finding against 10 CFR Part 51 is not requested.

2. NuScale updates the US460 SAR as follows:
(i) The references to SAMDAs are deleted from Section 19.1 (e.g., page 19.1-1 and Section 19.1.3.4).
(ii) In Section 19.2.6, all references to SAMDAs are either deleted or changed to "improvements" (the term used in 10 CFR 50.34(f)(1)(i)).
(iii) COL Item 19.2-3 is either removed or clarified to apply only for demonstrating COLA compliance with 10 CFR 50.34(f), specifically 10 CFR 50.34(f)(1)(i).

Response:

As stated in FSAR Section 19.2.6, the content therein is provided to demonstrate compliance with, and only with, 10 CFR 50.34(f)(1)(i), as applicable to the standard design approval application (SDAA) via 10 CFR 52.137(a)(8).

While severe accident mitigation design alternative (SAMDA) is a term of art for an applicant's environmental report and NRC's environmental review, the term also has broader implications. In its "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," the Commission mandated that design approvals address the TMI requirements (then only applicable to pending construction permit applications), and "the applicant for approval or certification of a reference design shall consider a <u>range of alternatives</u> and combination of alternatives to address the unresolved and generic safety issues and to search for <u>cost-effective reductions in the risk from severe accidents</u>." That direction eventually became the Part 52 rules that require applicants to address 50.34(f)(1)(i), et al.

Thus, NuScale intended "SAMDAs" in SDAA Chapter 19 not to refer to the environmental review, but to the issue of severe accident design alternatives required to be considered separately from an environmental review. Nevertheless, NuScale recognizes the confusion this has caused and agrees that the application can be made clearer by referring to severe accident design "improvements" as used in 50.34(f)(1)(i) rather than SAMDAs. References to SAMDAs in Sections 19.1 and 19.2 have been revised accordingly.

NuScale agrees that COL Item 19.2-3 is not necessary and could cause confusion for the SDAA, and has been deleted.

Markups of the affected changes, as described in the response, are provided below:

Table 1.8-1: Combined License Information Items (Continued)

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Item No.	Description of COL Information Item	Section
~~COL Item 19.2-3:~~	~~An applicant that references the NuScale Power Plant US460 standard design will evaluate severe accident mitigation design alternatives screened as not required for the standard design.~~	~~19.2~~
COL Item 19.3-1:	An applicant that references the NuScale Power Plant US460 standard design will identify site-specific Regulatory Treatment of Nonsafety Systems structures, systems, and components and applicable process controls.	19.3

CHAPTER 19 PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION

19.1 Probabilistic Risk Assessment

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The PRA is performed consistent with the requirements of 10 CFR 52.137(a)(25). It assesses the risk for a single NuScale Power Module (NPM) and includes Level 1 and Level 2 evaluations. The PRA follows the guidance in interim staff guidance (ISG) DC/COL-ISG-028 (Reference 19.1-3). This ISG applies to a standard design as an acceptable approach to conforming with American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-S-2008 (Reference 19.1-1) and addenda ASME/ANS RA-Sa-2009 (Reference 19.1-2), as endorsed by Regulatory Guide (RG) 1.200, Revision 3. The PRA supporting the standard design does not include a Level 3 evaluation (although NuScale performed a limited offsite consequence assessment to support the evaluation of potential design improvements in Section 19.2.6~~severe accident management design alternatives [SAMDA] analysis~~).

When addressing general concepts, the term "PRA" refers collectively to the Level 1 and Level 2 risk metric evaluation as well as the phenomenological evaluation of severe accident response. Because of a small radionuclide inventory in a single module compared to typical, currently operating plants, risk metrics associated with small modular reactors have different implications for public health and safety. To reflect this perspective, and to clarify that the calculated risk metric values are based on a PRA for a single module, this chapter uses the terms core damage frequency (CDF) and large release frequency (LRF) to present results for CDF and large release frequency calculations for a single module. When referring to multi-module (MM) risk metrics, the chapter uses terms "multi-module core damage frequency" (MM-CDF) and “multi-module large release frequency” (MM-LRF). The conditional containment failure probability (CCFP) refers to the risk metric associated with failure of a containment vessel (CNV), which houses a reactor pressure vessel (RPV). Together, the CNV and RPV comprise the NPM.

The PRA evaluates the risk associated with operation of a single module at full power as well as low power and shutdown (LPSD) modes of operation for both the internal and the external initiating events (IEs) that can be addressed at the standard design stage. NuScale assesses the risk associated with multi-module operation using a systematic approach that includes both a qualitative evaluation of the potential impact of shared systems and a quantitative assessment based on the single-module, full-power, internal-events PRA to identify potential multi-module risk contributors.

This section summarizes key aspects of the PRA and associated insights. Supporting documentation including fault trees, initiating and basic event frequency calculations, human error calculation worksheets, and success criteria modeling is available to support U.S. Nuclear Regulatory Commission (NRC) reviews and audits.

19.1.3.4 Uses of Probabilistic Risk Assessment in the Design Process

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The design was developed in consideration of issues associated with typical currently operating plants. Thus, there are several design features inherent to the design that address characteristics of currently operating plants related to operational risk. Table 19.1-2 summarizes these features, which contribute to a low risk profile. The PRA was used to further reduce the risk profile by evaluating design options during the design process. Table 19.1-3 summarizes key design decisions that were supported by PRA analyses. Further, evaluation of potential design improvements~~SAMDA~~, as described in Section 19.2.6, is supported by PRA analyses.

19.1.4 Safety Insights from the Internal Events Probabilistic Risk Assessment for Operations at Power

This section discusses the internal events PRA for a single NPM operating at full power.

19.1.4.1 Level 1 Internal Events Probabilistic Risk Assessment for Operations at Power

Internal events, within the scope of the PRA, are those events that originate within the plant boundary that directly or indirectly perturb the steady-state operation of the plant and could lead to an undesired plant condition.

This section summarizes the Level 1 PRA (i.e., risk assessment associated with core damage) associated with operation of a single NPM. The full-power PRA addresses the risk associated with operation in Technical Specification Mode 1 (Operations).

19.1.4.1.1 Description of the Level 1 Probabilistic Risk Assessment for Operations at Power

The following sections address the methodology, data, and analytical tool used to perform the full power, internal events Level 1 PRA.

19.1.4.1.1.1 Methodology

NuScale constructed the PRA by first developing a representative spectrum of potential internal initiating events as discussed in Section 19.1.4.1.1.2. For each initiating event, a "Level 1" event tree illustrates the sequence logic for the module response. This logic illustrates module response to an initiating event by identifying appropriate "top events." The top events represent systems that can mitigate the respective initiating event, either by themselves or in combination with other systems. The top events of the event trees, presented in Section 19.1.4.1.1.4, include safety-related and nonsafety-related mitigating systems.

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Table 19.1-1: Uses of Probabilistic Risk Assessment at the Design Phase

Use	Applicable Section
Identify dominant risk contributors	Section 19.1.4, Section 19.1.5, Section 19.1.6, Section 19.1.7
With regard to capability in comparison to currently operating plants: • Address significant risk contributors of currently operating plants • Demonstrate that the design addresses known issues related to the reliability of core and containment heat removal systems at some operating plants (i.e., the additional Three Mile Island-related requirements in 10 CFR 50.34(f)) • Evaluate whether plant design, including potential effect of site-specific characteristics, represents a reduction in risk compared to currently operating plants	 • Section 19.1.3 • Section 19.2.6 • Section 19.1.3
Evaluate design robustness and tolerance of severe accidents	Section 19.2
Evaluate risk-significance of human error including a characterization of the significant human errors that may be used as an input to operator training programs and procedure refinement	Section 19.1.4, Section 19.1.5
Evaluate conformance with NRC safety goals	Section 19.1.4, Section 19.1.9
Assess the balance of preventive and mitigative features and consistency with SECY-93-087 (1993) and associated staff requirement memorandum	Section 19.2.2
Support Design Reliability Assurance Program including RTNSS classification of structures, systems, and components	Section 17.4, Section 19.3
Potential design improvements~~Severe Accident Management Design Alternatives~~	Section 19.2.6
Support Regulatory Oversight Processes, for example, • Mitigating Systems Performance Index • Significance Determination Process	Section 19.1.8
Technical Specifications support • Design-specific surveillance frequencies • Criterion 4 of 10CFR50.36(c)(2)(ii)(D)	Section 19.1.8
Maintenance Rule (SSC classification)	Section 17.6
Human performance insights	Chapters 18, 19

<u>Minimize Off-site Releases</u>

The small size of an NPM core results in a correspondingly small radionuclide source term. Although not credited in the PRA, potential releases would be further minimized because

- most of the CNV is below water; thus, radionuclide release due to CNV failure of the lower head would be minimized because of the scrubbing effect of the reactor pool.
- for severe accidents with CNV bypass or containment isolation failure, there is potential deposition in the bypass piping, and the release would potentially be further reduced by the RXB.

19.2.5.2 Accident Management Programmatic Structure

The programmatic structure of management of severe accidents occurring in an NPM reflects lessons learned from industry experience and recent developments in severe accident response, specifically:

- Accident mitigation focuses on the containment of fission products. When an accident can no longer be mitigated by emergency operating procedures (EOPs), activities transition to severe accident management guidelines (SAMGs) or other administrative controls. Section 13.5 addresses EOPs and other operating procedures.
- The response to an ATWS defined by 10 CFR 50.62 is addressed in SAMGs or other administrative controls. Section 19.2.2.1 summarizes the NPM capability to accommodate an ATWS event.
- The response to an SBO defined by 10 CFR 50.63 is addressed in SAMGs or other administrative controls. Section 19.2.2.3 summarizes the NPM capability to accommodate an SBO and related events.
- The response to an aircraft impact event defined in 10 CFR 50.150 is addressed in SAMGs or other administrative controls. Section 19.5 addresses key design features associated with the design capability to survive an aircraft impact.
- Strategies and guidelines for mitigation of beyond-design-basis events, as required by 10 CFR 50.155, are addressed in Section 19.4.

COL Item 19.2-1: An applicant that references the NuScale Power Plant US460 standard design will develop severe accident management guidelines and other administrative controls to define the response to beyond-design-basis events.

19.2.6 Consideration of Potential Design Improvements Under 10 CFR 50.34(f)

Audit Issue A-19.2.6-1

As described in prior sections, the design-specific PRA performed is consistent with the requirement in 10 CFR 50.34(f)(1)(i) to identify improvements in the reliability of core and containment heat removal systems that are significant and practical. ~~The potential improvements considered are identified as severe accident management~~

~~design alternatives (SAMDAs).~~ The following sections summarize the method for identifying and evaluating these design improvements~~alternatives~~ and the conclusions of the ~~SAMDA~~ evaluation.

19.2.6.1 Introduction

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The design improvement~~SAMDA~~ analysis is a cost-benefit analysis wherein the cost of modifying the nuclear power plant design is weighed against the monetized estimation of risk associated with the consequences stemming from a possible severe accident.

19.2.6.2 Estimate of Risk for Design

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The estimate of the risk that provides the basis for the design improvement ~~SAMDA~~ evaluation is developed from the PRA performed for the standard design and an estimate of the characteristics of a potential site. Key points of the evaluation include.

- The PRA provides Level 1 and Level 2 information for all modes of operation. In addition to full power, low power, and shutdown internal events, the PRA addresses internal flood, internal fire, high winds, external flooding, and seismic hazard.
- Site characteristics are based on the State-of-the-Art Reactor Consequence Analysis Project Surry Nuclear Power Station off-site consequence model in NUREG/CR-7110 (2013), updated with 2022 economic information and 2060 population estimates, which are considered representative for the purposes of the design improvement~~SAMDA~~ evaluation for standard design.
- To determine the offsite dose and economic consequences required for the calculation of the cost of maximum benefit, the two release categories identified in the Level 2 PRA are redefined into three release categories to more realistically estimate the offsite consequences of severe accidents. Radionuclide source terms corresponding to each release category are determined with MELCOR severe accident simulations.
- The MACCS code in NUREG/CR-6613 (1998) is used to evaluate the population dose and off-site economic consequences.
- Onsite operational dose estimates and cleanup and decontamination cost estimates are used from NUREG/BR-0184 (1997).
- Multiple-module events are addressed by applying multipliers (corresponding to the maximum number of NPMs that could be involved in an accident corresponding to each release category) to the severe accident effects when evaluating the maximum benefit of a design improvement~~alternative~~.

Following the guidance in NEI 05-01 (Reference 19.2-13), the maximum benefit associated with eliminating all risk in the design (which can be viewed as an estimate of the severe accident risk for the design) is conservatively calculated to be $110,000 for a 6-NPM configuration. This maximum benefit is bounding for a configuration with a smaller number of NPMs.

19.2.6.3 Identification of Potential Design Improvements

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The design improvement~~SAMDA~~ evaluation is performed using the guidance in NEI 05-01 and NUREG/BR-0184. Design improvements~~alternatives~~ include those typically considered for currently operating pressurized water reactor plants and those that may be beneficial to the design. Design improvements~~alternatives~~ specific to the NuScale design are identified to improve the reliability of the SSC determined to be candidates for risk-significance; in some cases a generic design improvement~~SAMDA~~ is applicable to the risk-significant candidate component, but in most cases, a design-specific design improvement~~SAMDA~~ is identified and evaluated. The design improvement~~SAMDA~~ candidates are considered based on the generic list provided in NEI 05-01 and NuScale-specific design considerations.

NuScale design-specific design improvement~~SAMDA~~ candidates are postulated for a variety of plant systems:

- chemical and volume control system
- containment flooding and drain system
- containment system
- control rod drive system
- decay heat removal system
- emergency core cooling system
- augmented DC power system (EDAS)
- module protection system
- Reactor Building crane system
- reactor coolant system
- reactor trip system

19.2.6.4 Risk-Reduction Potential of Design Improvements

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The candidate design improvements ~~SAMDAs~~ identified are qualitatively screened into one of seven initial screening categories. The intent of the screening is to identify the candidates with the potential for risk reduction in the design that warrant a detailed cost-benefit evaluation. The categories and the screening process itself are based on the "Phase I" analysis screening criteria in NEI 05-01; the categories are:

- Not applicable: Design improvement~~SAMDA~~ candidates that are not considered applicable to the design are those with specific pressurized water reactor equipment that is not in the design.
- Already implemented: Candidate design improvements ~~SAMDAs~~ that are already included in the design or whose intent is already fulfilled by a different design feature are considered "already implemented" in the design. If a particular design improvement~~SAMDA~~ is already implemented in the design, it is not retained for further analysis.
- Combined: The design improvement~~SAMDA~~ candidates that are similar to one another are combined and evaluated in conjunction with each other. This combination of design improvement~~SAMDA~~ candidates leads to a more comprehensive or plant-specific ~~SAMDA~~ candidate set. The combined candidate set~~SAMDA~~ would then be assessed against the remaining six screening categories.
- Excessive implementation cost: If a design improvement~~SAMDA~~ requires extensive changes that exceed the value shown in Section 19.2.6.2 even without an implementation cost estimate, it is not retained for further analysis.
- Very low benefit: If a proposed design improvement~~SAMDA~~ is related to a system for which improved reliability would have a negligible impact on overall plant risk, it is judged to have a very low benefit and is not retained for further analysis.
- Not required for standard design: Design improvement~~SAMDA~~ candidates related to potential procedural enhancements, surveillance action enhancements, multiple plant sites, or design elements that are to be finalized in a later stage of the design process are outside of the scope of this report.
- Considered for further evaluation: Any design improvement~~SAMDA~~ candidate that did not screen into any of the previous six screening categories is subject to a more in-depth cost-benefit analysis.

19.2.6.5 Cost Impacts of Candidate Design Improvements

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A total of 22 design improvements ~~SAMDAs~~ are screened into the "excessive implementation cost" or "considered for further evaluation" categories. Of the 22 design improvements ~~SAMDAs~~, one is screened in Phase I as exceeding the maximum benefit value shown in Section 19.2.6.2 for the design. The remaining Phase I candidates considered to be potentially cost beneficial are evaluated further in Phase II.

19.2.6.6 Cost-Benefit Comparison

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The contribution to the maximum benefit for each design improvement~~SAMDA~~ evaluated in Phase II is below the estimated cost of implementation of greater than $100,000 for each design improvement~~SAMDA~~ candidate. Therefore, none of the candidates are considered to be potentially cost beneficial in the Phase II screening.

Maximum benefit sensitivity analyses are performed using different assumptions of on-site dose, dollar per person-rem conversion factor, discount rate, and off-site consequence modeling assumptions for the release characteristics, site characteristics, and emergency planning characteristics.

19.2.6.7 Conclusions of Design Improvement~~Severe Accident Mitigation Design Alternative~~ Evaluation

Audit Issue A-19.2.6-1

Design improvements~~alternatives~~ that are considered in the ~~SAMDA~~ evaluation include those typically considered for currently operating plants and those that may be beneficial to the design. There are no design improvements~~alternatives~~ determined to be cost-beneficial for severe accident mitigation.

COL Item 19.2-2: An applicant that references the NuScale Power Plant US460 standard design will use the site-specific probabilistic risk assessment to evaluate and identify improvements in the reliability of core and containment heat removal systems as specified by 10 CFR 50.34(f)(1)(i).

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COL Item 19.2-3: ~~An applicant that references the NuScale Power Plant US460 standard design will evaluate severe accident mitigation design alternatives screened as not required for the standard design.~~

19.2.7 References

19.2-1 American Society of Mechanical Engineers/American Nuclear Society, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME/ANS RA-S-2008 (Revision 1 RA-S-2002), New York, NY.

19.2-2 ANSYS (Release 19.2) [Computer Program]. (2019). Canonsburg, PA, ANSYS Incorporated.

19.2-3 Theofanous, T.G., et al., "In-vessel Coolability and Retention of a Core Melt," DOE/ID-10460, Vol. I, October 1996.

19.2-4 Rempe, J.L., "Potential for AP600 In-Vessel Retention through Ex-Vessel Flooding," INEEL/EXT-97-00779, December 1997.

19.2-5 Z. Guo and M.S. El-Genk, "An experimental study of saturated pool boiling from downward facing and inclined surfaces," *International Journal of Heat Mass Transfer*, (1992): 35: 9, 1992.

19.2-6 Theofanous, T.G. and S. Syri, "The coolability limit of a reactor pressure vessel lower head," *Nuclear Engineering and Design*, (1997): 169: 1-3:59-76.

19.2-7 Theofanous, T.G., et al., "Critical heat flux through curved, downward facing, thick walls," *Nuclear Engineering and Design*, (1994): 151: 1:247-258.

19.2-8 Kutateladze, S. "On the transition to film boiling under natural convection", Kotloturbostronie, no. 3, p. 10, 1948.

19.2-9 Kutateladze, S. "Heat Transfer in Condensation and Boiling," Tech. Rep., State Scientific and Technical Publishers of Literature on Machinery, 1952.

19.2-10 Seongchul Jun et. al. "Effect of Subcooling on Pool Boiling of Water from Sintered Copper Microporous Coating at Different Orientations", Science and Technology of Nuclear Installations, 2018