RULEMAKING ISSUE
NOTATION VOTE

RESPONSE SHEET

TO: Annette Vietti-Cook, Secretary

FROM: Chairman Hanson

SUBJECT: SECY-21-0004: Proposed Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029)

Approved XX Disapproved ____ Abstain ____ Not Participating ____

COMMENTS: Below _____ Attached XX None _____

Christopher T. Hanson

Digitally signed by Christopher T. Hanson
Date: 2021.03.24 13:38:19 -04'00'

Signature
Christopher T. Hanson

Date 03/24/2021

Entered in STARS
Yes ✓
No ______

Chairman Hanson's Comments on SECY-21-0004: Proposed Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029)

First and foremost, I want to congratulate the staff for its successful completion of the NuScale design review. As the first design certification for a small modular reactor, this is a major accomplishment for the agency, and I would personally like to thank every NRC employee who contributed to this effort. The staff addressed many unique policy and technical issues with a risk-informed and safety-focused mindset, resolving first-of-a-kind design and safety approaches in many areas of the review. I commend the staff for its regulatory and technical excellence, along with its commitment to the NRC's important safety, security, and environmental mission. I approve the staff's recommendation to publish the proposed rule to certify the NuScale standard design, subject to the attached edits and with caveats for future applications.

I want to briefly touch on the three unresolved technical issues in this rulemaking. Faced with a deficit of information, the staff left unresolved three issues after determining that they do not affect other aspects of the review and that the missing information can be provided by combined license applicants without significant impacts on safety or standardization. While I appreciate the staff's initiative in seeking a solution in this circumstance, I hope to see this approach used very sparingly in the future, if at all. With appropriate and timely information from the applicant, these issues would have been resolved at the design certification stage, as intended by our regulatory structure in part 52. The staff needs sufficient and timely information to determine whether a design meets the applicable standards and requirements of the Commission's regulations. This principle will be incredibly important as the staff prepares to review a spectrum of advanced reactor applications, which will certainly raise even more novel and unique design issues. Applicants need to ensure the availability of quality, complete, and timely information to the staff to support effective and efficient reviews.

Finally, the appropriate treatment of uncertainties in probabilistic risk assessment (PRA) models is an important issue for the NRC to consider in its risk-informed decision-making. These uncertainties are derived from knowledge gaps, which may be greater during the design phase, especially for first-of-a-kind designs like NuScale. While I support the use of risk-informed approaches to focus resources on matters of safety significance, a trustworthy risk-informed process should include a transparent discussion of uncertainties associated with the risk estimates and consideration of the risk contributors that may have been excluded from the analysis. An approach focused on numerical risk estimates without a sufficiently complete, application-specific discussion of the treatment of uncertainties and PRA limitations could lead to delay and inefficiency in the NRC's review. Applicants, particularly those with novel and unique design elements, need to appropriately evaluate and address these issues when using quantitative risk information for decision-making for the staff to be able to make a decision.

[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 52

[NRC-2017-0029]

RIN 3150-AJ98

NuScale Small Modular Reactor Design Certification

AGENCY: U.S. Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is proposing to amend its regulations to certify the NuScale standard design for a small modular reactor. Applicants or licensees intending to construct and operate a NuScale standard design may do so by referencing this design certification rule. The applicant for certification of the NuScale standard design is NuScale Power, LLC. The public is invited to submit comments on this proposed rule.

DATES: Submit comments by **[INSERT DATE 60 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. Comments received after this date will be considered if it is practical to do so, but the NRC is able to ensure consideration only for comments received before this date.

Deleted: 30

ADDRESSES: You may submit comments by any of the following methods (unless this document describes a different method for submitting comments on a specific subject);

however, the NRC encourages electronic comment submission through the Federal Rulemaking Web site:

- **Federal Rulemaking Web site:** Go to https://www.regulations.gov and search for Docket ID NRC-2017-0029. Address questions about NRC dockets to Dawn Forder; telephone: 301-415-3407; e-mail: Dawn.Forder@nrc.gov. For technical questions, contact the individual(s) listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **E-mail comments to:** Rulemaking.Comments@nrc.gov. If you do not receive an automatic e-mail reply confirming receipt, then contact us at 301-415-1677.

For additional direction on obtaining information and submitting comments, see "Obtaining Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

FOR FURTHER INFORMATION CONTACT: Yanely Malave, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-1519, e-mail: Yanely.Malave@nrc.gov, and Prosanta Chowdhury, Office of Nuclear Reactor Regulation, telephone: 301-415-1647, e-mail: Prosanta.Chowdhury@nrc.gov. Both are staff of the U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

TABLE OF CONTENTS

I. Obtaining Information and Submitting Comments

A. Obtaining Information

Please refer to Docket ID NRC-2017-0029 when contacting the NRC about the availability of information for this proposed rule. You may obtain publicly available information related to this proposed rule by any of the following methods:

- **Federal Rulemaking Web site:** Go to https://www.regulations.gov and search for Docket ID NRC-2017-0029.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may obtain publicly available documents online in the ADAMS Public Documents collection at https://www.nrc.gov/reading-rm/adams.html. To begin the search, select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, at 301-415-4737, or by e-mail to PDR.Resource@nrc.gov. The ADAMS accession number for each document referenced in this proposed rule (if that document is available in

ADAMS) is provided the first time that it is mentioned in this document. In addition, for the convenience of the reader, instructions about obtaining materials referenced in this document are provided in Section XV, "Availability of Documents," of this document.

- **Attention:** The Public Document Room (PDR), where you may examine and order copies of public documents, is currently closed. You may submit your request to the PDR via e-mail at PDR.Resource@nrc.gov or by calling 1-800-397-4209 between 8:00 a.m. and 4:00 p.m. (ET), Monday through Friday, except Federal holidays.
- **Attention:** The Technical Library, which is located at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852, is open by appointment only. Interested parties may make appointments to examine documents by contacting the NRC Technical Library by e-mail at Library.Resource@nrc.gov between 8:00 a.m. and 4:00 p.m. (ET), Monday through Friday, except Federal holidays.

B. Submitting Comments

The NRC encourages electronic comment submission through the Federal Rulemaking Web site (https://www.regulations.gov). Please include Docket ID NRC-2017-0029 in your comment submission.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at https://www.regulations.gov as well as enter the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment submissions into ADAMS.

II. Background

Part 52 of title 10 of the *Code of Federal Regulations* (10 CFR), “Licenses, Certifications, and Approvals for Nuclear Power Plants,” subpart B, “Standard Design Certifications,” presents the process for obtaining standard design certifications. By letter dated December 31, 2016, NuScale Power, LLC, (NuScale Power) filed its application for certification of the NuScale standard design (hereafter referred to as NuScale) (ADAMS Accession No. ML17013A229). The NRC published a notification of receipt of the design certification application (DCA) in the *Federal Register* on February 22, 2017 (82 FR 11372). On March 30, 2017, the NRC published a notification of acceptance for docketing of the application in the *Federal Register* (82 FR 15717) and assigned docket number 52-048. The preapplication information submitted before the NRC formally accepted the application can be found in ADAMS under Docket No. PROJ0769.

NuScale is the first small modular reactor design reviewed by the NRC. NuScale is based on a small light water reactor developed at Oregon State University in the early 2000s. It consists of one or more NuScale power modules (hereafter referred to as power module(s)). A power module is a natural circulation light water reactor composed of a reactor core, a pressurizer, and two helical coil steam generators located in a common reactor pressure vessel that is housed in a compact cylindrical steel containment. The NuScale reactor building is designed to hold up to 12 power modules. Each power module has a rated thermal output of 160 megawatt thermal (MWt) and electrical output of 50 megawatt electric (MWe), yielding a total capacity of 600 MWe for

12 power modules. All NuScale power modules are partially submerged in one safety-related pool, which is also the ultimate heat sink for the reactor. The pool portion of the reactor building is located below grade. The design utilizes several first-of-a-kind approaches while accomplishing key safety functions, such as no Class 1E safety-related power (no emergency diesel generators), no need for pumps to inject water into the core for post-accident coolant injection, and reduced control room staffing while providing safe operation of the plant during normal and post-accident operation.

III. Regulatory and Policy Issues

A. Control Room Staffing Requirements

The requirements in § 50.54(k) and § 50.54(m) identify the minimum number of licensed operators that must be on site, in the control room, and at the controls. The requirements are conditions in every nuclear power reactor operating license issued under 10 CFR part 50, “Domestic Licensing of Production and Utilization Facilities.” The requirements also are conditions in every combined license (COL) issued under 10 CFR part 52; however, they are applicable only after the Commission makes the finding under § 52.103(g) that the acceptance criteria in the COL are met.

In a letter to the NRC, dated September 15, 2015 (ADAMS Accession No. ML15258A846), NuScale Power proposed that 6 licensed operators will operate up to 12 power modules from a single control room. However, the staffing proposal would not meet the requirements in § 50.54(m)(2)(i) because the minimum requirements for the onsite staffing table in § 50.54(m)(2)(i) do not address operation of more than two units from a single control room. The proposal also would not meet § 50.54(m)(2)(iii) because the regulation requires a licensed operator at the controls for each fueled unit (i.e., 12 licensed operators). Absent alternative staffing requirements, future applicants referencing the NuScale design would need to request an exemption from these requirements.

In the DCA Part 7, Section 6.2, “Justification for Rulemaking,” NuScale Power provided a technical basis for rulemaking language that would address control room staffing in conjunction with control room configuration. NuScale Power’s approach is consistent with SECY-11-0098, “Operator Staffing for Small or Multi-Module Nuclear Power Plant Facilities,” dated July 22, 2011 (ADAMS Accession No. ML111870574). In Chapter 18, Section 18.5.4.2, “Evaluation of the Applicant’s Technical Basis,” of the final safety evaluation report (ADAMS Accession No. ML20023B605), the NRC found that NuScale Power’s proposed staffing level, as described in the DCA Part 7, Section 6, is acceptable. Because Section V, “Applicable Regulations,” of this proposed rule includes the alternative requirement provisions, staffing table, and appropriate table notes, a future applicant or licensee that references proposed appendix G to 10 CFR part 52 would not need to request an exemption from § 50.54(m).

B. Incorporation by Reference

The proposed Section III.A, “Incorporation by reference approval,” of appendix G to 10 CFR part 52 explicitly lists documents that are to be approved by the Director of the Office of the Federal Register for incorporation by reference into this appendix. Proposed Section III.B.2 identifies information that is not within the scope of the design certification and, therefore, is not incorporated by reference into this appendix. This information includes conceptual design information, as defined in § 52.47(a)(24), and the discussion of “first principles” described in the Design Control Document (DCD) Part 2, Tier 2, Section 14.3.2, “Tier 1 Design Description and Inspections, Tests, Analyses, and Acceptance Criteria First Principles.”

C. Issues Not Resolved by the Design Certification

The NRC identified three issues as not resolved within the meaning of § 52.63(a)(5). There was insufficient information available for the NRC to resolve issues regarding (1) the shielding wall design in certain areas of the plant; (2) the potential for containment leakage from the combustible gas monitoring system, and (3) the ability of the steam generator tubes to maintain structural and leakage integrity during density wave oscillations in the secondary fluid system, including the method of analysis to predict the thermal-hydraulic conditions of the steam generator secondary fluid system and resulting loads, stresses, and deformations from density wave oscillations from reverse flow.

1. Shielding Wall Design

As discussed in Section 12.3.4.1.2 of the final safety evaluation report, the NRC found that there were insufficient design details available regarding shielding wall design with the presence of large penetrations, such as the main steam lines; main feed water lines; and power module bay heating, ventilation, and air conditioning lines in the radiation shield wall between the power module bay and the reactor building steam gallery area. Without this shielding design information, the NRC is unable to confirm that the radiological doses to workers will be maintained within the radiation zone limits specified in the application.

This issue is narrowly focused on the shielding walls between the reactor module bays and the reactor building steam gallery areas. The radiation zones and dose calculations, including dose calculations for the dose to workers, members of the public, and environmental qualification, in areas outside of the reactor module bay are calculated assuming a solid wall and currently do not account for penetrations in the shield wall. A COL applicant will be required to demonstrate penetration shielding adequate to match the information in the NuScale DCD (specifically, the plant radiation zones, environmental qualification dose calculations, and dose estimates for workers and the public). A COL applicant can provide this information for the NRC to review because this issue involves a localized area of the plant without affecting other aspects of the NRC's review of the NuScale design. Therefore, the NRC has determined that this information can be provided by a COL applicant that references this appendix without a demonstrable impact on safety or standardization. Appendix G to 10 CFR part 52, Section VI, "Issue Resolution," will clarify that this issue is not resolved within the meaning of § 52.63(a)(5), and Section IV, "Additional Requirements and Restrictions," will state that the COL applicant is responsible for providing the design information to address this issue.

2. Containment Leakage from the Combustible Gas Monitoring System

As documented in Section 12.3.4.1.3 of the final safety evaluation report, there was insufficient information available regarding NuScale combustible gas monitoring system and the potential for leakage from this system outside containment. Without additional information regarding the potential for leakage from this system, the NRC was unable to determine whether this leakage could impact analyses performed to assess main control room dose consequences and offsite dose consequences to members of the public and whether this system can be safely re-isolated after monitoring is initiated due to potentially high dose levels at or near the isolation valve location. The isolation valve can only be operated locally, and dose levels at the valve location have not been determined.

This issue is narrowly focused on the radiation dose implications as a result of using the post-accident combustible gas monitoring loop. A COL applicant will be required to demonstrate either that offsite and main control room dose calculations are not exceeded or that the system can be safely re-isolated, if needed. This issue does not affect normal plant operation or non-core damage accidents. The issue may be resolved by performing radiation dose calculations and demonstrating that doses will remain within applicable dose limits in 10 CFR part 20, “Standards for Protection Against Radiation.” More information may be available at the COL application stage that would allow for more detailed calculations. Any design changes to address this issue would only affect the combustible gas monitoring loop to ensure it can be re-isolated or to ensure that dose limits are not exceeded. Such design changes would likely not have an impact on other systems or equipment, and the NRC would review such changes and any resulting effects on other structures, systems, and components during the COL application review to provide reasonable assurance of adequate protection. Therefore, the NRC has determined that this information can be provided by a COL applicant that references this appendix without a demonstrable impact on safety or standardization. Appendix G to 10 CFR part 52, Section VI, “Issue Resolution,” will clarify that this issue is not resolved within the meaning of § 52.63(a)(5), and Section IV, “Additional Requirements and Restrictions,” will state that the COL applicant is responsible for providing the design information to address this issue.

3. Steam Generator Stability during Density Wave Oscillations and Associated Method of Analysis

Section 5.4.1.2, “System Design,” in Revision 2 of the DCA Part 2, Tier 2, stated that a flow restriction device at the inlet to each steam generator tube “ensures secondary-side flow stability and precludes density wave oscillations.” However, the applicant modified this section in Revision 3 of the DCA Part 2, Tier 2 to state that the steam generator inlet flow restrictors provide the necessary secondary-side pressure drop “to reduce flow oscillations to acceptable limits.” Revision 4.1 of the DCA (ADAMS Accession No. ML20205L562) revised Section 5.4.1.2 to state that the steam generator inlet flow restrictors are designed “to reduce the potential for density wave oscillations.” Note that Revision 5 of the DCA (ADAMS Accession No. ML20225A071) provides only editorial changes to Revision 4.1 and does not change the technical content or conclusions.

Sections 3.9.2, 3.9.5, and 5.4.1 of the final safety evaluation report relied on the applicant’s statements in Revision 2 and Revision 3 of the DCA that flow oscillations in the secondary fluid system of the steam generators would either be precluded or minimal. After issuance of the advanced safety evaluation report, the NRC noted inconsistencies and gaps in the information provided in Sections 3.9.1, 3.9.2, and 5.4.1 of Revision 4.1 of the DCA Part 2, Tier 2 regarding the potential for significant density wave oscillations in the steam generator tubes, including both forward and reverse secondary flow. The testing performed by the applicant on various conceptual designs of the steam generator inlet flow restrictors only involved flow in the forward direction without oscillation or reverse flow.

As a result, NuScale Power has not demonstrated that the flow oscillations that are predicted to occur on the secondary-side of the steam generators will not cause failure of the inlet flow restrictors. Structural and leakage integrity of the inlet flow restrictors in the steam generators is necessary to avoid damage to multiple steam generator tubes, caused directly by broken parts or indirectly by unexpected density wave oscillation loads. Damage to multiple steam generator tubes could disrupt natural circulation in the reactor coolant pathway and interfere with the decay heat removal system and the emergency core cooling system, which is relied upon to cool the reactor core in a NuScale nuclear power module. The failure of multiple steam generator tubes resulting from failure of an inlet flow restrictor has not been included within the scope of the NuScale accident analyses in DCA Part 2, Tier 2, Chapter 15. Therefore, the NRC concludes that NuScale Power has not demonstrated compliance with 10 CFR part 20 and 10 CFR part 50, appendix A, General Design Criterion (GDC) 4 and GDC 31, relative to potential impacts on steam generator tube integrity from inlet flow restrictor failure.

As described previously, NuScale Power made a change to the description of inlet flow restrictor performance beginning with DCA Part 2, Tier 2, Revision 3, that indicates that the design no longer precludes density wave oscillations in the secondary-side of the steam generators. As a result, the design needs a method of analysis to predict the thermal-hydraulic conditions of the steam generator secondary fluid system and resulting loads, stresses, and deformations from density wave oscillations including reverse flow. However, an appropriate method of analysis has not been provided to the NRC.

The DCA Part 2, Tier 2, Section 3.9.1.2, “Computer Programs Used in Analyses,” lists the computer programs used by NuScale Power in the dynamic and static analyses of mechanical loads, stresses, and deformations, and in the hydraulic transient load analyses of seismic Category I components and supports for the NuScale nuclear power plant. Section 3.9.1.2 states that NRELAP5 is NuScale’s proprietary system thermal-hydraulics code for use in safety-related design and analysis calculations and is pre-verified and configuration-managed. The advanced safety evaluation report, Section 3.9.1.4.9, “Computer Programs Used in Analyses,” states that the NRELAP5 computer program had received verification and validation. Following preparation of the advanced safety evaluation report, the NRC noted a discrepancy between two statements in the DCA about validation for NRELAP5: DCA Part 2, Tier 2, Section 5.4.1.3 in Revision 4 stated that NRELAP5 was validated for determining density wave oscillation thermal-hydraulic conditions, referring to Section 15.0.2 for more information, but neither Section 15.0.2 nor TR-1016-51669 describe validation for determining density wave oscillation thermal-hydraulic conditions.

On June 19, 2020, NuScale submitted Revision 4.1 of the DCA Part 2, Tier 2 (ADAMS Accession No. ML20205L562; subsequently included in Revision 5 of the DCA submitted on July 29, 2020 (ADAMS Accession No. ML20225A071)) to correct the discrepancies, and acknowledges the need for a COL applicant to address secondary-side instabilities in the steam generator design. Specifically, the update to Section 3.9.1.2 in Revision 4.1 of DCA Part 2, Tier 2, references DCA Part 2, Tier 2, Section 15.0.2, "Review of Transient and Accident Analysis Methods," for the discussion of the development, use, verification, validation, and code limitations of the NRELAP5 computer program for application to transient and accident analyses. The correction to Section 3.9.1.2 also references technical report TR-1016-51669, "NuScale Power Module Short-Term Transient Analysis," incorporated by reference in DCA Part 2, Tier 2, Table 1.6-2, for application of the NRELAP5 computer program to short-term transient dynamic mechanical loads, such as pipe breaks and valve actuations. In addition, the correction to Section 3.9.1.2 includes a new COL item specifying that a COL applicant that references the NuScale DCD will develop an evaluation methodology for the analysis of secondary-side instabilities in the steam generator design. The COL item states that this methodology will address the identification of potential density wave oscillations in the steam generator tubes and qualification of the applicable portions of the reactor coolant system integral reactor pressure vessel and steam generator given the occurrence of density wave oscillations, including the effects of reverse fluid flows within the tubes. These corrections to the DCA clarify that the evaluation methodology for the analysis of secondary-side instabilities in the steam generator design was not verified and validated as part of the NuScale DCA but will be accomplished by the COL applicant.

This steam generator design issue is narrowly focused on the effects of density wave oscillations in the secondary fluid system on steam generator tubes to maintain structural and leakage integrity, including the method of analysis to predict the thermal-hydraulic conditions of the steam generator secondary fluid system and resulting loads, stresses, and deformations from density wave oscillations including reverse flow. No other reactor safety aspect of the steam generators is impacted by this design issue. As a result, the NRC finds that this is an isolated issue that does not affect other aspects of the NRC's review of the design of the NuScale nuclear power plant. Therefore, the NRC has determined that this information can be provided by a COL applicant that references this appendix, consistent with the other design information regarding steam generator integrity described in DCA Part 2, Tier 2, Sections 3.9.1, 3.9.2, and 5.4.1, without a demonstrable impact on safety or standardization. Therefore, appendix G to 10 CFR part 52, Section VI, "Issue Resolution," will clarify that this issue is not resolved within the meaning of § 52.63(a)(5), and Section IV, "Additional Requirements and Restrictions," will state that the COL applicant is responsible for providing the design information to address this issue.

IV. Technical Issues Associated with the NuScale Design

The NRC identified significant technical issues associated with the following design areas that were resolved by NuScale Power during the review:

- Comprehensive vibration assessment program;
- Containment safety analysis;
- Emergency core cooling system inadvertent actuation block valve;
- Conformance with GDC 27, "Combined Reactivity Control System Capability," of appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR part 50;
- Absence of safety-related Class 1E alternating current (AC) or direct current (DC)

electrical power;

- Accident source term methodology;
- Boron redistribution during passive cooling modes.

In addition, the NRC granted 17 exemptions from 10 CFR part 50 to address various aspects of NuScale's design.

A. Comprehensive Vibration Assessment Program

The NuScale comprehensive vibration assessment program limits potentially adverse effects from flow, acoustic, and mechanically induced vibrations and resonances on NuScale power module components, including the helical coil steam generators. The NuScale steam generators are different from those of operating pressurized water reactors in that the primary reactor coolant is on the outside of the steam generator tubes and the steam is on the inside. Because of this design, there is the possibility of density wave oscillation instabilities in the secondary coolant which could challenge the integrity of the tubes. The NRC's review and findings, including independent analyses and observation of vibration testing, are documented in detail in Chapter 3, "Design of Structures, Components, Equipment, and Systems," Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," of the final safety evaluation report. The review focused on assuring that the design of the helical coil steam generator tubes would not result in issues with flow-induced vibration.

As part of the comprehensive vibration assessment, the NRC also reviewed and found acceptable the steam generator tube margin against fluid-elastic instability, steam generator tube margin against vortex shedding, control rod drive shaft margin against vortex shedding, in-core instrument guide tube against vortex shedding, decay heat removal system piping against acoustic resonance, and control rod assembly guide tube against turbulence buffeting. The steam generator tube margins against fluid-elastic instability and vortex shedding will be validated in the TF-3 testing facility as described in DCA Part 2, Tier 1, Section 2.1.1, “Design Description.” In addition, the initial startup testing will confirm that flow-induced vibration will not cause adverse effects on the plant system components including the steam generator tubes. With the exception of the steam generator tube and inlet flow restrictor issue discussed above, the NRC found the comprehensive vibration assessment program adequate to ensure the structural integrity of the NuScale power module components.

B. Containment Safety Analysis

NuScale incorporates novel and unique features which result in transient thermal-hydraulic responses that are different from those of currently licensed reactors.

There are several peak containment pressure analysis technical issues unique to NuScale, including the associated thermal-hydraulic analyses. In support of containment safety analysis, NuScale Power submitted technical report TR-0516-49084-P, Revision 3, “Containment Response Analysis Methodology,” May 2020 (ADAMS Accession No. ML20141L808) that describes the conservative containment pressure and temperature safety analyses for several design-basis events related to the containment design margins. NuScale also submitted topical report TR-0516-49422, “Loss-of-Coolant Accident Evaluation Model,” Revision 1, dated November 2019 (ADAMS Accession No. ML19331B585). This topical report describes the evaluation model used to analyze the power module response during a design-basis loss-of-coolant accident. The NRC reviewed this topical report as part of the containment safety analysis.

The NRC also observed thermal-hydraulic performance testing at NuScale Power’s integrated system test facility, which validates the analytical model. Based on initial testing results and thermal-hydraulic analyses, NuScale Power made design changes to increase the initial reactor building pool level and the in-containment vessel design pressure to account for some uncertainties.

The NRC reviewed the details of the computer thermal-hydraulic evaluation model described in the DCA Part 2, Tier 2, Section 6.2.1.1 to determine whether any uncertainties were properly accounted for and found the containment design margins to be acceptable. The associated safety evaluation report approving topical report TR-0516-49422 was issued on February 18, 2020 (ADAMS Accession No. ML20044E199). The NRC's review and specific findings, including independent analyses and observation of NuScale testing, are documented in Chapter 6, "Engineered Safety Features," Section 6.2.1.1, "Containment Structure," of the safety evaluation report.

C. Emergency Core Cooling System Inadvertent Actuation Block Valve

The NuScale emergency core cooling system relies on natural circulation cooling of the reactor core by releasing the heated reactor coolant steam from the top of the reactor pressure vessel through three reactor vent valves into the containment vessel and returning the cooled condensed reactor coolant water to the reactor pressure vessel through two reactor recirculation valves. Each reactor vent valve and reactor recirculation valve consists of a first-of-a-kind arrangement of a main valve, an inadvertent actuation block (IAB) valve, a solenoid trip valve, and a solenoid reset valve. The IAB valve for each reactor vent valve and reactor recirculation valve is designed to close rapidly to prevent its corresponding emergency core cooling system main valve from opening when the reactor coolant system is at high pressure conditions. Premature opening of the emergency core cooling system main valves could result in fuel damage. The IAB valve then opens at reduced reactor coolant system pressure to allow the main valve to open and permit natural circulation cooling of the reactor core in response to a plant event. Although the valve assemblies are considered an active component, NuScale does not apply the single failure criterion to the IAB valve, including to the IAB valve's function to close. Consistent with Commission safety goals and the practice of risk-informed decision-making, the NRC evaluated the NuScale emergency core cooling system valve system without assuming a single active failure of the IAB valve to close.

During design demonstration tests of the first-of-a-kind emergency core cooling system valve system performed in accordance with § 50.43(e), NuScale Power implemented design modifications to the main valve and IAB valve to demonstrate that the IAB valve will operate within a specific design pressure range. The DCD specifies that the emergency core cooling system valves (including the IAB valves) will be qualified in accordance with American Society of Mechanical Engineers Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," as endorsed by NRC Regulatory Guide 1.100, Revision 3, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," prior to installation in a NuScale nuclear power plant. Additionally, the NRC regulations in § 50.55a require that a NuScale nuclear power plant satisfy American Society of Mechanical Engineers Operation and Maintenance of Nuclear Power Plants, Division 1, OM Code: Section IST (OM Code) as incorporated by reference in § 50.55a for inservice testing of the emergency core cooling system valves, unless relief is granted or an alternative is authorized by the NRC. The NRC's review and findings related to the IAB valve are documented in safety evaluation report Chapter 3, "Design of Structures, Components, Equipment, and Systems," Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints." These findings show that the NRC regulatory requirements and DCD Part 2, Tier 2 provisions provide reasonable assurance that the emergency core system valve system will be

Deleted: accepted

Deleted: in

Deleted: (

Deleted:)

capable of performing its design-basis functions in light of the safety significance of the required opening and closing pressures for the individual IAB valves.

Further, Chapter 15, “Transient and Accident Analyses,” Section 15.0.0.5, “Limiting Single Failures,” of the safety evaluation report states that the IAB valve is a first-of-a-kind, safety-significant, active component integral to the NuScale emergency core cooling system. NuScale does not apply the single failure criterion to the IAB valve, and the Commission directed the staff in SRM-SECY-19-0036, “Staff Requirements—SECY-19-0036—Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves,” (ADAMS Accession No. ML19183A408) to “review Chapter 15 of the NuScale Design Certification Application without assuming a single active failure of the inadvertent actuation block valve to close.” The Commission further stated that “[t]his approach is consistent with the Commission’s safety goal policy and associated core damage and large release frequency goals and existing Commission direction on the use of risk-informed decision-making, as articulated in the 1995 Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities and the White Paper on Risk-Informed and Performance-Based Regulation (in SRM-SECY-98-144, “White Paper on Risk-Informed and Performance-Based Regulation,” and Yellow Announcement 99-019).”

Based on the NRC’s historic application of the single failure criterion and Commission direction on the subject, as described in SECY-77-439, “Single Failure Criterion” (ADAMS Accession No. ML060260236), SRM-SECY-94-084, “Policy and Technical Issues associated with the Regulatory Treatment of Non-Safety Systems and Implementation of Design Certification and Light-Water Reactor Design Issues” (ADAMS Accession No. ML003708098), and SRM-SECY-19-0036, the NRC has retained some discretion, in fact- or application-specific circumstances, to decide when to apply the single failure criterion. The Commission’s decision in SRM-SECY-19-0036 provides direction regarding the appropriate application and interpretation of the regulatory requirements in 10 CFR part 50 to the NuScale IAB valve's function to close. This decision is similar to those in previous Commission documents that addressed the use of the single failure criterion and provided clarification on when to apply the single failure criterion in other specific instances.

Deleted: documented

D. Exemption to General Design Criterion 27, "Combined Reactivity Control System Capability"

NuScale Power determined that, under certain end-of-cycle scenarios with one control rod stuck out, the NuScale reactivity control systems could not prevent re-criticality and return to power. This result does not meet GDC 27 of appendix A to 10 CFR part 50, which requires reactivity control systems to reliably control reactivity changes under postulated accident conditions with margin for stuck control rods. Therefore, NuScale Power submitted an exemption request for GDC 27 (refer to Section 15, "10 CFR 50, Appendix A, Criterion 27, Combined Reactivity Control Systems Capability," of DCA Part 7, "Exemptions").

Deleted: the requirements of

NuScale Power analyses determined that the specified acceptable fuel design limits would not be exceeded and that core cooling would be maintained during a return to power under these scenarios. The global core power level would be less than 10 percent and within capacity of the safety-related, passive decay heat removal system. The NRC independently verified NuScale Power's results and found that NuScale achieves the fundamental safety functions for nuclear reactor safety, which are to control heat generation, remove heat, and limit the release of radioactive materials. Chapter 15, Section 15.0.6.4.1, of the safety evaluation report contains details of the evaluation of this exemption request. Additional information is provided in SECY-18-0099, "NuScale Power Exemption Request from 10 CFR Part 50, Appendix A, General Design Criterion

27, ‘Combined Reactivity Control Systems Capability’” (ADAMS Accession No. ML18065A431), dated October 9, 2018. The NRC granted the exemption request.

E. Safety-Related Class 1E AC or DC Electrical Power

NuScale does not contain safety-related Class 1E AC or DC electrical power systems. The purpose of appendix A to 10 CFR part 50, GDC 17, “Electric Power Systems,” is to ensure that sufficient electric power is available to accomplish plant functions important to safety. NuScale provides passive safety systems and features to accomplish plant safety-related functions without reliance on electrical power.

NuScale incorporates several innovative features that reduce the overall complexity of the design and reduce the number of safety-related systems necessary to mitigate postulated accidents. NuScale has no safety-related functions that rely on electrical power. For example, the emergency core cooling system performs its safety function without reliance on safety-related electrical power or external sources of coolant inventory makeup. NuScale Power provided a methodology to substantiate its assertion that the safety-related systems do not rely on Class 1E electrical power in topical report TR-0815-16497, “Safety Classification of Passive Nuclear Power Plant Electrical Systems,” dated February 23, 2018 (ADAMS Accession No. ML18054B607). The NRC reviewed topical report TR-0815-16497 and concluded that NuScale Power demonstrated that NuScale safety-related systems do not rely on Class 1E electrical power. The NRC’s review and conclusions are documented in a safety evaluation report approving topical report TR-0815-16497 (ADAMS Accession No. ML17048A459) issued December 13, 2017, as described in the final safety evaluation report for Chapter 1, “Introduction and General Discussion,” (ADAMS Accession No. ML20204A968).

Deleted: lower

Deleted: There

Deleted: are

Deleted: in NuScale

Commented [NT1]: Please check this ADAMS Accession No.

Because no safety-related functions of NuScale rely on electrical power, NuScale does not need any safety-related electrical power systems. Therefore, NuScale Power requested an exemption from GDC 17, which requires the provision of onsite and offsite power to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. The NRC determined that, subject to limitations and conditions stipulated in its safety evaluation report for TR-0815-16497, the underlying purpose of GDC 17 (to ensure sufficient electric power is available to accomplish the safety functions of the respective systems), is met without reliance on Class 1E electric power; in other words, the onsite and offsite electric power systems are classified as non-Class 1E systems and electric power is not needed (1) to achieve or maintain safe shutdown, (2) to assure specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences, or (3) to maintain core cooling, containment integrity, and other vital functions during postulated accidents. Further, the onsite and offsite power systems are not needed to permit functioning of SSCs important to safety. Therefore, NuScale Power was granted an exemption from GDC 17. The NRC's evaluation of NuScale Power's exemption request from the requirements of GDC 17 is documented in Section 8.1.5, "Technical Evaluation for Exemptions," of the final safety evaluation report for Chapter 8, "Electric Power" (ADAMS Accession No. ML20023B614).

F. Accident Source Term Methodology

The NRC reviewed NuScale Power's methods for developing accident source terms and performing accident radiological consequence analyses. As defined in § 50.2, "Definitions," a source term "refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release." NuScale Power developed source terms for deterministic accidents for NuScale that are similar to those which have been used in safety and siting assessments for large light water reactors. The design-basis accidents for NuScale are the main steam line break outside containment, rod ejection accident, fuel handling accident, steam generator tube failure, and the failure of small lines carrying primary coolant outside containment.

To address the source term regulatory requirements, NuScale Power submitted topical report TR-0915-17565, Revision 3, "Accident Source Term Methodology," dated April 2019 (ADAMS Accession No. ML19112A172). The topical report proposes a methodology to develop a source term based on several severe accident scenarios that result in core damage, taken from the design probabilistic risk assessment. This source term is the surrogate radiological source term for a core damage event.

Deleted: ,

Deleted: that

The topical report also provides methods for determining radiation sources not developed from core damage scenarios for use in the evaluation of environmental qualification of equipment in accordance with § 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants." Specifically, the report describes an iodine spike source term not involving core damage, which is a surrogate accident that bounds potential accidents with release of the reactor coolant into the containment vessel.

Deleted: The NRC's review of the NuScale Power DCA was done in conjunction with the review and approval of topical report TR-0915-17565.

Additionally, SECY-19-0079, "Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application," dated August 16, 2019 (ADAMS Accession No. ML19107A455), is a related information paper sent to the Commission. The paper describes the regulatory and technical issues raised by unique aspects of NuScale Power's proposed methodology and the staff's approach to reviewing topical report TR-0915-17565.

The NRC's review and findings of topical report TR-0915-17565, Revision 3, are documented in the topical report final safety evaluation report issued on October 29, 2019 (ADAMS Accession No. ML19297G520). The approved version TR-0915-17565-NP-A, Revision 4 (ADAMS Accession No. ML20057G132) is discussed in the DCA safety evaluation report Section 12.2, “Radiation Sources,” Section 12.3, “Radiation Protection Design Features,” Section 3.11 “Environmental Qualification of Mechanical and Electrical Equipment,” and Section 15.0.3, “Radiological Consequences of Design Basis Accidents.” The NRC found the accident source terms acceptable for the purposes described in each of the above safety evaluation report sections.

G. Boron Redistribution during Passive Cooling Modes

The NRC evaluated the effects of boron volatility and redistribution during long-term passive cooling. During this mode of operation, boron-free steam will enter the downcomer and containment which can potentially challenge reactor core shutdown margin and could lead to a return to power. The NRC reviewed analyses provided by NuScale Power demonstrating that the reactor remains subcritical and that specified acceptable fuel design limits are not exceeded. The NRC evaluated the technical basis for NuScale Power’s approach and conducted confirmatory calculations and independent assessments to determine its acceptability. The staff’s review is primarily documented in Chapter 15, Section 15.0.5, “Long Term Decay Heat and Residual Heat Removal,” and Section 15.6.5, “Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary,” of the safety evaluation report. Specifically, the staff concluded that the top of active fuel remains covered with acceptably low cladding temperatures and that for beginning-of-cycle and middle-of-cycle conditions, with no operator actions, the core remains subcritical. The potential for an end-of-cycle return to power is discussed in Section IV.D , “Exemption to General Design Criterion 27, “Combined Reactivity Control System Capability”,” of this document. In addition, Chapter 19, Section 19.1.4.6.4, "Success Criteria, Accident Sequences, and Systems Analyses," of the safety evaluation report concludes that an operator error during recovery of the module from an uneven boron distribution scenario is unlikely to lead to core damage and is not a significant risk contributor.

H. Exemptions

Deleted: G

NuScale Power submitted a total of 17 requests for exemptions from the following regulations, including those discussed as part of the significant technical issues mentioned above (see Table 1.14-1, "NuScale Design Certification Exemptions," in Chapter 1 of the final safety evaluation report (ADAMS Accession No. ML20204A986)):

1. §§ 50.46a and 50.34(f)(2)(vi) (Reactor Coolant System Venting)
2. § 50.44 (Combustible Gas Control)
3. § 50.62(c)(1) (Reduction of Risk from Anticipated Transients Without Scram)
4. Appendix A to 10 CFR part 50, GDC 17, "Electric Power Systems"; GDC 18, "Inspection and Testing of Electric Power Systems"; and related provisions of GDC 34, "Residual Heat removal"; GDC 35, "Emergency Core Cooling"; GDC 38, "Containment Heat Removal"; GDC 41, "Containment Atmosphere Cleanup"; and GDC 44, "Cooling Water" (Electric Power Systems GDCs)
5. Appendix A to 10 CFR part 50, GDC 33, "Reactor Coolant Makeup"
6. § 50.54(m) (Control Room Staffing) (Alternative to meet the regulation)
7. Appendix A to 10 CFR part 52, GDC 52, "Capability of Containment Leakage Rate Testing"
8. Appendix A to 10 CFR part 50, GDC 40, "Testing of Containment Heat Removal System"
9. Appendix A to 10 CFR part 50, GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," GDC 56, "Primary Containment

Isolation," and GDC 57, "Closed Systems Isolation Valves" (Containment Isolation)

10. Appendix K to 10 CFR part 50 (Emergency Core Cooling System Evaluation Model)
11. § 50.34(f)(2)(xx) (Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators)
12. § 50.34(f)(2)(xiii) (Pressurizer Heater Power Supplies)
13. § 50.34(f)(2)(xiv)(E) (Containment Evacuation System Isolation)
14. § 50.46 (Fuel Rod Cladding Material)
15. Appendix A to 10 CFR part 50, GDC 27, "Combined Reactivity Control Systems Capability"
16. § 50.34(f)(2)(viii) (Post-Accident Sampling)
17. Appendix A to 10 CFR part 50, GDC 19, "Control Room"

NRC's safety evaluation report for Chapter 1, "Introduction and General Discussion" Section 1.14, "Index of Exemptions," lists these exemption requests with the corresponding sections of the safety evaluation reports where these exemption requests have been dispositioned. The NRC granted all of these exemption requests.

V. Discussion

Final Safety Evaluation Report

NuScale Power submitted the final revision of the NuScale DCA, Revision 5, in July 2020 (ADAMS Accession No. ML20225A071). In August 2020, the NRC issued a final safety evaluation report (ADAMS Accession No. ML20023A318) after the Advisory Committee on Reactor Safeguards (ACRS) performed its final independent review and issued its letter to the Commission in July 2020 on its findings and recommendations

(ADAMS Accession No. ML20211M386). The final safety evaluation report is a collection of reports written by the NRC documenting the safety findings from its review of the standard design application, and it reflects all changes resulting from interactions with the ACRS as well as changes in the final version of the DCA. The final safety evaluation report reflects that NuScale Power has resolved all technical and safety issues with the exception of the three issues discussed above. The final safety evaluation report describes these portions of the design that are not receiving finality in this rule and, therefore, will not be part of the certified design. The final safety evaluation report includes an index of all NRC requests for additional information, a chronology of all documents related to the NuScale DCA review, and summaries of public meetings and audits.

Deleted: and all previously identified confirmatory items

Deleted: also includes a description of

Deleted: design certification

NuScale Design Certification Proposed Rule

The following discussion describes the purpose and key aspects of each section of this NuScale design certification proposed rule. All section and paragraph references are to the provisions being added as appendix G to 10 CFR part 52, unless otherwise noted. The NRC has modeled this NuScale design certification proposed rule on existing design certification rules, with certain modifications where necessary to account for differences in the design documentation, design features, and environmental assessment (including severe accident mitigation design alternatives). As a result, design certification rules are standardized to the extent practical.

A. Introduction (Section I)

The purpose of Section I of appendix G to 10 CFR part 52 is to identify the standard design that would be approved by this design certification proposed rule and the applicant for certification of the standard design. Identification of the design certification applicant is necessary to implement appendix G to 10 CFR part 52 for two reasons. First, the implementation of § 52.63(c) depends on whether an applicant for a COL contracts with the design certification applicant to obtain the generic DCD and supporting design information. If the COL applicant does not use the design certification applicant to provide the design information and instead uses an alternate nuclear plant vendor, then the COL applicant must meet the requirements in § 52.73. Second, paragraph X.A.1 would require that the identified design certification applicant maintain the generic DCD throughout the time that appendix G to 10 CFR part 52 may be referenced.

B. Definitions (Section II)

The purpose of Section II of appendix G to 10 CFR part 52 is to define specific terminology with respect to this design certification proposed rule. During development of the first two design certification rules, the NRC decided that there would be both generic DCDs maintained by the NRC and the design certification applicant, as well as individual plant-specific DCDs maintained by each applicant or licensee that references a 10 CFR part 52 appendix. This distinction is necessary in order to specify the relevant plant-specific requirements to applicants and licensees referencing appendix G to 10 CFR part 52.

In order to facilitate the maintenance of the generic DCDs, the NRC requires that applicants for a standard design certification update their application to include an electronic copy of the final version of the DCD. The final version incorporates all amendments to the DCA submitted since the original application and any changes directed by the NRC as a result of its review of the original DCA or as a result of public comments. This final version is then incorporated by reference in the design certification rule. Once incorporated by reference, the final version becomes the "generic DCD," which will be maintained by the design certification applicant and the NRC and updated as needed to include any generic changes made after this design certification rulemaking. These changes would occur as the result of generic rulemaking by the NRC, under the change criteria in Section VIII of appendix G to 10 CFR part 52.

The NRC also requires each applicant and licensee referencing appendix G to 10 CFR part 52 to submit and maintain a plant-specific DCD as part of the COL final safety analysis report. The plant-specific DCD must either include or incorporate by reference the information in the generic DCD. The COL licensee will be required to maintain the plant-specific DCD, updating it as necessary to reflect the generic changes to the DCD that the NRC may adopt through rulemaking, plant-specific departures from the generic DCD that the NRC imposes on the licensee by order, and any plant-specific departures that the licensee chooses to make in accordance with the relevant processes in Section VIII of appendix G to 10 CFR part 52. A COL applicant may also have to include considerations for multi-module facilities in the plant-specific DCD that were not previously evaluated as part of the design certification rule, depending on the contents of the application. Therefore, the plant-specific DCD functions like an updated final safety analysis report because it would provide the most complete and accurate information on a plant's design basis for that part of the plant that would be within the scope of appendix G to 10 CFR part 52.

The NRC is treating the technical specifications in Chapter 16, "Technical Specifications," of the generic DCD as a special category of information and designating them as generic technical specifications in order to facilitate the special treatment of this information under appendix G to 10 CFR part 52. A COL applicant must submit plant-specific technical specifications that consist of the generic technical specifications, which may be modified as specified in paragraph VIII.C, and the remaining site-specific information needed to complete the technical specifications. The final safety analysis report that is required by § 52.79 will consist of the plant-specific DCD, the site-specific final safety analysis report, and the plant-specific technical specifications.

The terms Tier 1, Tier 2, and COL items (license information) are defined in appendix G to 10 CFR part 52 because these concepts were not envisioned when 10 CFR part 52 was developed. The design certification applicants and the NRC use these terms in implementing a two-tiered rule structure (the DCD is divided into Tier 1 and Tier 2 to support the rule structure) that was proposed by representatives of the nuclear industry after publication of 10 CFR part 52. The Commission approved the use of the two-tiered rule structure in its staff requirements memorandum, dated February 15, 1991, on SECY-90-377, "Requirements for Design Certification under 10 CFR part 52," dated November 8, 1990 (ADAMS Accession No. ML003707892).

The change process for Tier 2 information is similar, but not identical to, the change process set forth in § 50.59. The regulations in § 50.59 describe when a licensee may make changes to a plant as described in its final safety analysis report without a license amendment. Because of some differences in how the change control requirements are structured in the design certification rules, certain definitions contained in § 50.59 are not applicable to 10 CFR part 52 and are not being included in this proposed rule. The NRC is including a definition for "*Departure from a method of evaluation*" (paragraph II.F of appendix G to 10 CFR part 52), which is appropriate to include in this proposed rule, so that the eight criteria in paragraph VIII.B.5.b will be implemented for new reactors as intended.

C. Scope and Contents (Section III)

The purpose of Section III of appendix G to 10 CFR part 52 is to describe and define the scope and content of this design certification, explain how to obtain a copy of the generic DCD, identify requirements for incorporation by reference of the design certification rule, and set forth how documentation discrepancies or inconsistencies are to be resolved.

Paragraph III.A is the required statement of the Office of the Federal Register for approval of the incorporation by reference of the NuScale DCD, Revision 5. In addition, this paragraph provides the information on how to obtain a copy of the DCD. Unlike previous design certifications, the documents submitted to the NRC by NuScale Power did not use the title "Design Control Document;" they used the title "Design Certification Application" instead.

Paragraph III.B is the requirement for COL applicants and licensees referencing the NuScale DCD. The legal effect of incorporation by reference is that the incorporated material has the same legal status as if it were published in the *Code of Federal Regulations*. This material, like any other properly issued regulation, has the force and effect of law. Tier 1 and Tier 2 information (including the technical and topical reports referenced in the DCD Tier 2, Chapter 1) and generic technical specifications have been combined into a single document called the generic DCD in order to effectively control this information and facilitate its incorporation by reference into the rule. In addition, paragraph III.B clarifies that the conceptual design information and NuScale Power's evaluation of severe accident mitigation design alternatives are not considered to be part of appendix G to 10 CFR part 52. As provided by § 52.47(a)(24), these conceptual designs are not part of appendix G to 10 CFR part 52 and, therefore, are not applicable to an application that references appendix G to 10 CFR part 52. Therefore, such an applicant would not be required to conform to the conceptual design information that was provided by the design certification applicant. The conceptual design information, which consists of site-specific design features, was required to facilitate the design certification review. Similarly, the severe accident mitigation design alternatives were required to facilitate the environmental assessment.

Paragraphs III.C and III.D set forth the manner by which potential conflicts are to be resolved and identify the controlling document. Paragraph III.C establishes the Tier 1 description in the DCD as controlling in the event of an inconsistency between the Tier 1 and Tier 2 information in the DCD. Paragraph III.D establishes the generic DCD as the controlling document in the event of an inconsistency between the DCD and the final safety evaluation report for the certified standard design.

Paragraph III.E makes it clear that design activities outside the scope of the design certification may be performed using actual site characteristics. This provision applies to site-specific portions of the plant, such as the administration building.

D. Additional Requirements and Restrictions (Section IV)

Section IV of appendix G to 10 CFR part 52 sets forth additional requirements and restrictions imposed upon an applicant who references appendix G to 10 CFR part 52.

Paragraph IV.A sets forth the information requirements for COL applicants and distinguishes between information and documents that must be *included* in the application or the DCD and those which may be *incorporated by reference*. Any incorporation by reference in the application should be clear and should specify the title, date, edition, or version of a document, the page number(s), and table(s) containing the relevant information to be incorporated. The legal effect of such an incorporation by reference into the application is that appendix G to 10 CFR part 52 would be legally binding on the applicant or licensee.

In paragraph IV.B the NRC reserves the right to determine how appendix G to 10 CFR part 52 may be referenced under 10 CFR part 50. This determination may occur in the context of a subsequent rulemaking modifying 10 CFR part 52 or this design certification rule, or on a case-by-case basis in the context of a specific application for a

10 CFR part 50 construction permit or operating license. This provision is necessary because the previous design certification rules were not implemented in the manner that was originally envisioned at the time that 10 CFR part 52 was issued. The NRC's concern is with the manner by which the inspections, tests, analyses, and acceptance criteria (ITAAC) were developed and the lack of experience with design certifications in a licensing proceeding. Therefore, it is appropriate that the NRC retain some discretion regarding the manner by which appendix G to 10 CFR part 52 could be referenced in a 10 CFR part 50 licensing proceeding.

E. Applicable Regulations (Section V)

The purpose of Section V of appendix G to 10 CFR part 52 is to specify the regulations that were applicable and in effect at the time this design certification was approved. These regulations consist of the technically relevant regulations identified in paragraph V.A, except for the regulations in paragraph V.B that would not be applicable to this certified design.

F. Issue Resolution (Section VI)

The purpose of Section VI of appendix G to 10 CFR part 52 is to identify the scope of issues that would be resolved by the NRC through this proposed rule and, therefore, are "matters resolved" within the meaning and intent of § 52.63(a)(5). The section is divided into five parts: paragraph VI.A identifies the NRC's safety findings in adopting appendix G to 10 CFR part 52, paragraph VI.B identifies the scope and nature of issues that would be resolved by this proposed rule, paragraph VI.C identifies issues which are not resolved by this proposed rule, and paragraph VI.D identifies the issue finality restrictions applicable to the NRC with respect to appendix G to 10 CFR part 52.

Paragraph VI.A describes the nature of the NRC's findings in general terms and makes the findings required by § 52.54 for the NRC's approval of this design certification proposed rule.

Paragraph VI.B sets forth the scope of issues that may not be challenged as a matter of right in subsequent proceedings. The introductory phrase of paragraph VI.B clarifies that issue resolution, as described in the remainder of the paragraph, extends to the delineated NRC proceedings referencing appendix G to 10 CFR part 52. The remainder of paragraph VI.B describes the categories of information for which there is issue resolution.

Paragraph VI.C reserves the right of the NRC to impose operational requirements on applicants that reference appendix G to 10 CFR part 52. This provision reflects the fact that only some operational requirements, including portions of the generic technical specification in Chapter 16 of the DCD, were completely or comprehensively reviewed by the NRC in this design certification proposed rule proceeding. The NRC notes that operational requirements may be imposed on licensees referencing this design certification through the inclusion of license conditions in the license or inclusion of a description of the operational requirement in the plant-specific final safety analysis report.[1] The NRC's choice of the regulatory vehicle for imposing the operational requirements will depend upon, among other things, (1) whether the development and/or implementation of these requirements must occur prior to either the issuance of the COL or the Commission finding under § 52.103(g), and (2) the nature of the change controls that are appropriate given the regulatory, safety, and security significance of each operational requirement.

[1] Certain activities ordinarily conducted following fuel load and, therefore, considered "operational requirements," but which may be relied upon to support a Commission finding under § 52.103(g), may themselves be the subject of ITAAC to ensure their implementation prior to the § 52.103(g) finding.

Also, paragraph VI.C allows the NRC to impose future operational requirements (distinct from design matters) on applicants who reference this design certification. License conditions for portions of the plant within the scope of this design certification (e.g., startup and power ascension testing) are not restricted by § 52.63. The requirement to perform these testing programs is contained in the Tier 1 information. However, ITAAC cannot be specified for these subjects because the matters to be addressed in these license conditions cannot be verified prior to fuel load and operation when the ITAAC are satisfied. In the absence of detailed design information to evaluate the need for and develop specific post-fuel load verifications for these matters, the NRC is reserving the right to impose, at the time of COL issuance, license conditions addressing post-fuel load verification activities for portions of the plant within the scope of this design certification.

Paragraph VI.D reiterates the restrictions (contained in Section VIII of appendix G to 10 CFR part 52) placed upon the NRC when ordering generic or plant-specific modifications, changes, or additions to structures, systems, and components, design features, design criteria, and ITAAC within the scope of the certified design.

Paragraph VI.E ensures that the NRC will specify at an appropriate time the procedures on how to obtain access to sensitive unclassified and non-safeguards information (SUNSI) and safeguards information (SGI) for the NuScale design certification rule. Access to such information would be for the sole purpose of requesting or participating in certain specified hearings, such as hearings required by § 52.85 or an adjudicatory hearing. For proceedings where the notice of hearing was published before the effective date of the final rule, the Commission’s order governing access to SUNSI and SGI shall be used to govern access to such information within the scope of the rulemaking. For proceedings in which the notice of hearing or opportunity for hearing is published after the effective date of the final rule, paragraph VI.E applies and governs access to SUNSI and SGI.

G. Duration of this Appendix (Section VII)

The purpose of Section VII of appendix G to 10 CFR part 52 is, in part, to specify the period during which this design certification may be referenced by an applicant for a COL, under § 52.55, and the period it will remain valid when the design certification is referenced. For example, if an application references this design certification during the 15-year period, then the design certification would be effective until the application is withdrawn or the license issued on that application expires. The NRC intends for appendix G to 10 CFR part 52 to remain valid for the life of any COL that references the design certification to achieve the benefits of standardization and licensing stability. This means that changes to, or plant-specific departures from, information in the plant-specific DCD must be made under the change processes in Section VIII for the life of the plant.

H. Processes for Changes and Departures (Section VIII)

The purpose of Section VIII of appendix G to 10 CFR part 52 is to set forth the processes for generic changes to, or plant-specific departures (including exemptions) from, the DCD. The NRC adopted this restrictive change process in order to achieve a more stable licensing process for applicants and licensees that reference design certification rules. Section VIII is divided into three paragraphs, which correspond to Tier 1, Tier 2, and operational requirements.

Generic *changes* (called “modifications” in § 52.63(a)(3)) must be accomplished by rulemaking because the intended subject of the change is this design certification rule itself, as is contemplated by § 52.63(a)(1). Consistent with § 52.63(a)(3), any generic rulemaking changes are applicable to all plants, absent circumstances which render the change technically irrelevant. By contrast, plant-specific *departures* could be either an order to one or more applicants or licensees; or an applicant or licensee-initiated departure applicable only to that applicant's or licensee's plant(s), similar to a § 50.59 departure or an exemption. Because these plant-specific departures will result in a DCD that is unique for that plant, Section X would require an applicant or licensee to maintain a plant-specific DCD. For purposes of brevity, the following discussion refers to the processes for both generic changes and plant-specific departures as "change processes." Section VIII refers to an exemption from one or more requirements of this appendix and addresses the criteria for granting an exemption. The NRC cautions that when the exemption involves an underlying substantive requirement (i.e., a requirement outside this appendix), then the applicant or licensee requesting the exemption must demonstrate that an exemption from the underlying applicable requirement meets the criteria of §§ 52.7 and 50.12.

For the NuScale review, the staff followed the approach described in SECY-17-0075, "Planned Improvements in Design Certification Tiered Information Designations," dated July 24, 2017 (ADAMS Accession No. ML16196A321), to evaluate the applicant's designation of information as Tier 1 or Tier 2 information. Unlike some of the prior DCAs, this application did not contain any Tier 2* information. As described in SECY-17-0075, prior design certification rules in 10 CFR part 52, appendices A through E, information contained in the DCD was divided into three designations: Tier 1, Tier 2, and Tier 2*. Appendix F to 10 CFR part 52, the certification for the APR1400 design, does not contain Tier 2*. Tier 1 information is the portion of design-related information in the generic DCD that the Commission approves in the 10 CFR part 52 design certification rule appendices. To change Tier 1 information, NRC approval by rulemaking or approval of an exemption from the certified design rule is required. Tier 2 information is also approved by the Commission in the 10 CFR part 52 design certification rule appendices, but it is not certified and licensees who reference the design can change this information using the process outlined in Section VIII of the appendices. This change process is similar to that in § 50.59 and is generally referred to as the "50.59-like" process. If the criteria in Section VIII are met, a licensee can change Tier 2 information without prior NRC approval.

As mentioned in the previous paragraph, the NRC has used a third category, Tier 2*, in other design certification rules. This third category was created to address industry requests to minimize the scope of Tier 1 information and provide greater flexibility for making changes. Unlike Tier 2 information, all changes to Tier 2* information require a license amendment, but unlike Tier 1 information, no exemption is required. In those rules, Tier 2* information has the same safety significance as Tier 1 information but is part of the Tier 2 section of the DCD to afford more flexibility for licensees to change this type of information.

The applicant did not designate or categorize any Tier 2* information in the NuScale DCA. The NRC evaluated the Tier 2 information to determine whether any of that information should require NRC approval before it is changed. If the NRC had identified any such information in Tier 2, then the NRC would have requested that the applicant revise the application to categorize that information as Tier 1 or Tier 2*. The NRC did not identify any information in Tier 2 that should be categorized as Tier 2*. Because neither the applicant nor the NRC have designated any information in the DCD as Tier 2*, that designation and related requirements are not being used in this design certification rule.

Tier 1 Information

Paragraph A of Section VIII describes the change process for changes to Tier 1 information that are accomplished by rulemakings that amend the generic DCD and are governed by the standards in § 52.63(a)(1). A generic change under § 52.63(a)(1) will not be made to a certified design while it is in effect unless the change: (1) is necessary for compliance with NRC regulations applicable and in effect at the time the certification was issued; (2) is necessary to provide adequate protection of the public health and safety or common defense and security; (3) reduces unnecessary regulatory burden and maintains protection to public health and safety and common defense and security; (4) provides the detailed design information necessary to resolve select design acceptance criteria; (5) corrects material errors in the certification information; (6) substantially increases overall safety, reliability, or security of a facility and the costs of the change are justified; or (7) contributes to increased standardization of the certification information. The rulemakings must provide for notice and opportunity for public comment on the proposed change, as required by § 52.63(a)(2). The NRC will give consideration as to whether the benefits justify the costs for plants that are already licensed or for which an application for a permit or license is under consideration.

Departures from Tier 1 may occur in two ways: (1) the NRC may order a licensee to depart from Tier 1, as provided in paragraph VIII.A.3; or (2) an applicant or licensee may request an exemption from Tier 1, as addressed in paragraph VIII.A.4. If the NRC seeks to order a licensee to depart from Tier 1, paragraph VIII.A.3 would require that the NRC find both that the departure is necessary for adequate protection or for compliance and that special circumstances are present. Paragraph VIII.A.4 would provide that exemptions from Tier 1 requested by an applicant or licensee are governed by the requirements of §§ 52.63(b)(1) and 52.98(f), which provide an opportunity for a hearing. In addition, the NRC would not grant requests for exemptions that may result in a significant decrease in the level of safety otherwise provided by the design.

Tier 2 Information

Paragraph B of Section VIII describes the change processes for the Tier 2 information; which have the same elements as the Tier 1 change process, but some of the standards for plant-specific orders and exemptions would be different. Generic Tier 2 changes would be accomplished by rulemaking that would amend the generic DCD and would be governed by the standards in § 52.63(a)(1). A generic change under § 52.63(a)(1) would not be made to a certified design while it is in effect unless the change: (1) is necessary for compliance with NRC regulations that were applicable and in effect at the time the certification was issued; (2) is necessary to provide adequate protection of the public health and safety or common defense and security; (3) reduces unnecessary regulatory burden and maintains protection to public health and safety and common defense and security; (4) provides the detailed design information necessary to resolve select design acceptance criteria; (5) corrects material errors in the certification information; (6) substantially increases overall safety, reliability, or security of a facility and the costs of the change are justified; or (7) contributes to increased standardization of the certification information.

Departures from Tier 2 would occur in four ways: (1) the NRC may order a plant-specific departure, as set forth in paragraph VIII.B.3; (2) an applicant or licensee may request an exemption from a Tier 2 requirement as set forth in paragraph VIII.B.4; (3) a licensee may make a departure without prior NRC approval under paragraph VIII.B.5; or (4) the licensee may request NRC approval for proposed departures which do not meet the requirements in paragraph VIII.B.5 as provided in paragraph VIII.B.5.e.

Similar to ordered Tier 1 departures and generic Tier 2 changes, ordered Tier 2 departures could not be imposed except when necessary, either to bring the certification into compliance with the NRC's regulations applicable and in effect at the time of approval of the design certification or to ensure adequate protection of the public health and safety or common defense and security, as set forth in paragraph VIII.B.3. However, unlike Tier 1 changes, the special circumstances for the ordered Tier 2 departures would not have to outweigh any decrease in safety that may result from the reduction in standardization caused by the plant-specific order, as required by § 52.63(a)(4). The NRC has determined that it is not necessary to impose an additional limitation similar to that imposed on Tier 1 departures by § 52.63(a)(4) and (b)(1). This type of additional limitation for standardization would unnecessarily restrict the flexibility of applicants and licensees with respect to Tier 2 information.

An applicant or licensee would be permitted to request an exemption from Tier 2 information as set forth in paragraph VIII.B.4. The applicant or licensee would have to demonstrate that the exemption complies with one of the special circumstances in regulations governing specific exemptions in § 50.12(a). In addition, the NRC would not grant requests for exemptions that may result in a significant decrease in the level of safety otherwise provided by the design. However, unlike Tier 1 changes, the special circumstances for the exemption do not have to outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. If the exemption is requested by an applicant for a license, the exemption would be subject to litigation in the same manner as other issues in the licensing hearing, consistent with § 52.63(b)(1). If the exemption is requested by a licensee, then the exemption would be subject to litigation in the same manner as a license amendment.

Paragraph VIII.B.5 would allow an applicant or licensee to depart from Tier 2 information, without prior NRC approval, if it does not involve a change to, or departure from, Tier 1 information, technical specification, or does not require a license amendment under paragraphs VIII.B.5.b or c. The technical specifications referred to in VIII.B.5.a of this paragraph are the technical specifications in Chapter 16 of the generic

DCD, including bases, for departures made prior to the issuance of the COL. After the issuance of the COL, the plant-specific technical specifications would be controlling under paragraph VIII.B.5. The requirement for a license amendment in paragraph VIII.B.5.b would be similar to the requirement in § 50.59 and would apply to all of the information in Tier 2 except for the information that resolves the severe accident issues or that affects information required by § 52.47(a)(28) to address aircraft impacts.

Paragraph VIII.B.5.b addresses information described in the DCD to address aircraft impacts, in accordance with § 52.47(a)(28). Under § 52.47(a)(28), applicants are required to include the information required by § 50.150(b) in their DCD. An applicant or licensee who changes this information is required to consider the effect of the changed design feature or functional capability on the original aircraft impact assessment required by § 50.150(a). The applicant or licensee is also required to describe in the plant-specific DCD how the modified design features and functional capabilities continue to meet the assessment requirements in § 50.150(a)(1). Submittal of this updated information is governed by the reporting requirements in Section X.B.

During an ongoing adjudicatory proceeding (e.g., for issuance of a COL) a party who believes that an applicant or licensee has not complied with paragraph VIII.B.5 when departing from Tier 2 information may petition to admit such a contention into the proceeding under paragraph VIII.B.5.g. As set forth in paragraph VIII.B.5.g, the petition would have to comply with the requirements of § 2.309 and show that the departure does not comply with paragraph VIII.B.5. If on the basis of the petition and any responses thereto, the presiding officer in the proceeding determines that the required showing has been made, the matter would be certified to the Commission for its final determination. In the absence of a proceeding, assertions of nonconformance with paragraph VIII.B.5 requirements applicable to Tier 2 departures would be treated as petitions for enforcement action under § 2.206.

Operational Requirements

The change process for technical specifications and other operational requirements that were reviewed and approved in the design certification rule is set forth in Section VIII, paragraph C. The key to using the change processes described in Section VIII is to determine if the proposed change or departure would require a change to a design feature described in the generic DCD. If a design change is required, then the appropriate change process in paragraph VIII.A or VIII.B would apply. However, if a proposed change to the technical specifications or other operational requirements does not require a change to a design feature in the generic DCD, then paragraph VIII.C would apply. This change process has elements similar to the Tier 1 and Tier 2 change processes in paragraphs VIII.A and VIII.B, but with significantly different change standards. Because of the different finality status for technical specifications and other operational requirements, the NRC designated a special category of information, consisting of the technical specifications and other operational requirements, with its own change process in paragraph VIII.C. The language in paragraph VIII.C also distinguishes between generic (Chapter 16 of the DCD) and plant-specific technical specifications to account for the different treatment and finality consistent with technical specifications before and after a license is issued.

The process in paragraph VIII.C.1 for making generic changes to the generic technical specifications in Chapter 16 of the DCD or other operational requirements in the generic DCD would be accomplished by rulemaking and governed by the backfit standards in § 50.109. The determination of whether the generic technical specifications and other operational requirements were completely reviewed and approved in the design certification rule would be based upon the extent to which the NRC reached a safety conclusion in the final safety evaluation report on this matter. If a technical specification or operational requirement was completely reviewed and finalized in the design certification rule, then the requirement of § 50.109 would apply because a position was taken on that safety matter. Generic changes made under paragraph VIII.C.1 would be applicable to all applicants or licensees (refer to paragraph VIII.C.2), unless the change is irrelevant because of a plant-specific departure.

Some generic technical specifications contain values in brackets [ ]. The brackets are placeholders indicating that the NRC's review is not complete, and represent a requirement that the applicant for a COL referencing the NuScale design certification rule must replace the values in brackets with final plant-specific values (refer to guidance provided in Regulatory Guide 1.206, Revision 1, "Applications for Nuclear Power Plants," dated October 2018 (ADAMS Accession No. ML18131A181)). The values in brackets are neither part of the design certification rule nor are they binding. Therefore, the replacement of bracketed values with final plant-specific values does not require an exemption from the generic technical specifications.

Plant-specific departures may occur by either an order under paragraph VIII.C.3 or an applicant's exemption request under paragraph VIII.C.4. The basis for determining if the technical specification or operational requirement was completely reviewed and approved for these processes would be the same as for paragraph VIII.C.1 previously discussed. If the technical specifications or operational requirement was comprehensively reviewed and finalized in the design certification rule, then the NRC must demonstrate that special circumstances are present before ordering a plant-specific departure. If not, there would be no restriction on plant-specific changes to the technical specifications or operational requirements, prior to the issuance of a license, provided a design change is not required. Although the generic technical specifications were reviewed and approved by the NRC in support of the design certification review, the NRC intends to consider the lessons learned from subsequent operating experience during its licensing review of the plant-specific technical specifications. The process for petitioning to intervene on a technical specification or operational requirement contained in paragraph VIII.C.5 would be similar to other issues in a licensing hearing, except that the petitioner must also demonstrate why special circumstances are present pursuant to § 2.335.

Paragraph VIII.C.6 states that the generic technical specifications would have no further effect on the plant-specific technical specifications after the issuance of a license that references this appendix and the change process. After a license is issued, the bases for the plant-specific technical specification would be controlled by the bases change provision set forth in the administrative controls section of the plant-specific technical specifications.

I. [RESERVED] (Section IX)

This section is reserved for future use. The matters discussed in this section of earlier design certification rules—inspections, tests, analyses, and acceptance criteria—are now addressed in the substantive provisions of 10 CFR part 52. Accordingly, there is no need to repeat these regulatory provisions in the NuScale design certification rule. However, this section is being reserved to maintain consistent section numbering with other design certification rules.

J. Records and Reporting (Section X)

The purpose of Section X of appendix G to 10 CFR part 52 is to set forth the requirements that will apply to maintaining records of changes to and departures from the generic DCD, which are to be reflected in the plant-specific DCD. Section X also sets forth the requirements for submitting reports (including updates to the plant-specific DCD) to the NRC. This section of appendix G to 10 CFR part 52 is similar to the requirements for records and reports in 10 CFR part 50, except for minor differences in information collection and reporting requirements.

Paragraph X.A.1 requires that a generic DCD including SUNSI and SGI referenced in the generic DCD be maintained by the applicant for this proposed rule. The generic DCD concept was developed, in part, to meet the requirements for incorporation by reference, including public availability of documents incorporated by reference. However, the SUNSI and SGI could not be included in the generic DCD because they are not publicly available. Nonetheless, the SUNSI and SGI were reviewed by the NRC and, as stated in paragraph VI.B.2, the NRC would consider the information to be resolved within the meaning of § 52.63(a)(5). Because this information is not in the generic DCD, this information, or its equivalent, is required to be provided by an applicant for a license referencing this design certification rule. Only the generic DCD is identified and incorporated by reference into this rule. The generic DCD and the NRC approved version of the SUNSI and SGI must be maintained by the applicant (NuScale Power) for the period of time that appendix G to 10 CFR part 52 may be referenced.

Paragraphs X.A.2 and X.A.3 place recordkeeping requirements on the applicant or licensee that reference this design certification so that its plant-specific DCD accurately reflects both generic changes to the generic DCD and plant-specific departures made under Section VIII. The term “plant-specific” is used in paragraph X.A.2 and other sections of appendix G to 10 CFR part 52 to distinguish between the generic DCD that would be incorporated by reference into appendix G to 10 CFR part 52, and the plant-specific DCD that the COL applicant is required to submit under paragraph IV.A. The requirement to maintain changes to the generic DCD is explicitly stated to ensure that these changes are not only reflected in the generic DCD, which will be maintained by the applicant for the design certification, but also in the plant-specific DCD. Therefore, records of generic changes to the DCD will be required to be maintained by both entities to ensure that both entities have up-to-date DCDs.

Paragraph X.A.4.a requires the design certification rule applicant to maintain a copy of the aircraft impact assessment analysis for the term of the certification and any renewal. This provision, which is consistent with § 50.150(c)(3), would facilitate any NRC inspections of the assessment that the NRC decides to conduct. Similarly, paragraph X.A.4.b requires an applicant or licensee who references appendix G to 10 CFR part 52 to maintain a copy of the aircraft impact assessment performed to comply with the requirements of § 50.150(a) throughout the pendency of the application and for the term of the license and any renewal. This provision is consistent with § 50.150(c)(4). For all applicants and licensees, the supporting documentation retained should describe the methodology used in performing the assessment, including the identification of potential design features and functional capabilities to show that the acceptance criteria in § 50.150(a)(1) will be met.

Paragraph X.A does not place recordkeeping requirements on site specific information that is outside the scope of this rule. As discussed in paragraph V.D of this document, the final safety analysis report required by § 52.79 will contain the plant-specific DCD and the site-specific information for a facility that references this rule. The phrase "site specific portion of the final safety analysis report" in paragraph X.B.3.c refers to the information that is contained in the final safety analysis report for a facility (required by § 52.79), but is not part of the plant-specific DCD (required by paragraph IV.A). Therefore, this proposed rule does not require that duplicate documentation be maintained by an applicant or licensee that references this rule because the plant-specific DCD is part of the final safety analysis report for the facility.

Paragraph X.B.1 requires applicants or licensees that reference this rule to submit reports that describe departures from the DCD and include a summary of the written evaluations. The requirement for the written evaluations is set forth in paragraph X.A.1. The frequency of the report submittals is set forth in paragraph X.B.3. The requirement for submitting a summary of the evaluations will be similar to the requirement in § 50.59(d)(2).

Paragraph X.B.2 requires applicants or licensees that reference this rule to submit updates to the DCD, which include both generic changes and plant-specific departures, as set forth in paragraph X.B.3. The requirements in paragraph X.B.3 for submitting reports will vary according to certain time periods during a facility's lifetime. If a potential applicant for a COL that references this rule decides to depart from the generic DCD prior to submission of the application, then paragraph X.B.3.a will require that the updated DCD be submitted as part of the initial application for a license. Under paragraph X.B.3.b, the applicant may submit any subsequent updates to its plant-specific DCD along with its amendments to the application provided that the submittals are made at least once per year. Because amendments to an application are typically made more frequently than once a year, this should not be an excessive burden on the applicant.

Paragraph X.B.3.b also requires semi-annual submission of the reports required by paragraph X.B.1 throughout the period of application review and construction. The NRC will use the information in the reports to support planning for the NRC's inspection and oversight during this phase, when the licensee is conducting detailed design, procurement of components and equipment, construction, and preoperational testing. In addition, the NRC will use the information in making its finding on ITAAC under § 52.103(g), as well as any finding on interim operation under Section 189.a(1)(B)(iii) of the Atomic Energy Act of 1954, as amended. Once a facility begins operation (for a COL under 10 CFR part 52, after the Commission has made a finding under

§ 52.103(g)), the frequency of reporting will be governed by the requirements in paragraph X.B.3.c.

VI. Section-by-Section Analysis

The following paragraphs describe the specific changes of this proposed rule:

Section 52.11, Information collection requirements: Office of Management and Budget (OMB) approval.

In § 52.11, this proposed rule would add new appendix G to 10 CFR part 52 to the list of information collection requirements in paragraph (b) of this section.

Appendix G to Part 52—Design Certification Rule for the NuScale Standard Design

This proposed rule would add appendix G to 10 CFR part 52 to incorporate the NuScale standard design into the NRC's regulations. Applicants intending to construct and operate a plant using NuScale may do so by referencing the design certification rule.

VII. Regulatory Flexibility Certification

Under the Regulatory Flexibility Act (5 U.S.C. 605(b)), the NRC certifies that this rule, if promulgated, will not have a significant economic impact on a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of “small entities” set forth in the Regulatory Flexibility Act or the size standards established by the NRC (§ 2.810).

VIII. Regulatory Analysis

The NRC has not prepared a regulatory analysis for this proposed rule. The NRC prepares regulatory analyses for rulemakings that establish generic regulatory requirements applicable to all licensees. Design certifications are not generic rulemakings in the sense that design certifications do not establish standards or requirements with which all licensees must comply. Rather, design certifications are NRC approvals of specific nuclear power plant designs by rulemaking, which then may be voluntarily referenced by applicants for combined licenses. Furthermore, design certification rules are requested by an applicant for a design certification, rather than the NRC. Preparation of a regulatory analysis in this circumstance would not be useful because the design to be certified is proposed by the applicant rather than the NRC. For these reasons, the NRC concludes that preparation of a regulatory analysis is neither required nor appropriate.

IX. Backfitting and Issue Finality

The NRC has determined that this proposed rule does not constitute a backfit as defined in the backfit rule (§ 50.109), and that it is not inconsistent with any applicable issue finality provision in 10 CFR part 52.

This initial design certification rule does not constitute backfitting as defined in the backfit rule (§ 50.109) because there are no operating licenses under 10 CFR part 50 referencing this design certification proposed rule.

This initial design certification rule is not inconsistent with any applicable issue finality provision in 10 CFR part 52 because it does not impose new or changed requirements on existing design certification rules in appendices A through F to 10 CFR part 52, and no combined licenses, construction permits, or manufacturing licenses issued by the NRC at this time reference this design certification proposed rule.

For these reasons, neither a backfit analysis nor a discussion addressing the issue finality provisions in 10 CFR part 52 was prepared for this proposed rule.

X. Plain Writing

The Plain Writing Act of 2010 (Pub. L. 111-274) requires Federal agencies to write documents in a clear, concise, well-organized manner that also follows other best practices appropriate to the subject or field and the intended audience. The NRC has written this document to be consistent with the Plain Writing Act as well as the Presidential Memorandum, "Plain Language in Government Writing," published June 10, 1998 (63 FR 31883). The NRC requests comment on the proposed rule with respect to clarity and effectiveness of the language used.

XI. Environmental Assessment and Finding of No Significant Impact

The NRC conducted an environmental assessment (ADAMS Accession No. ML19303C179) and has determined under the National Environmental Policy Act of 1969, as amended, and the NRC's regulations in subpart A of 10 CFR part 51, that this proposed rule, if adopted, would not be a major Federal action significantly affecting the quality of the human environment and, therefore, an environmental impact statement is not required. The NRC's generic determination in this regard is reflected in

§ 51.32(b)(1). The Commission has determined in § 51.32 that there is no significant environmental impact associated with the issuance of a standard design certification or a design certification amendment, as applicable. Comments on the environmental assessment will be limited to the consideration of severe accident mitigation design alternatives as required by § 51.30(d).

The basis for the NRC's categorical exclusion in this regard, as discussed in the 2007 final rule amending 10 CFR parts 51 and 52 (72 FR 49352; August 28, 2007), is based upon the following considerations. A design certification rule does not authorize the siting, construction, or operation of a facility referencing any particular design; it only codifies the NuScale design in a rule. The NRC will evaluate the environmental impacts and issue an environmental impact statement as appropriate under the National Environmental Policy Act as part of the application for the construction and operation of a facility referencing any particular DC rule.

In addition, consistent with § 51.30(d) and § 51.32(b), the NRC has prepared an environmental assessment (ADAMS Accession No. ML19303C179) for the NuScale design addressing various design alternatives to prevent and mitigate severe accidents. The environmental assessment is based, in part, upon the NRC's review of NuScale Power's evaluation of various design alternatives to prevent and mitigate severe accidents in Revision 5 of the DCA Part 3, "Application Applicant's Environmental Report - Standard Design Certification" (ADAMS Accession No. ML20224A512). Based on a review of NuScale Power's evaluation, the NRC concludes that: (1) NuScale Power identified a reasonably complete set of potential design alternatives to prevent and mitigate severe accidents for the NuScale design and (2) none of the potential design alternatives appropriate at the design certification stage are justified on the basis of cost-benefit considerations. These issues are considered resolved for the NuScale design.

Based on its own independent evaluation, the NRC concluded that none of the possible candidate design alternatives appropriate at this design certification stage are potentially cost beneficial for NuScale for accident events. This independent evaluation was based on reasonable treatment of costs, benefits, and sensitivities. The NRC's conclusion is applicable for sites with site characteristics that fall within those site parameters specified in the NuScale environmental report. The NRC concludes that NuScale Power has adequately identified areas appropriate at this design certification stage where risk potentially could be reduced in a cost beneficial manner and that NuScale Power has adequately assessed whether the implementation of the identified potential severe accident mitigation design alternatives (SAMDAs) or candidate design alternatives would be cost beneficial for the given site parameters. Site-specific SAMDAs, multi-unit aspects, procedural and training SAMDAs, and the reactor building crane design would need to be assessed when a specific site is proposed for constructing and operating a NuScale power plant.

The determination of this environmental assessment is that there will be no significant offsite impact to the public from this action. The environmental assessment is available as indicated under Section XVI of this proposed rule.

XII. Paperwork Reduction Act

This proposed rule contains new or amended collections of information subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq). This proposed rule has been submitted to the OMB for review and approval of the information collections.

Type of submission: Revision.

The title of the information collection: Appendix G to 10 CFR part 52 Design Certification Rule for NuScale.

The form number if applicable: NA.

How often the collection is required or requested: On occasion

Who will be required or asked to respond: Applicant for a combined license, construction permit, or a design certification amendment.

An estimate of the number of annual responses: 5 (2 annual responses and 3 recordkeepers).

The estimated number of annual respondents: 3.

An estimate of the total number of hours needed annually to comply with the information collection requirement or request: 389 hours (346 reporting hours + 43 recordkeeping hours).

Abstract: The NRC is proposing to amend its regulations to certify the NuScale standard design. This action is necessary so that applicants or licensees intending to construct and operate an NuScale standard design may do so by referencing this design certification rule. The applicant for certification of the NuScale standard design is NuScale Power, LLC.

The NRC is seeking public comment on the potential impact of the information collection contained in this proposed rule and on the following issues:

1) Is the proposed information collection necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?

2) Is the estimate of the burden of the proposed information collection accurate?

3) Is there a way to enhance the quality, utility, and clarity of the information to be collected?

4) How can the burden of the proposed information collection on respondents be minimized, including the use of automated collection techniques or other forms of information technology?

A copy of the OMB clearance package is available in ADAMS under Accession

No. ML20242A000 or can be obtained free of charge by contacting the NRC's Public Document Room reference staff at 1-800-397-4209, at 301-415-4737, or by e-mail to PDR.resource@nrc.gov. You may obtain information and comment submissions related to the OMB clearance package by searching on https://www.regulations.gov under Docket ID NRC-2017-0029.

You may submit comments on any aspect of these proposed information collection(s), including suggestions for reducing the burden and on the above issues, by the following methods:

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID NRC-2017-0029.

- **Mail comments to:** FOIA, Library, and Information Collections Branch, Office of the Chief Information Officer, Mail Stop: T6-A10M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001 or to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0151), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW, Washington, DC 20503; e-mail: oira_submission@omb.eop.gov.

Additionally, this proposed rule provides procedures for requesting access to proprietary and safeguards information for preparation of comments on the NuScale design certification proposed rule. These procedures are guidance for completing mandatory information collections located in 10 CFR parts 9 and 73 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by OMB under approval numbers 3150-0043 and 3150-0002. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch (T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555 0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the

OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0043 and 3150-0002), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW, Washington, DC 20503; e-mail: oira_submission@omb.eop.gov.

Submit comments by **[INSERT DATE 60 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**. Comments received after this date will be considered if it is practical to do so, but the NRC is able to ensure consideration only for comments received on or before this date.

Deleted: 30

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

XIII. Agreement State Compatibility

Under the "Policy Statement on Adequacy and Compatibility of Agreement States Programs," approved by the Commission on June 20, 1997, and published in the *Federal Register* (62 FR 46517; September 3, 1997), this proposed rule is classified as compatibility "NRC." Compatibility is not required for Category "NRC" regulations. The NRC program elements in this category are those that relate directly to areas of regulation reserved to the NRC by the Atomic Energy Act or the provisions of 10 CFR, and although an Agreement State may not adopt program elements reserved to the NRC, it may wish to inform its licensees of certain requirements by a mechanism that is consistent with a particular State's administrative procedure laws, but does not confer regulatory authority on the State.

XIV. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this proposed rule, the NRC intends to certify the NuScale standard design for use in nuclear power plant licensing under 10 CFR parts 50 or 52. Design certifications are not generic rulemakings establishing a generally applicable standard with which all 10 CFR parts 50 and 52 nuclear power plant licensees must comply. Design certifications are Commission approvals of specific nuclear power plant designs by rulemaking. Furthermore, design certifications are initiated by an applicant for rulemaking, rather than by the NRC. This action does not constitute the establishment of a standard that contains generally applicable requirements.

XV. Availability of Documents

The documents identified in the following table are available to interested persons through one or more of the following methods, as indicated.

DOCUMENT	ADAMS ACCESSION NO.
SECY-XX-XXXX, "Proposed Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029)"	ML19353A003

Staff Requirements Memorandum for SECY-XX-XXXX, “Proposed Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029)”	MLXXXXXXXXX
NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application (NRC Project No. 0769) (December 2016)	ML17013A229
NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application, Revision 5 (July 2020)	ML20225A071
NuScale DCA Final Safety Evaluation Reports (August 2020)	ML20023A318
NuScale Standard Design Certification Application, Part 3, “Applicant’s Environmental Report - Standard Design Certification,” Revision 5 (July 2020)	ML20224A512
Environmental Assessment by the U.S. Nuclear Regulatory Commission Relating to the Certification of the NuScale Standard Design	ML19303C179
Regulatory History of Design Certification (April 2000)[2]	ML003761500
NuScale Technical and Topical Reports	
ES-0304-1381-NP, Human-System Interface Style Guide, Rev. 4 (December 2019)	ML19338E948
RP-0215-10815-NP, Concept of Operations, Rev. 3 (May 2019)	ML19133A293
RP-0316-17614-NP, Human Factors Engineering Operating Experience Review Results Summary Report, Rev. 0 (December 2016)	ML16364A342
RP-0316-17615-NP, Human Factors Engineering Functional Requirements Analysis and Function Allocation Results Summary Report, Rev. 0 (December 2016)	ML16364A342
RP-0316-17616-NP, Human Factors Engineering Task Analysis Results Summary Report, Rev. 2 (April 2019)	ML19119A393
RP-0316-17617-NP, Human Factors Engineering Staffing and Qualifications Results Summary Report, Rev. 0 (December 2016)	ML17004A222

Commented [NT2]: Please check this ADAMS Accession No.

[2] The regulatory history of the NRC’s design certification reviews is a package of documents that is available in the NRC’s PDR and NRC Library. This history spans the period during which the NRC simultaneously developed the regulatory standards for reviewing these designs and the form and content of the rules that certified the designs.

RP-0316-17618-NP, Human Factors Engineering Treatment of Important Human Actions Results Summary Report, Rev. 0 (December 2016)	ML17004A222
RP-0316-17619-NP, Human Factors Engineering Human-System Interface Design Results Summary Report, Rev. 2, (April 2019)	ML19119A398
RP-0516-49116-NP, Control Room Staffing Plan Validation Results, Rev. 1 (December 2016)	ML16364A356
RP-0914-8534-NP, Human Factors Engineering Program Management Plan, Rev. 5 (April 2019)	ML19119A342
RP-0914-8543-NP, Human Factors Verification and Validation Implementation Plan, Rev. 5 (April 2019)	ML19119A372
RP-0914-8544-NP, Human Factors Engineering Design Implementation Implementation Plan, Rev. 4 (November 2019)	ML19331A910
RP-1018-61289-NP, Human Factors Engineering Verification and Validation Results Summary Report, Rev. 1 (July 2019)	ML19212A773
RP-1215-20253-NP, Control Room Staffing Plan Validation Methodology, Rev. 3 (December 2016)	ML16364A353
TR-0116-20781-NP, Fluence Calculation Methodology and Results, Rev. 1 (July 2019)	ML19183A485
TR-0116-20825-NP-A, Applicability of AREVA Fuel Methodology for the NuScale Design, Rev. 1 (February 2018)	ML18040B306
TR-0116-21012-NP-A, NuScale Power Critical Heat Flux Correlations, Rev. 1 (December 2018)	ML18360A632
TR-0316-22048-NP, Nuclear Steam Supply System Advanced Sensor Technical Report, Rev. 3 (May 2020)	ML20141M764
TR-0515-13952-NP-A, Risk Significance Determination, Rev. 0 (October 2016)	ML16284A016
TR-0516-49084-NP, Containment Response Analysis Methodology Technical Report, Rev. 3 (May 2020)	ML20141L808
TR-0516-49416-NP-A, Non-Loss-of-Coolant Accident Analysis Methodology, Rev. 3 (July 2020)	ML20191A281
TR-0516-49417-NP-A, Evaluation Methodology for Stability Analysis of the NuScale Power Module, Rev. 1 (March 2020)	ML20078Q094
TR-0516-49422-NP-A, Loss-of-Coolant Accident Evaluation Model, Rev. 2 (July 2020)	ML20189A644

TR-0616-48793-NP-A, Nuclear Analysis Codes and Methods Qualification, Rev. 1 (December 2018)	ML18348B036
TR-0616-49121-NP, NuScale Instrument Setpoint Methodology Technical Report, Rev. 3 (May 2020)	ML20141M114
TR-0716-50350-NP-A, Rod Ejection Accident Methodology, Rev. 1 (June 2020)	ML20168B203
TR-0716-50351-NP-A, NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces, Rev. 1 (May 2020)	ML20122A248
TR-0716-50424-NP, Combustible Gas Control, Rev. 1 (March 2019)	ML19091A232
TR-0716-50439-NP, NuScale Comprehensive Vibration Assessment Program Analysis Technical Report, Rev. 2 (July 2019)	ML19212A776
TR-0815-16497-NP-A, Safety Classification of Passive Nuclear Power Plant Electrical Systems Topical Report, Rev. 1 (February 2018)	ML18054B607
TR-0816-49833-NP, Fuel Storage Rack Analysis, Rev. 1 (November 2018)	ML18310A154
TR-0816-50796-NP, Loss of Large Areas Due to Explosions and Fires Assessment, Rev. 1 (June 2019)	ML19165A294
TR-0816-50797 (NuScale Nonproprietary), Mitigation Strategies for Loss of All AC Power Event, Rev. 3 (October 2019)	ML19302H598
TR-0816-51127-NP, NuFuel-HTP2™ Fuel and Control Rod Assembly Designs, Rev. 3 (December 2019)	ML19353A719
TR-0818-61384-NP, Pipe Rupture Hazards Analysis, Rev. 2 (July 2019)	ML19212A682
TR-0915-17564-NP-A, Subchannel Analysis Methodology, Rev. 2 (March 2019)	ML19067A256
TR-0915-17565-NP-A, Accident Source Term Methodology, Rev. 4 (February 2020)	ML20057G132
TR-0916-51299-NP, Long-Term Cooling Methodology, Rev. 3 (May 2020)	ML20141L816
TR-0916-51502-NP, NuScale Power Module Seismic Analysis, Rev. 2 (April 2019)	ML19093B850
TR-0917-56119-NP, CNV Ultimate Pressure Integrity, Rev. 1 (June 2019)	ML19158A382
TR-0918-60894-NP, Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report, Rev, 1 (August 2019)	ML19214A248

TR-1010-859-NP-A, NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant, Rev. 5 (June 2020)	ML20176A494
TR-1015-18177-NP, Pressure and Temperature Limits Methodology, Rev. 2 (October 2018)	ML18298A304
TR-1015-18653-NP-A, Design of the Highly Integrated Protection System Platform Topical Report, Rev. 2 (September 2017)	ML17256A892
TR-1016-51669-NP, NuScale Power Module Short-Term Transient Analysis, Rev. 1 (July 2019)	ML19211D411
TR-1116-51962-NP, NuScale Containment Leakage Integrity Assurance Technical Report, Rev. 1 (May 2019)	ML19149A298
TR-1116-52065-NP, Effluent Release (GALE Replacement) Methodology and Results, Rev. 1 (November 2018)	ML18317A364

The NRC may post materials related to this document, including public comments, on the Federal Rulemaking Web site at https://www.regulations.gov under Docket ID NRC-2017-0029.

XVI. Procedures for Access to Proprietary and Safeguards Information for Preparation of Comments on the NuScale Design Certification Proposed Rule

This section contains instructions regarding how the non-publicly available documents related to this rule, and specifically those listed in Table 1.6-1 and 1.6-2 beginning on page 1.6-2 of Tier 2 of the DCD, may be accessed by interested persons who wish to comment on the design certification. These documents contain proprietary information and safeguards information (SGI). Requirements for access to SGI are primarily set forth in 10 CFR parts 2 and 73. This section provides information specific to this proposed rule; however, nothing in this section is intended to conflict with the SGI regulations.

Interested persons who desire access to proprietary information on NuScale should first request access to that information from NuScale Power, LLC, the design certification applicant. Requests to the applicant must be sent to NuScale Power, LLC, 6650 SW Redwood Lane, Suite 210, Portland, Oregon 97224. A request for access should be submitted to the NRC if the applicant does not either grant or deny access by the 10-day deadline described in the following section.

One of the non-publicly available documents, TR-0416-48929, "NuScale Design of Physical Security Systems," contains both proprietary information and SGI. If you need access to proprietary information in that document in order to develop comments within the scope of this rule, then your request for access should first be submitted to NuScale Power, in accordance with the previous paragraph. By contrast, if you need access to the SGI in order to provide comments, then your request for access to the SGI must be submitted to the NRC as described further in this section. Therefore, if you need access to both proprietary information and SGI in that document, then you should request access to the information in separate requests submitted to both NuScale Power and the NRC.

Submitting a Request to the NRC for Access

Within 10 days after publication of this proposed rule, any individual or entity who believes access to proprietary information or SGI is necessary in order to submit comments on this proposed rule may request access to such information. Requests for access to proprietary information or SGI submitted more than 10 days after publication of this document will not be considered absent a showing of good cause for the late filing explaining why the request could not have been filed earlier.

The requestor shall submit a letter requesting permission to access proprietary information and/or SGI to the Office of the Secretary, U.S. Nuclear Regulatory Commission, Attention: Rulemakings and Adjudications Staff, Washington, DC 20555–0001. The e-mail address for the Office of the Secretary is Rulemaking.Comments@nrc.gov. The requester must send a copy of the request to the design certification applicant at the same time as the original transmission to the NRC using the same method of transmission. Requests to the applicant must be sent to NuScale Power, LLC, 6650 SW Redwood Lane, Suite 210, Portland, Oregon 97224.

The request must include the following information:

(1) The name of this design certification, NuScale Design Certification; the rulemaking identification number, RIN 3150–AJ98; the rulemaking docket number, NRC–2017–0029; and the *Federal Register* citation for this rule.

(2) The name and address of the requester.

(3) The identity of the individual(s) to whom access is to be provided, including the identity of any expert, consultant, or assistant who will aid the requestor in evaluating the information.

(4) If the request is for proprietary information, the requester's need for the information in order to prepare meaningful comments on the design certification must be demonstrated. Each of the following areas must be addressed with specificity:

   (a) The specific issue or subject matter on which the requester wishes to comment.

   (b) An explanation why information which is publicly available is insufficient to provide the basis for developing meaningful comment on the NuScale design certification proposed rule with respect to the issue or subject matter described in paragraph 4.a. of this section.

(c) The technical competence (demonstrable knowledge, skill, training or education) of the requestor to effectively utilize the requested proprietary information to provide the basis for meaningful comment. Technical competence may be shown by reliance on a qualified expert, consultant, or assistant who satisfies these criteria.

(d) A chronology and discussion of the requester's attempts to obtain the information from the design certification applicant, and the final communication from the requester to the applicant and the applicant's response, if any was provided, with respect to the request for access to proprietary information must be submitted.

(5) If the request is for SGI, the request must include the following:

(a) A statement that explains each individual's "need to know" the SGI, as required by §§ 73.2 and 73.22(b)(1). Consistent with the definition of "need to know" as stated in § 73.2, the statement must explain:

(i) Specifically why the requestor believes that the information is necessary to enable the requestor to proffer and/or adjudicate a specific contention in this proceeding;[3] and

(ii) The technical competence (demonstrable knowledge, skill, training or education) of the requestor to effectively utilize the requested SGI to provide the basis and specificity for meaningful comment. Technical competence may be shown by reliance on a

[3] Broad SGI requests under these procedures are unlikely to meet the standard for need to know. Furthermore, NRC redaction of information from requested documents before their release may be appropriate to comport with this requirement. The procedures in this document do not authorize unrestricted disclosure or less scrutiny of a requester's need to know than ordinarily would be applied in connection with either adjudicatory or non-adjudicatory access to SGI.

qualified expert, consultant, or assistant who satisfies these criteria.

(b) A completed Form SF-85, “Questionnaire for Non-Sensitive Positions,” for each individual who would have access to SGI. The completed Form SF-85 will be used by the Office of Administration to conduct the background check required for access to SGI, as required by 10 CFR part 2, subpart C, and § 73.22(b)(2), to determine the requestor’s trustworthiness and reliability. For security reasons, Form SF-85 can be submitted only electronically through the Electronic Questionnaires for Investigations Processing website, a secure website that is owned and operated by the Defense Counterintelligence and Security Agency (DCSA). To obtain online access to the form, the requestor should contact the NRC’s Office of Administration at 301-415-3710.[4]

(c) A completed Form FD-258 (fingerprint card), signed in original ink, and submitted in accordance with § 73.57(d). Copies of Form FD-258 may be obtained by sending an e-mail to MAILSVC.Resource@nrc.gov or by sending a written request to U.S. Nuclear Regulatory Commission, Attn: Mailroom/Fingerprint Card Request, 11555 Rockville Pike, Rockville, MD 20852. The fingerprint card will be used to satisfy the requirements of 10 CFR part 2, subpart C, § 73.22(b)(1), and Section 149 of the Atomic Energy Act of 1954, as amended, which mandates that all persons with access to SGI must be fingerprinted for an FBI identification and criminal history records check.

[4] The requester will be asked to provide his or her full name, social security number, date and place of birth, telephone number, and e-mail address. After providing this information, the requestor usually should be able to obtain access to the online form within one business day.

(d) A check or money order in the amount of $326.00[5] payable to the U.S. Nuclear Regulatory Commission for each individual for whom the request for access has been submitted; and

(e) If the requester or any individual who will have access to SGI believes they belong to one or more of the categories of individuals that are exempt from the criminal history records check and background check requirements, as stated in § 73.59, the requester should also provide a statement identifying which exemption the requester is invoking, and explaining the requester's basis for believing that the exemption applies. While processing the request, the Office of Administration, Personnel Security Branch, will make a final determination whether the claimed exemption applies. Alternatively, the requester may contact the Office of Administration for an evaluation of their exemption status prior to submitting their request. Persons who are exempt from the background check are not required to complete the SF–85 or Form FD–258; however, all other requirements for access to SGI, including the need to know, are still applicable.

Note: Copies of documents and materials required by paragraphs (5)(b), (c), and (d), of this section must be sent to the following address: U.S. Nuclear Regulatory Commission, ATTN: Personnel Security Branch, Mail Stop TWFN–07D04M, 11555 Rockville Pike, Rockville, MD 20852.

These documents and materials should *not* be included with the request letter to the Office of the Secretary, but the request letter should state that the forms and fees have been submitted as required.

5 This fee is subject to change pursuant to DCSA's adjustable billing rates.

To avoid delays in processing requests for access to SGI, all forms should be reviewed for completeness and accuracy (including legibility) before submitting them to the NRC. The NRC will return incomplete or illegible packages to the sender without processing.

Based on an evaluation of the information submitted under paragraphs (4) or (5) of this section, as applicable, the NRC will determine within 10 days of receipt of the request whether the requester has established a legitimate need for access to proprietary information or need to know the SGI requested.

Determination of Legitimate Need for Access

For proprietary information access requests, if the NRC determines that the requester has established a legitimate need for access to proprietary information, the NRC will notify the requester in writing that access to proprietary information has been granted. The written notification will contain instructions on how the requestor may obtain copies of the requested documents, and any other conditions that may apply to access to those documents. These conditions may include, but are not limited to, the signing of a Non-Disclosure Agreement or Affidavit by each individual who will be granted access.

For requests for access to SGI, if the NRC determines that the requester has established a need to know the SGI, the NRC's Office of Administration will then determine, based upon completion of the background check, whether the proposed recipient is trustworthy and reliable, as required for access to SGI by § 73.22(b). If the NRC's Office of Administration determines that the individual or individuals are trustworthy and reliable, the NRC will promptly notify the requester in writing. The notification will provide the names of approved individuals as well as the conditions under which the SGI will be provided. Those conditions may include, but are not limited to, the signing of a Non-Disclosure Agreement or Affidavit by each individual who will be granted access to SGI.

Release and Storage of SGI

Prior to providing SGI to the requester, the NRC will conduct (as necessary) an inspection to confirm that the recipient's information protection system is sufficient to satisfy the requirements of § 73.22. Alternatively, recipients may opt to view SGI at an approved SGI storage location rather than establish their own SGI protection program to meet SGI protection requirements.

Filing of Comments on the NuScale Design Certification Proposed Rule Based on Non-Public Information

Any comments in this rulemaking proceeding that are based upon the information received as a result of the request made for proprietary or SGI information must be filed by the requester no later than 25 days after receipt of (or access to) that information, or the close of the public comment period, whichever is later.

Review of Denials of Access

If the request for access to proprietary information or SGI is denied by the NRC, either after a determination on requisite need or after a determination on trustworthiness and reliability, the NRC shall promptly notify the requester in writing, briefly stating the reason or reasons for the denial.

Before the Office of Administration makes a final adverse determination regarding the trustworthiness and reliability of the proposed recipient(s) for access to SGI, the Office of Administration, in accordance with § 2.336(f)(1)(iii), must provide the proposed recipient(s) any records that were considered in the trustworthiness and reliability determination, including those required to be provided under § 73.57(e)(1), so that the proposed recipient(s) have an opportunity to correct or explain the record.

The requestor may challenge the NRC's adverse determination with respect to access to proprietary information or with respect to need to know for SGI by filing a challenge within 5 days of receipt of that determination with the NRC's Executive Director for Operations under § 9.29(d).

The requestor may challenge the Office of Administration's final adverse determination with respect to trustworthiness and reliability for access to SGI by filing a request for review in accordance with § 2.336(f)(1)(iv).

XVII. Incorporation by Reference—Reasonable Availability to Interested Parties

The NRC proposes to incorporate by reference the NuScale DCA, Revision 5. As described in the "Discussion" sections of this document, the generic DCD includes Tier 1 and Tier 2 information (including the technical and topical reports referenced in Chapter 1) and generic technical specifications in order to effectively control this information and facilitate its incorporation by reference into the rule. NuScale Power submitted Revision 5 of the DCA to the NRC in July 2020.

The NRC is required by law to obtain approval for incorporation by reference from the Office of the Federal Register (OFR). The OFR's requirements for incorporation by reference are set forth in 1 CFR part 51. The OFR regulations require an agency to include in a proposed rule a discussion of the ways that the materials the agency incorporates by reference are reasonably available to interested parties or how it worked to make those materials reasonably available to interested parties. The discussion in this section complies with the requirement for a proposed rule as set forth in 1 CFR 51.5(a)(1).

The NRC considers “interested parties” to include all potential NRC stakeholders, not only the individuals and entities regulated or otherwise subject to the NRC’s regulatory oversight. These NRC stakeholders are not a homogenous group but vary with respect to the considerations for determining reasonable availability. Therefore, the NRC distinguishes between different classes of interested parties for the purposes of determining whether the material is “reasonably available.” The NRC considers the following to be classes of interested parties in NRC rulemakings with regard to the material to be incorporated by reference:

- Individuals and small entities regulated or otherwise subject to the NRC’s regulatory oversight (this class also includes applicants and potential applicants or licenses and other NRC regulatory approvals) and who are subject to the material to be incorporated by reference by rulemaking. In this context, “small entities” has the same meaning as a “small entity” under § 2.810.
- Large entities otherwise subject to the NRC’s regulatory oversight (this class also includes applicants and potential applicants for licenses and other NRC regulatory approvals) and who are subject to the material to be incorporated by reference by rulemaking. In this context, “large entities” are those which do not qualify as a “small entity” under § 2.810.
- Non-governmental organizations with institutional interests in the matters regulated by the NRC.
- Other Federal agencies, States, and local governmental bodies (within the meaning of § 2.315(c)).

- Federally-recognized and State-recognized[6] Indian tribes.

- Members of the general public (i.e., individual, unaffiliated members of the public who are not regulated or otherwise subject to the NRC's regulatory oversight) who may wish to gain access to the materials which the NRC incorporates by reference by rulemaking in order to participate in the rulemaking process.

The NRC makes the materials incorporated by reference available for inspection to all interested parties, by appointment, at the NRC Technical Library, which is located at Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852; telephone: 301-415-7000; e-mail: Library.Resource@nrc.gov. In addition, as described in Section XVI of this proposed rule, documents related to this proposed rule are available online in the NRC's ADAMS Public Documents collection at https://www.nrc.gov/reading-rm/adams.html.

The NRC concludes that the materials the NRC is incorporating by reference in this proposed rule are reasonably available to all interested parties because the materials are available in multiple ways and in a manner consistent with their interest in the materials.

List of Subjects in 10 CFR Part 52

Administrative practice and procedure, Antitrust, Combined license, Early site permit, Emergency planning, Fees, Incorporation by reference, Inspection, Issue finality, Limited work authorization, Nuclear power plants and reactors, Probabilistic risk

[6] State-recognized Indian tribes are not within the scope of 10 CFR 2.315(c). However, for purposes of the NRC's compliance with 1 CFR 51.5, "interested parties" includes a broad set of stakeholders, including State-recognized Indian tribes.

assessment, Prototype, Reactor siting criteria, Redress of site, Penalties, Reporting and recordkeeping requirements, Standard design, Standard design certification.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; the Nuclear Waste Policy Act of 1982, as amended; and 5 U.S.C. 552 and 553, the NRC proposes the following amendments to 10 CFR part 52:

PART 52 – LICENSES, CERTIFICATIONS, AND APPROVALS FOR NUCLEAR POWER PLANTS

1. The authority citation for part 52 continues to read as follows:

Authority: Atomic Energy Act of 1954, secs. 103, 104, 147, 149, 161, 181, 182, 183, 185, 186, 189, 223, 234 (42 U.S.C. 2133, 2134, 2167, 2169, 2201, 2231, 2232, 2233, 2235, 2236, 2239, 2273, 2282); Energy Reorganization Act of 1974, secs. 201, 202, 206, 211 (42 U.S.C. 5841, 5842, 5846, 5851); 44 U.S.C. 3504 note.

§ 52.11 [Amended]

2. In § 52.11(b), add “G,” in alphabetical order to the list of appendices.

3. Add Appendix G to part 52 to read as follows:

Appendix G to Part 52—Design Certification Rule for NuScale

I. INTRODUCTION

Appendix G constitutes the standard design certification for NuScale, in accordance with 10 CFR part 52, subpart B. The applicant for the standard design certification of NuScale is NuScale Power, LLC.

II. DEFINITIONS

A. *Generic design control document (generic DCD)* means the document containing the Tier 1 and Tier 2 information (including the technical and topical reports referenced in Chapter 1) and generic technical specifications that is incorporated by reference into this appendix.

B. *Generic technical specifications (generic TS)* means the information required by 10 CFR 50.36 and 50.36a for the portion of the plant that is within the scope of this appendix.

C. *Plant-specific DCD* means that portion of the COL final safety analysis report that sets forth both the generic DCD information and any plant-specific changes to generic DCD information.

D. *Tier 1* means the portion of the design-related information contained in the generic DCD that is approved and certified by this appendix (Tier 1 information). The design descriptions, interface requirements, and site parameters are derived from Tier 2 information. Tier 1 information includes:

1. Definitions and general provisions;
2. Design descriptions;
3. Inspections, tests, analyses, and acceptance criteria (ITAAC);
4. Significant site parameters; and
5. Significant interface requirements.

E. *Tier 2* means the portion of the design-related information contained in the generic DCD that is approved but not certified by this appendix (Tier 2 information). Compliance with Tier 2 is required, but generic changes to and plant-specific departures from Tier 2 are governed by Section VIII of this appendix. Compliance with Tier 2 provides a sufficient, but not the only acceptable, method for complying with Tier 1. Compliance methods differing from Tier 2 must satisfy the change process in Section

VIII of this appendix G. Regardless of these differences, an applicant or licensee must meet the requirement in paragraph III.B of this appendix to reference Tier 2 when referencing Tier 1. Tier 2 information includes:

1. Information required by § 52.47(a) and (c), with the exception of generic TS and conceptual design information;

2. Supporting information on the inspections, tests, and analyses that will be performed to demonstrate that the acceptance criteria in the ITAAC have been met; and

3. COL Items (COL license information) identify certain matters that must be addressed in the site-specific portion of the final safety analysis report by an applicant who references this appendix. These items constitute information requirements but are not the only acceptable set of information in the final safety analysis report. An applicant may depart from or omit these items, provided that the departure or omission is identified and justified in the final safety analysis report. After issuance of a construction permit or COL, these items are not requirements for the licensee unless such items are restated in the final safety analysis report.

F. *Departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses* means:

1. Changing any of the elements of the method described in the plant-specific DCD unless the results of the analysis are conservative or essentially the same; or

2. Changing from a method described in the plant-specific DCD to another method unless that method has been approved by the NRC for the intended application.

G. All other terms in this appendix have the meaning set out in 10 CFR 50.2, 10 CFR 52.1, or Section 11 of the Atomic Energy Act of 1954, as amended, as applicable.

III. Scope and Contents

A. Incorporation by reference approval.

NuScale standard design (hereafter referred as NuScale) material is approved for incorporation by reference by the Director of the Office of the Federal Register under 5 U.S.C. 552(a) and 1 CFR part 51, "Incorporation by Reference." You may obtain copies of the generic DCD from NuScale Power, LLC, 6650 SW Redwood Lane, Suite 210, Portland, Oregon 97224. You can view the generic DCD online in the NRC Library at https://www.nrc.gov/reading-rm/adams.html. In ADAMS, search under ADAMS Accession No. ML20225A071. If you do not have access to ADAMS or if you have problems accessing documents located in ADAMS, contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-3747, or by e-mail at PDR.Resource@nrc.gov. Copies of the NuScale materials are available in the ADAMS Public Documents collection. All approved material is available for inspection at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, e-mail at fedreg.legal@nara.gov or go to https://www.archives.gov/federal-register/cfr/ibrlocations.html.

1. NuScale Standard Plant Design Certification Application, Part 2 - Tier 1, Revision 5, July 2020, Certified Design Descriptions and Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC).

Deleted: , Part 2 - Tier 1, Revision 5, July 2020

2. NuScale Standard Plant Design Certification Application, Part 2 - Tier 2, Revision 5, July 2020, including:

a. Chapter One, Introduction and General Description of the Plant.

b. Chapter Two, Site Characteristics and Site Parameters.

c. Chapter Three, Design of Structures, Systems, Components and Equipment.

d. Chapter Four, Reactor.

e. Chapter Five, Reactor Coolant System and Connecting Systems.

f. Chapter Six, Engineered Safety Features.

g. Chapter Seven, Instrumentation and Controls.

h. Chapter Eight, Electric Power.

i. Chapter Nine, Auxiliary Systems.

j. Chapter Ten, Steam and Power Conversion System.

k. Chapter Eleven, Radioactive Waste Management.

l. Chapter Twelve, Radiation Protection.

m. Chapter Thirteen, Conduct of Operations.

n. Chapter Fourteen, Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria.

o. Chapter Fifteen, Transient and Accident Analyses.

p. Chapter Sixteen, Technical Specifications.

q. Chapter Seventeen, Quality Assurance and Reliability Assurance.

r. Chapter Eighteen, Human Factors Engineering.

s. Chapter Nineteen, Probabilistic Risk Assessment and Severe Accident Evaluation.

t. Chapter Twenty, Mitigation of Beyond-Design-Basis Events.

u. Chapter Twenty-One, Multi-Module Design Considerations.

3. NuScale Standard Plant Design Certification Application, Part 4, Volume 1, Revision 5, July 2020, Generic Technical Specifications, NuScale Nuclear Power Plants, Volume 1: Specifications.

4. NuScale Standard Plant Design Certification Application, Part 4, Volume 2, Revision 5, July 2020, Generic Technical Specifications, NuScale Nuclear Power Plants, Volume 2: Bases.

5. ES-0304-1381-NP, Human-System Interface Style Guide, December 2019, Revision 4, Docket: 52-048.

Deleted: DCA

Deleted: .0

Deleted: DCA

Deleted: .0

6. RP-0215-10815-NP, Concept of Operations, May 2019, Revision 3, Docket: 52-048.

7. RP-0316-17614-NP, Human Factors Engineering Operating Experience Review Results Summary Report, 12/07/2016, Revision 0, Docket: PROJ0769.

8. RP-0316-17615-NP, Human Factors Engineering Functional Requirements Analysis and Function Allocation Results Summary Report, 12/2/16, Revision 0, Docket: PROJ0769.

9. RP-0316-17616-NP, Human Factors Engineering Task Analysis Results Summary Report, April 2019, Revision 2, Docket: 52-048.

10. RP-0316-17617-NP, Human Factors Engineering Staffing and Qualifications Results Summary Report, 12/02/2016, Revision 0, Docket: PROJ0769.

11. RP-0316-17618-NP, Human Factors Engineering Treatment of Important Human Actions Results Summary Report, 12/02/2016, Revision 0, Docket: PROJ0769.

12. RP-0316-17619-NP, Human Factors Engineering Human-System Interface Design Results Summary Report, April 2019, Revision 2, Docket: 52-048.

13. RP-0516-49116-NP, Control Room Staffing Plan Validation Results, 12/02/2016, Revision 1, Docket: PROJ0769.

14. RP-0914-8534-NP, Human Factors Engineering Program Management Plan, April 2019, Revision 5, Docket: 52-048.

15. RP-0914-8543-NP, Human Factors Verification and Validation Implementation Plan, April 2019, Revision 5, Docket: 52-048.

16. RP-0914-8544-NP, Human Factors Engineering Design Implementation Implementation Plan, November 2019, Revision 4, Docket: 52-048, NuScale Nonproprietary.

17. RP-1018-61289-NP, Human Factors Engineering Verification and Validation Results Summary Report, July 2019, Revision 1, Docket: 52-048.

18. RP-1215-20253-NP, Control Room Staffing Plan Validation Methodology, 12/02/2016, Revision 3, Docket: PROJ0769.

19. TR-0116-20781-NP, Fluence Calculation Methodology and Results, July 2019, Revision 1, Docket: 52-048.

20. TR-0116-20825-NP-A, Applicability of AREVA Fuel Methodology for the NuScale Design, June 2016, Revision 1, Docket: PROJ0769.

21. TR-0116-21012-NP-A, NuScale Power Critical Heat Flux Correlations, December 2018, Revision 1, Docket: PROJ0769.

22. TR-0316-22048-NP, Nuclear Steam Supply System Advanced Sensor Technical Report, May 2020, Revision 3, Docket: 52-048.

23. TR-0515-13952-NP-A, Risk Significance Determination, October 2016, Revision 0, Docket: PROJ0769, NuScale Nonproprietary.

24. TR-0516-49084-NP, Containment Response Analysis Methodology Technical Report, May 2020, Revision 3, Docket: 52-048.

25. TR-0516-49416-NP-A, Non-Loss-of-Coolant Accident Analysis Methodology, July 2020, Revision 3, Docket: PROJ0769.

26. TR-0516-49417-NP-A, Evaluation Methodology for Stability Analysis of the NuScale Power Module, March 2020, Revision 1, Docket: PROJ0769.

27. TR-0516-49422-NP-A, Loss-of-Coolant Accident Evaluation Model, July 2020, Revision 2, Docket: PROJ0769.

28. TR-0616-48793-NP-A, Nuclear Analysis Codes and Methods Qualification, November 2018, Revision 1, Docket: PROJ0769.

29. TR-0616-49121-NP, NuScale Instrument Setpoint Methodology Technical Report, May 2020, Revision 3, Docket 52-048.

30. TR-0716-50350-NP-A, Rod Ejection Accident Methodology, June 2020, Revision 1, Docket: PROJ0769.

31. TR-0716-50351-NP-A, NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces, April 2020, Revision 1, Docket: PROJ0769.

32. TR-0716-50424-NP, Combustible Gas Control, March 2019, Revision 1, Docket: PROJ0769.

33. TR-0716-50439-NP, NuScale Comprehensive Vibration Assessment Program Analysis Technical Report, July 2019, Revision 2, Docket: 52-048.

34. TR-0815-16497-NP-A, Safety Classification of Passive Nuclear Power Plant Electrical Systems, January 2018, Revision 1, Docket: PROJ0769.

35. TR-0816-49833-NP, Fuel Storage Rack Analysis, November 2018, Revision 1, Docket: 52-048.

36. TR-0816-50796-NP, Loss of Large Areas Due to Explosions and Fires Assessment, June 2019, Revision 1, Docket: 52-048.

37. TR-0816-50797, Mitigation Strategies for Loss of All AC Power Event, October 2019, Revision 3, Docket: 52-048, NuScale Nonproprietary.

38. TR-0816-51127-NP, NuFuel-HTP2™ Fuel and Control Rod Assembly Designs, December 2019, Revision 3, Docket: 52-048.

39. TR-0818-61384-NP, Pipe Rupture Hazards Analysis, July 2019, Revision 2, Docket No.: 52-048.

40. TR-0915-17564-NP-A, Subchannel Analysis Methodology, February 2019, Revision 2, Docket: PROJ0769.

41. TR-0915-17565-NP-A, Accident Source Term Methodology, February 2020, Revision 4, Docket: PROJ0769.

42. TR-0916-51299-NP, Long-Term Cooling Methodology, May 2020, Revision 3, Docket: 52-048.

43. TR-0916-51502-NP, NuScale Power Module Seismic Analysis, April 2019, Revision 2, Docket: 52-048.

44. TR-0917-56119-NP, CNV Ultimate Pressure Integrity, June 2019, Revision 1, Docket No. 52-048.

45. TR-0918-60894-NP, NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report, August 2019, Revision 1, Docket No.: 52-048.

46. NP-TR-1010-859-NP-A, NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant, May, 2020, Revision 5, Docket: PROJ0769, NuScale Nonproprietary.

47. TR-1015-18177-NP, Pressure and Temperature Limits Methodology, October 2018, Revision 2, Docket: 52-048.

48. TR-1015-18653-NP-A, Design of the Highly Integrated Protection System Platform, May 2017, Revision 2, Docket: PROJ0769.

49. TR-1016-51669-NP, NuScale Power Module Short-Term Transient Analysis, July 2019, Revision 1, Docket: 52-048.

50. TR-1116-51962-NP, NuScale Containment Leakage Integrity Assurance, May 2019, Revision 1, Docket: 52-048.

51. TR-1116-52065-NP, Effluent Release (GALE Replacement) Methodology and Results, November 2018, Revision 1, Docket: 52-048.

B.1. An applicant or licensee referencing this appendix, in accordance with Section IV of this appendix, shall incorporate by reference and comply with the requirements of this appendix except as otherwise provided in this appendix.

2. Conceptual design information, as set forth in the design certification application Part 2, Tier 2, Section 1.2, and the discussion of “first principles” contained in

design certification application Part 2, Tier 2, Section 14.3.2 are not incorporated by reference into this appendix.

C. If there is a conflict between Tier 1 and Tier 2 of the DCD, then Tier 1 controls.

D. If there is a conflict between the generic DCD and either the application for the design certification of NuScale or the final safety evaluation report related to certification of the NuScale standard design, then the generic DCD controls.

E. Design activities for structures, systems, and components that are entirely outside the scope of this appendix may be performed using site characteristics, provided the design activities do not affect the DCD or conflict with the interface requirements.

IV. Additional Requirements and Restrictions

A. An applicant for a COL that wishes to reference this appendix shall, in addition to complying with the requirements of §§ 52.77, 52.79, and 52.80, comply with the following requirements:

1. Incorporate by reference, as part of its application, this appendix.

2. Include, as part of its application:

a. A plant-specific DCD containing the same type of information and using the same organization and numbering as the generic DCD for NuScale, either by including or incorporating by reference the generic DCD information, and as modified and supplemented by the applicant’s exemptions and departures;

b. The reports on departures from and updates to the plant-specific DCD required by paragraph X.B of this appendix;

c. Plant-specific TS, consisting of the generic and site-specific TS that are required by 10 CFR 50.36 and 50.36a;

d. Information demonstrating that the site characteristics fall within the site parameters and that the interface requirements have been met;

e. Information that addresses the COL items;

f. Information required by § 52.47(a) that is not within the scope of this appendix;

g. Information demonstrating that necessary shielding to limit radiological dose consistent with the radiation zones specified in design certification application Part 2, Tier 2, Chapter 12, Figure 12.3-1, "Reactor Building Radiation Zone Map," is provided to account for penetrations in the radiation shield wall between the power module bay and the reactor building steam gallery area;

h. Information demonstrating that the requirements of 10 CFR 50.34(f)(2)(xxviii) are met with respect to potential radiological releases under accident conditions from the systems used for post-accident hydrogen and oxygen monitoring described in design certification application Part 2, Tier 2, Section 6.2.5; information demonstrating that post-accident leakage from these systems does not result in the total main control room dose exceeding the dose criteria for the surrogate event with significant core damage, which may include use of design features compliant with 10 CFR 50.34(f)(2)(vii), as appropriate; and information demonstrating that post-accident leakage from these systems does not result in the total dose for the surrogate event with significant core damage exceeding the offsite dose criteria, as required by 10 CFR 52.47(a)(2)(iv); and

i. Information demonstrating that the criteria of 10 CFR part 20 and the requirements of 10 CFR part 50, appendix A, General Design Criterion (GDC) 4 and GDC 31 are met with respect to the structural and leakage integrity of the steam generator tubes that might be compromised by effects from density wave oscillations in the secondary fluid system, including the method of analysis to predict the thermal-hydraulic conditions of the steam generator secondary fluid system and resulting loads, stresses, and deformations from density wave oscillations and reverse flow. This information must be consistent with the other design information regarding steam generator integrity contained in design certification application Part 2, Tier 2, Sections 3.9.2 and 5.4.1.

3. Include, in the plant-specific DCD, the sensitive, unclassified, non-safeguards information (including proprietary information and security-related information) and safeguards information referenced in the NuScale generic DCD.

4. Include, as part of its application, a demonstration that an entity other than NuScale Power, LLC, is qualified to supply the NuScale generic DCD, unless NuScale Power, LLC, supplies the design for the applicant's use.

B. The Commission reserves the right to determine in what manner this appendix may be referenced by an applicant for a construction permit or operating license under 10 CFR part 50.

V. APPLICABLE REGULATIONS

A. Except as indicated in paragraph B of this section, the regulations that apply to NuScale are in 10 CFR parts 20, 50, 52, 73, and 100, codified as of **[DATE 120 DAYS AFTER DATE OF PUBLICATION OF FINAL RULE IN THE *FEDERAL REGISTER*]**, that are applicable and technically relevant, as described in the final safety evaluation report.

B. The NuScale design is exempt from portions of the following regulations:

2. Paragraph (m) of 10 CFR 50.54 – Minimum Staffing – codified as of **[DATE 120 DAYS AFTER DATE OF PUBLICATION OF FINAL RULE IN THE *FEDERAL REGISTER*]**. In lieu of these requirements, a licensee that references this appendix must comply with the following:

a. A senior operator licensed pursuant to part 55 of this chapter shall be present at the facility or readily available on call at all times during its operation, and shall be present at the facility during initial startup and approach to power, recovery from an

Commented [NT3]: Please list all exemptions listed in Section IV.G for clarity, and consistently revise the numbering.

Commented [NT4]: Please follow the practice of appendices A through D of part 52 to provide more useful information to the reader by including the specific topic in lieu of the title of the section of the regulation.

Deleted: 1. Paragraph (f)(2)(iv) of 10 CFR 50.34 – Contents of Applications: Technical Information – codified as of **[DATE 120 DAYS AFTER DATE OF PUBLICATION OF FINAL RULE IN THE *FEDERAL REGISTER*]**.¶

Deleted: Conditions of licenses

unplanned or unscheduled shutdown or significant reduction in power, and refueling, or as otherwise prescribed in the facility license.

b. Licensees shall meet the following requirements:

i. Each licensee shall meet the minimum licensed operator staffing requirements in the following table:

Table 1: Minimum Requirements Per Shift for On-Site Staffing of NuScale Power Plants by Operators and Senior Operators Licensed Under 10 CFR Part 55

Number of units operating (a nuclear power unit is considered to be operating when it is in MODE 1, 2, or 3 as defined by the unit's technical specifications)	Position	One to twelve units One control room
None	Senior operator	1
	Operator	2
One to twelve	Senior operator	3
	Operator	3

Source: Design Certification Application, Part 7, Section 6.1.3, "Requested Action."

ii. Each facility licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall plant operation at all times there is fuel in any unit. At all times any module is fueled, regardless of Mode, there must be a licensed operator or senior operator in the control room.

iii. When a nuclear power unit is in MODE 1, 2, or 3, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, a second person who is either a licensed operator or licensed senior operator shall be present at the controls at all times. A third person who is either a licensed operator or licensed senior operator shall be in the control room envelope at all times.

iv. Each licensee shall have present, during alteration or movement of the core of a nuclear power unit (including fuel loading, fuel transfer, or movement of a module that contains fuel), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person.

VI. Issue Resolution

A. The Commission has determined that the structures, systems, and components and design features of NuScale comply with the provisions of the Atomic Energy Act of 1954, as amended, and the applicable regulations identified in Section V of this appendix; and therefore, provide adequate protection to the health and safety of the public. A conclusion that a matter is resolved includes the finding that additional or alternative structures, systems, and components, design features, design criteria, testing, analyses, acceptance criteria, or justifications are not necessary for NuScale.

B. The Commission considers the following matters resolved within the meaning of § 52.63(a)(5) in subsequent proceedings for issuance of a COL, amendment of a COL, or renewal of a COL, proceedings held under § 52.103, and enforcement proceedings involving plants referencing this appendix:

1. All nuclear safety issues associated with the information in the final safety evaluation report, Tier 1, Tier 2, and the rulemaking record for certification of the NuScale design, with the exception of the following:

a. generic TS and other operational requirements;

b. the adequacy of the design of the shield wall between the NuScale power module and the reactor building steam gallery to limit potential radiological doses consistent with the radiation zones specified in design certification application Part 2, Tier 2, Chapter 12, Figure 12.3-1, “Reactor Building Radiation Zone Map”;

c. the adequacy of the design of the systems used for post-accident hydrogen and oxygen monitoring described in design certification application Part 2, Tier 2, Section 6.2.5 to meet the requirements of 10 CFR 50.34(f)(2)(vii), 10 CFR 50.34(f)(2)(xxviii), and 10 CFR 52.47(a)(2)(iv), with respect to radiological releases caused by leakage from these systems under accident conditions; and

d. the ability of the steam generator tubes to maintain structural and leakage integrity during density wave oscillations in the secondary fluid system, including the method of analysis to predict the thermal-hydraulic conditions of the steam generator secondary fluid system and resulting loads, stresses, and deformations from density wave oscillations and reverse flow, consistent with the other design information regarding steam generator integrity described in DCA Part 2, Tier 2, Sections 3.9.1, 3.9.2, 5.4.1, and 15.6.3, and in accordance with 10 CFR part 50, GDC 4, 10, and 31;

2. All nuclear safety and safeguards issues associated with the referenced information in the non-public documents in Tables 1.6-1 and 1.6-2 of Tier 2 of the DCD, which contain sensitive unclassified non-safeguards information (including proprietary information and security-related information) and safeguards information and which, in context, are intended as requirements in the generic DCD for the NuScale design;

3. All generic changes to the DCD under and in compliance with the change processes in paragraphs VIII.A.1 and VIII.B.1 of this appendix;

4. All exemptions from the DCD under and in compliance with the change processes in paragraphs VIII.A.4 and VIII.B.4 of this appendix, but only for that plant;

5. All departures from the DCD that are approved by license amendment, but only for that plant;

6. Except as provided in paragraph VIII.B.5.g of this appendix, all departures from Tier 2 under and in compliance with the change processes in paragraph VIII.B.5 of this appendix that do not require prior NRC approval, but only for that plant; and

7. All environmental issues concerning severe accident mitigation design alternatives associated with the information in the NRC's environmental assessment for NuScale (ADAMS Accession No. ML19303C179) and DCD Part 3, "Applicant's Environmental Report - Standard Design Certification," Revision 5, dated July 2020 (ADAMS Accession No. ML20224A512), for plants referencing this appendix whose site characteristics fall within those site parameters specified in the NuScale environmental report.

C. The Commission does not consider operational requirements for an applicant or licensee who references this appendix to be matters resolved within the meaning of § 52.63(a)(5). The Commission reserves the right to require operational requirements for an applicant or licensee who references this appendix by rule, regulation, order, or license condition.

D. Except under the change processes in Section VIII of this appendix, the Commission may not require an applicant or licensee who references this appendix to:

1. Modify structures, systems, and components or design features as described in the generic DCD;

2. Provide additional or alternative structures, systems, and components or design features not discussed in the generic DCD; or

3. Provide additional or alternative design criteria, testing, analyses, acceptance criteria, or justification for structures, systems, and components or design features discussed in the generic DCD.

E. The NRC will specify, at an appropriate time, the procedures to be used by an interested person who wishes to review portions of the design certification or references containing safeguards information or sensitive unclassified non-safeguards information (including proprietary information, such as trade secrets and commercial or financial information obtained from a person that are privileged or confidential (10 CFR 2.390 and

10 CFR part 9), and security-related information), for the purpose of participating in the hearing required by § 52.85, the hearing provided under § 52.103, or in any other proceeding relating to this appendix, in which interested persons have a right to request an adjudicatory hearing.

VII. Duration of this Appendix

This appendix may be referenced for a period of 15 years from **[INSERT DATE 120 DAYS AFTER DATE OF PUBLICATION IN THE *FEDERAL REGISTER*]**, except as provided for in §§ 52.55(b) and 52.57(b). This appendix remains valid for an applicant or licensee who references this appendix until the application is withdrawn or the license expires, including any period of extended operation under a renewed license.

VIII. Processes for Changes and Departures

A. Tier 1 Information

1. Generic changes to Tier 1 information are governed by the requirements in § 52.63(a)(1).

2. Generic changes to Tier 1 information are applicable to all applicants or licensees who reference this appendix, except those for which the change has been rendered technically irrelevant by action taken under paragraphs A.3 or A.4 of this section.

3. Departures from Tier 1 information that are required by the Commission through plant-specific orders are governed by the requirements in § 52.63(a)(4).

4. Exemptions from Tier 1 information are governed by the requirements in §§ 52.63(b)(1) and 52.98(f). The Commission will deny a request for an exemption from Tier 1, if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design.

B. Tier 2 Information

1. Generic changes to Tier 2 information are governed by the requirements in § 52.63(a)(1).

2. Generic changes to Tier 2 information are applicable to all applicants or licensees who reference this appendix, except those for which the change has been rendered technically irrelevant by action taken under paragraphs B.3, B.4, or B.5, of this section.

3. The Commission may not require new requirements on Tier 2 information by plant-specific order, while this appendix is in effect under § 52.55 or § 52.61, unless:

a. A modification is necessary to secure compliance with the Commission's regulations applicable and in effect at the time this appendix was approved, as set forth in Section V of this appendix, or to ensure adequate protection of the public health and safety or the common defense and security; and

b. Special circumstances as defined in 10 CFR 50.12(a) are present.

4. An applicant or licensee who references this appendix may request an exemption from Tier 2 information. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of 10 CFR 50.12(a). The Commission will deny a request for an exemption from Tier 2, if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design. The granting of an exemption to an applicant must be subject to litigation in the same manner as other issues material to the license hearing. The granting of an exemption to a licensee must be subject to an opportunity for a hearing in the same manner as license amendments.

5.a. An applicant or licensee who references this appendix may depart from Tier 2 information, without prior NRC approval, unless the proposed departure involves a change to or departure from Tier 1 information, or the TS, or requires a license amendment under paragraph B.5.b or B.5.c of this section. When evaluating the proposed departure, an applicant or licensee shall consider all matters described in the plant-specific DCD.

b. A proposed departure from Tier 2, other than one affecting resolution of a severe accident issue identified in the plant-specific DCD or one affecting information required by § 52.47(a)(28) to address aircraft impacts, requires a license amendment if it would:

(1) Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD;

(2) Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety and previously evaluated in the plant-specific DCD;

(3) Result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD;

(4) Result in more than a minimal increase in the consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the plant-specific DCD;

(5) Create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD;

(6) Create a possibility for a malfunction of a structure, system, or component important to safety with a different result than any evaluated previously in the plant-specific DCD;

(7) Result in a design-basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered; or

(8) Result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.

c. A proposed departure from Tier 2, affecting resolution of an ex-vessel severe accident design feature identified in the plant-specific DCD, requires a license amendment if:

(1) There is a substantial increase in the probability of an ex-vessel severe accident such that a particular ex-vessel severe accident previously reviewed and determined to be not credible could become credible; or

(2) There is a substantial increase in the consequences to the public of a particular ex-vessel severe accident previously reviewed.

d. A proposed departure from Tier 2 information required by § 52.47(a)(28) to address aircraft impacts shall consider the effect of the changed design feature or functional capability on the original aircraft impact assessment required by 10 CFR 50.150(a). The applicant or licensee shall describe, in the plant-specific DCD, how the modified design features and functional capabilities continue to meet the aircraft impact assessment requirements in 10 CFR 50.150(a)(1).

e. If a departure requires a license amendment under paragraph B.5.b or B.5.c of this section, it is governed by 10 CFR 50.90.

f. A departure from Tier 2 information that is made under paragraph B.5 of this section does not require an exemption from this appendix.

g. A party to an adjudicatory proceeding for either the issuance, amendment, or renewal of a license or for operation under § 52.103(a), who believes that an applicant or licensee who references this appendix has not complied with paragraph VIII.B.5 of this appendix when departing from Tier 2 information, may petition to admit into the proceeding such a contention. In addition to complying with the general requirements of 10 CFR 2.309, the petition must demonstrate that the departure does not comply with paragraph VIII.B.5 of this appendix. Further, the petition must demonstrate that the change stands on an asserted noncompliance with an ITAAC acceptance criterion in the case of a § 52.103 preoperational hearing, or that the change stands directly on the amendment request in the case of a hearing on a license amendment. Any other party may file a response. If, on the basis of the petition and any response, the presiding officer determines that a sufficient showing has been made, the presiding officer shall certify the matter directly to the Commission for determination of the admissibility of the contention. The Commission may admit such a contention if it determines the petition raises a genuine issue of material fact regarding compliance with paragraph VIII.B.5 of this appendix.

C. Operational Requirements

1. Changes to NuScale design certification generic TS and other operational requirements that were completely reviewed and approved in the design certification rule and do not require a change to a design feature in the generic DCD are governed by the requirements in 10 CFR 50.109. Changes that require a change to a design feature in the generic DCD are governed by the requirements in paragraphs A or B of this section.

2. Changes to NuScale design certification generic TS and other operational requirements are applicable to all applicants who reference this appendix, except those for which the change has been rendered technically irrelevant by action taken under paragraphs C.3 or C.4 of this section.

3. The Commission may require plant-specific departures on generic TS and other operational requirements that were completely reviewed and approved, provided a change to a design feature in the generic DCD is not required and special circumstances, as defined in 10 CFR 2.335 are present. The Commission may modify or supplement generic TS and other operational requirements that were not completely reviewed and approved or require additional TS and other operational requirements on a plant-specific basis, provided a change to a design feature in the generic DCD is not required.

4. An applicant who references this appendix may request an exemption from the generic TS or other operational requirements. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of § 52.7. The granting of an exemption must be subject to litigation in the same manner as other issues material to the license hearing.

5. A party to an adjudicatory proceeding for the issuance, amendment, or renewal of a license, or for operation under § 52.103(a), who believes that an operational requirement approved in the DCD or a TS derived from the generic TS must be changed, may petition to admit such a contention into the proceeding. The petition must comply with the general requirements of § 2.309 of this chapter and must either demonstrate why special circumstances as defined in § 2.335 of this chapter are present or demonstrate that the proposed change is necessary for compliance with the Commission's regulations in effect at the time this appendix was approved, as set forth in Section V of this appendix. Any other party may file a response to the petition. If, on the basis of the petition and any response, the presiding officer determines that a sufficient showing has been made, the presiding officer shall certify the matter directly to the Commission for determination of the admissibility of the contention. All other issues with respect to the plant-specific TS or other operational requirements are subject to a hearing as part of the licensing proceeding.

6. After issuance of a license, the generic TS have no further effect on the plant-specific TS. Changes to the plant-specific TS will be treated as license amendments under 10 CFR 50.90.

IX. [RESERVED]

X. RECORDS AND REPORTING

A. Records

1. The applicant for this appendix shall maintain a copy of the generic DCD that includes all generic changes that are made to Tier 1 and Tier 2, and the generic TS and other operational requirements. The applicant shall maintain the sensitive unclassified non-safeguards information (including proprietary information and security-related information) and safeguards information referenced in the generic DCD for the period that this appendix may be referenced, as specified in Section VII of this appendix.

2. An applicant or licensee who references this appendix shall maintain the plant-specific DCD to accurately reflect both generic changes to the generic DCD and plant-specific departures made under Section VIII of this appendix throughout the period of application and for the term of the license (including any periods of renewal).

3. An applicant or licensee who references this appendix shall prepare and maintain written evaluations which provide the bases for the determinations required by Section VIII of this appendix. These evaluations must be retained throughout the period of application and for the term of the license (including any periods of renewal).

4.a. The applicant for NuScale shall maintain a copy of the aircraft impact assessment performed to comply with the requirements of 10 CFR 50.150(a) for the term of the certification (including any period of renewal).

b. An applicant or licensee who references this appendix shall maintain a copy of the aircraft impact assessment performed to comply with the requirements of 10 CFR 50.150(a) throughout the pendency of the application and for the term of the license (including any periods of renewal).

B. Reporting

1. An applicant or licensee who references this appendix shall submit a report to the NRC containing a brief description of any plant-specific departures from the DCD, including a summary of the evaluation of each departure. This report must be filed in accordance with the filing requirements applicable to reports in § 52.3.

2. An applicant or licensee who references this appendix shall submit updates to its plant-specific DCD, which reflect the generic changes to and plant-specific departures from the generic DCD made under Section VIII of this appendix. These updates shall be filed under the filing requirements applicable to final safety analysis report updates in 10 CFR 50.71(e) and 52.3.

3. The reports and updates required by paragraphs X.B.1 and X.B.2 of this appendix must be submitted as follows:

a. On the date that an application for a license referencing this appendix is submitted, the application must include the report and any updates to the generic DCD.

b. During the interval from the date of application for a license to the date the Commission makes its finding required by § 52.103(g), the report must be submitted semiannually. Updates to the plant-specific DCD must be submitted annually and may be submitted along with amendments to the application.

c. After the Commission makes the finding required by § 52.103(g), the reports and updates to the plant-specific DCD must be submitted, along with updates to the site-specific portion of the final safety analysis report for the facility, at the intervals required by 10 CFR 50.59(d)(2) and 50.71(e)(4), respectively, or at shorter intervals as specified in the license.

Dated at Rockville, Maryland, this xxth day of Xxxxx, 2020.

For the Nuclear Regulatory Commission.

Annette Vietti-Cook,
Secretary of the Commission.

CTH edits

ENVIRONMENTAL ASSESSMENT BY THE

U.S. NUCLEAR REGULATORY COMMISSION

RELATING TO THE CERTIFICATION OF THE

NUSCALE STANDARD DESIGN

DOCKET NO. 52-048

TABLE OF CONTENTS

UNITED STATES NUCLEAR REGULATORY COMMISSION

ENVIRONMENTAL ASSESSMENT AND FINDING OF

NO SIGNIFICANT IMPACT

RELATING TO THE CERTIFICATION OF THE

NUSCALE STANDARD DESIGN

DOCKET NO. 52-048

The U.S. Nuclear Regulatory Commission (NRC) is issuing a design certification for the NuScale standard design (NuScale) in response to an application submitted in December 2016 by NuScale Power, LLC (NuScale Power). The NRC adopts design certification rules as appendices to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.”

The NRC has performed the following environmental assessment of the environmental impacts of the new rule and has documented its finding of no significant impact in accordance with the requirements of 10 CFR 51.21, “Criteria for and identification of licensing and regulatory actions requiring environmental assessments”; 10 CFR 51.31, “Determinations based on environmental assessment”; and the National Environmental Policy Act of 1969, as amended (NEPA). This environmental assessment addresses the severe accident mitigation design alternatives (SAMDAs) that the NRC has considered for NuScale. This environmental assessment does not address the site-specific environmental impacts of constructing and operating any facility that references the NuScale design certification rule at a particular site; the

NRC will evaluate those impacts as part of its review of any application(s) for the siting, construction, or operation of such a facility.

The NRC has determined that issuing this design certification does not constitute a major Federal action significantly affecting the quality of the human environment. This finding is based on the generic finding made in 10 CFR 51.32(b)(1) that there is no significant environmental impact associated with the certification of a standard design under 10 CFR Part 52, Subpart B, “Standard Design Certifications.” The action does not authorize the siting, construction, or operation of a facility using NuScale. Rather, it codifies the NuScale standard design in a rule that could be referenced in a future licensing application. Furthermore, because the certification is a rule rather than a physical action, it does not involve the commitment of any resources that have alternative uses. The 10 CFR 51.32(b)(1) generic finding of no significant impact is, essentially, the legal equivalent of a categorical exclusion (72 FR 49427; August 28, 2007). Therefore, the NRC has not prepared an environmental impact statement for the action.

Under 10 CFR 51.30(d), an environmental assessment for a standard design certification must identify the proposed action and is limited to consideration of the costs and benefits of SAMDAs and the bases for not incorporating SAMDAs in the design certification. As discussed in Section 4.0 of this environmental assessment, the NRC also reviewed NuScale Power’s assessment of SAMDAs that generically apply to the NuScale design. The NRC finds that NuScale Power’s assessment, as presented in its Environmental Report (ER), considered a reasonable set of SAMDAs at the design certification stage and adequately demonstrated that none of the evaluated SAMDAs would provide cost beneficial risk improvements. This finding is applicable only to the SAMDAs considered at the time of the certification and may be referenced in a future licensing action. Under Appendix G, “Design Certification Rule for NuScale,” to 10 CFR Part 52, a plant referencing the NuScale design certification rule should be sited at a location with site characteristics that are encompassed by the site parameters for the design certification reference plant site in the NuScale ER submitted as design certification application Part 3, “Applicant’s Environmental Report—Standard Design Certification,” Revision 5, issued July 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20224A512), and in the supporting documents.

During the review process, NuScale Power identified 199 potential SAMDAs in the NuScale ER, Table A-1, “Screening of Proposed SAMDAs.” NuScale Power evaluated the SAMDAs and binned them into categories. Several of the potential SAMDAs were determined to be not applicable to the NuScale design. Others were already incorporated into the design. Thirty-seven SAMDAs were related to a procedural or surveillance action, to a multiple plant site, or to design elements to be finalized as part of a future licensing action. Therefore, NuScale Power determined that these SAMDAs were not applicable at the design certification stage. NuScale Power evaluated the remaining SAMDAs to determine if any provided cost beneficial risk improvements. NuScale Power determined that none of the evaluated SAMDAs were cost beneficial.

Thirty-four of the thirty-seven SAMDAs identified by NuScale Power to be finalized as part of a future licensing action are related to a procedural or surveillance action. For the remaining three SAMDAs identified by NuScale Power, two of the SAMDA candidates (SAMDAs 17 and 85) are for multi-unit, or multiple plant, sites. Namely, a multi-unit site has at least two complete plants, each with 12 NuScale power modules (NPMs) in their respective reactor buildings. Since the NuScale design certification is for a single plant site, a future licensing applicant should reevaluate these two SAMDAs if the site would have multiple plants. The third SAMDA candidate not evaluated at the design certification stage (SAMDA 197) involves the reactor building crane (RBC). The detailed design of the RBC will be provided as part of a future licensing action and is therefore not considered part of the design certification.

Until the design of the RBC is completed, any specific RBC SAMDAs cannot be determined to appropriately address reducing the risk from a dropped module, nor can an associated cost for any such specific RBC SAMDA be calculated. Based on the above, the staff determined that the set of 51 SAMDA candidates evaluated in the NuScale ER are appropriate for further review in this design certification. A future licensing applicant's NuScale ER will need to evaluate the 37 SAMDAs that NuScale Power identified as not required for design certification.

ENVIRONMENTAL ASSESSMENT

1.0 Identification of the Proposed Action

The proposed action is to certify NuScale in Appendix G to 10 CFR Part 52.

2.0 Need for the Proposed Action

The need for the proposed action is to allow an applicant to reference the NuScale design certification rule as part of a future licensing application. Specifically, the NuScale design certification could be referenced in a combined license (COL) application under 10 CFR Part 52, or it may allow for a construction permit application under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." Those portions of the NuScale design included in the scope of the design certification rulemaking are not subject to further safety review or approval in a future licensing proceeding. In addition, the NuScale design certification rule resolves the SAMDAs evaluated at the design certification stage for any future COL applications that reference the NuScale design certification rule and fall within the associated site parameters for the design certification reference plant site in the NuScale ER.

3.0 Environmental Impact of the Proposed Action

As stated in 10 CFR 51.32(b)(1), the NRC has determined that there is no significant environmental impact associated with the issuance of a design certification. The design certification codifies the NRC's approval of the NuScale standard design through its final safety evaluation report on the design issued during rulemaking (ADAMS Accession No. ML20023A318). Furthermore, because the certification of the design does not authorize any action, it would not involve the commitment of any resources that have alternative uses.

As described in Section 4.0 of this environmental assessment, the NRC reviewed various alternative design features for preventing and mitigating severe accidents. NEPA requires the consideration of such alternatives to show that the design certification rule is the appropriate course of action. The NRC's regulations at 10 CFR 51.55(a) ensure that the design referenced in rulemaking does not exclude any cost-beneficial design changes related to the prevention and mitigation of severe accidents.

Through its own independent analysis, the NRC concludes that NuScale Power adequately considered an appropriate set of SAMDAs at the design certification stage and that none met the criteria to be considered cost beneficial. Before conducting the SAMDA evaluation, NuScale Power had already incorporated certain severe accident prevention and mitigation design features into the NuScale design based on probabilistic risk assessment (PRA) results. (See design certification application Part 2, Tier 2, Section 19.2.2, "Severe Accident Prevention," and Section 19.2.3, "Severe Accident Mitigation" (ADAMS Accession No. ML20224A508)). NuScale Power made no design changes as a result of considering SAMDAs.

Finally, the design certification rule does not authorize the siting, construction, or operation of a nuclear power plant. An applicant for a construction permit, early site permit, COL, or operating license that references the NuScale design certification rule will be required to address the environmental impacts of construction and operation for its specific site. As part of this evaluation, such an applicant would evaluate the 37 SAMDAs that NuScale Power identified as not required for design certification. These NuScale SAMDAs cannot be evaluated at the design certification stage, given the procedural, training, siting, or design details that would be needed to reliably assess the costs of each improvement. The NRC will evaluate the environmental impacts for that particular site and issue an environmental impact statement in accordance with 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.” The SAMDA analysis that has been completed as part of this environmental assessment can be incorporated by reference into an environmental impact statement related to an application for siting, construction, or operation of a nuclear plant that references the NuScale design certification and falls within the associated site parameters for the design certification reference plant site in the NuScale ER.

4.0 Severe Accident Mitigation Design Alternatives

This section provides a summary of the NRC’s review of the NuScale ER and the related SAMDAs, as provided in the NuScale ER and supporting documents. The staff provides the specific details of the NRC’s evaluation, summarized in this environmental assessment, in a technical analysis report, “Staff Technical Analysis in Support of the NuScale Design Certification Environmental Assessment” (ADAMS Accession No. ML19302E819).

Consistent with the Commission’s objectives of standardization and early resolution of design issues, the staff is evaluating the SAMDAs identified as within the scope of the NuScale design certification. In “Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants” (50 FR 32138; August 8, 1985), the Commission defined the term “severe accident” as an event that is beyond the substantial coverage of design-basis events,[1] including events in which substantial damage is done to the reactor core (whether or not there are serious offsite consequences).

[1] Design-basis events are events analyzed in accordance with NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” and documented in the NuScale design certification application, Part 2, Tier 2, Chapter 15, “Transient and Accident Analyses.”

Alternative design features for severe accidents in a design certification must be evaluated in two ways:

- 10 CFR 52.47(a)(27) requires a design certification applicant to describe the design-specific PRA and its results.
- 10 CFR 51.30(d) requires the consideration of SAMDAs in an environmental assessment for a design certification.

Although these requirements are not directly related, they share common purposes, which are to consider alternatives to the proposed design, to evaluate whether potential alternative improvements in the plant design might significantly enhance safety performance during severe accidents, and to prevent the foreclosure of reasonable alternatives.

The PRA required for a design certification application comprises two major areas of analysis: (1) the identification of sequences of events that could lead to core damage and the estimation of their frequencies of occurrence (the Level 1 PRA analysis) and (2) the evaluation of the potential response of the containment to these sequences, with emphasis on the possible modes of containment failure and the corresponding radionuclide source terms (the Level 2 PRA analysis).

NuScale Power performed a PRA, described in the NuScale design certification application Part 2, Tier 2, Chapter 19, to achieve the following objectives:

- Identify the dominant severe accident sequences that account for most of the core damage frequency and associated source terms for the design.
- Modify the design, on the basis of PRA insights, to prevent severe accidents or mitigate their consequences and thereby reduce the risk of such accidents.
- Provide a qualitative basis for concluding that all reasonable steps have been taken to reduce the chances of severe accidents occurring and to mitigate the consequences.

Deleted: 79

Deleted: 46

Deleted: plant

The NuScale PRA evaluates the risk of core damage and release of radioactive material associated with both internal and external events that can occur during plant operation at power or while shut down. The NuScale Level 1 and Level 2 PRA models quantified seven risk categories:

(1) internal events

(2) low-power shutdown

(3) internal flooding

(4) internal fires

(5) external floods

(6) high winds

(7) multimodule

NuScale Power used insights from the NuScale PRA by applying a SAMDA analysis approach as described in Nuclear Energy Institute (NEI) 05-01, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document," Revision A (NEI 05-1A), issued November 2005 (ADAMS Accession No. ML060530203). This guidance is an acceptable methodology to the NRC for the assessment of SAMAs[2] for license renewals under 10 CFR 51.53(c)(3)(ii)(L) (see Final License Renewal Interim Staff Guidance LR-ISG-2006-03, ADAMS Accession No. ML071640133) but has not been endorsed or accepted for the assessment of new reactor SAMDAs under 10 CFR 51.55(a). However, the staff recognizes that there is useful information and guidance contained in NEI 05-01A for application to the SAMDA assessment for a design certification and it has been applied in other design certifications (e.g., the APR1400 design certification's ER, ADAMS Accession No. ML15006A038). First, NEI 05-01A applies the cost formulas from NUREG/BR-0184, Regulatory Analysis Technical Evaluation Handbook, for assessing the maximum benefit and includes guidance on applying cost formulas to a SAMA

[2] SAMAs are a subset of SAMDAs, which are attributes of design alternatives, procedural modifications, and training activities for the mitigation of severe accidents.

assessment. Second, NEI 05-01A provides a standard list of SAMAs for pressurized light water reactors (PWRs) (NuScale is a PWR) to aid in the identification of candidate SAMDAs. Finally, NEI 05-01A provides a process for screening and assessing whether a SAMDA is potentially cost beneficial. Therefore, the staff accepts NuScale's application of the NEI 05-01A guidance for this SAMDA assessment.

NuScale Power analyzed the various combinations of events leading to radiological releases from the NPM.   NuScale Power then grouped these combinations of events into eight release categories (RCs) specifically to be applied in the SAMDA analysis based on the initiating events and mitigation system availabilities. RC 1 through 7 are generally based on internal initiating events such as pipe breaks, steam generator tube failures, spurious emergency core cooling system actuation, and general transients. RC 8 is associated with a dropped NPM during refueling operations. A potential cause of a dropped NPM could be failures associated with the RBC.

The NuScale design certification is for a single plant located on a site that falls within the associated site parameters. In its application, NuScale Power stated that certain siting and calculated offsite consequences were based on data and parameters associated with the Surry Power Station (Surry) site. NuScale Power asserts this site to be a reasonably representative site for the purposes of the SAMDA analysis. The Surry site has been applied for this same purpose in other severe accident analyses, with the most recent analysis being the State-of-the-Art Reactor Consequence Analysis (SOARCA). As part of the SOARCA documentation, information was provided in a supporting document with the necessary consequence code input information developed by the NRC staff for NuScale Power to use in the consequence analysis in their SAMDA assessment. Because the detailed information about the Surry site makes it a good source of information for a SAMDA assessment, and there are no reasons that the Surry site would otherwise be inappropriate, the staff determined that the Surry site is an acceptable representative site for the purposes of the applicant’s SAMDA analysis.

Based on the staff’s review of NuScale Power’s SAMDA evaluation, the staff determined that NuScale Power adequately identified SAMDAs for RCs 1 through 7 that could potentially reduce risk, and that these SAMDAs would not be cost beneficial based on applying the representative site parameters in the NuScale ER. A SAMDA for RC 8 (SAMDA 199) is associated with design elements (i.e., the RBC) to be finalized at a later stage in the design process and, therefore, not applicable at the design certification stage. Additionally, NuScale has two other SAMDA candidates, namely SAMDAs 17 (create a cross-tie for diesel fuel oil) and 85 (provide cross-unit connection of uninterruptible compressed air supply), that address multiple plant site risks. A future applicant referencing the NuScale design certification rule and proposing a site with multiple plants would need to reevaluate these two SAMDA candidates.

The NRC has determined that the generic evaluation of SAMDAs for NuScale is both practical, and warranted, for two reasons. First, all plants referencing the NuScale design certification rule will be constructed according to the same design. Second, the site parameters in the NuScale ER and supporting documents establish the consequences for a reasonable set of SAMDAs for the NuScale design certification. The low residual risk posed by the NuScale design certification, and the limited potential for further risk reduction, provide high confidence that additional cost beneficial SAMDAs would not be found for sites with characteristics that fit within the site parameter envelope. If an actual characteristic for a particular site does not fall within the site parameters, then SAMDAs that could be affected by the value of the site characteristic must be reevaluated in the site-specific ER and the environmental impact statement prepared in connection with the application. If the actual characteristics of a proposed site fall within the site parameters stated in the ER, then the SAMDA analysis can be incorporated by reference in the site-specific environmental impact statement and SAMDAs need not be reevaluated in the environmental impact statement.

4.1. Potential Design Improvements Identified by NuScale Power

In the NuScale ER and the supporting documents, NuScale Power identified 199 candidate design alternatives, or design improvements, based on a review of the standard list of design alternatives provided in Table 14 of NEI-05-01, Revision A, and several other license renewal ERs. NuScale Power eliminated certain candidate design alternatives from further consideration on the following bases:

- They were already implemented in NuScale.
- They were not applicable to NuScale or to the design certification.
- They had excessive implementation costs.
- They were of very low benefit.
- They were combined into a comprehensive SAMDA candidate.
- They were not required for design certification.

NuScale Power had already incorporated 18 candidate design alternatives, such as the following:

- installing a gas turbine generator
- providing additional direct current battery capacity
- creating a reactor cavity flooding system

NuScale Power applied a SAMDA screening process based on NEI-05-01, Revision A, and presented its assessment in NuScale ER Section 6. In summary, NuScale Power performed three screening steps to assess whether a SAMDA candidate should be considered for a cost benefit analysis.

As described in Section 4.3.1 of this environmental assessment, if the expected implementation costs for a SAMDA candidate would exceed the calculated maximum benefit, resulting in a negative net present value (NPV), NuScale Power did not consider the SAMDA further. NuScale Power began the screening process with 199 SAMDAs. This screening process eliminated one potential design alternative that was identified as being unfeasible due to excessive implementation costs and that provided negligible benefit. The applicant identified another 45 SAMDA candidates as not applicable to the design certification stage of plant development (such as procedural processes, training, or design features not applicable at the design certification stage). The applicant combined similar SAMDA candidates to develop a more comprehensive SAMDA candidates, eliminating 13. It determined 34 potential design alternatives to be of very low benefit. NuScale Power retained the remaining 51 SAMDAs for further assessment in the cost-benefit analysis.

4.2. NRC Evaluation of Potential Design Improvements

The NRC reviewed NuScale Power's SAMDA candidate screening and selection process and determined that the methods applied, and the implementation of the methods, are appropriate. The NRC also determined that the set of SAMDA candidates evaluated by NuScale Power addressed the major contributors to core damage. For those SAMDA candidates that continued to SAMDA screening, NuScale Power applied a systematic and comprehensive process for identifying potential plant improvements for NuScale, and the set of potential plant improvements identified by NuScale Power is reasonably comprehensive and, therefore, acceptable for the NRC's evaluation. As discussed previously, NuScale Power's SAMDA candidate search included reviewing insights from the design-specific PRA study as well as assessing SAMAs based on accepted industry guidance (see Section 4.3.1 of this environmental assessment).

4.3. Cost Impacts of Candidate Severe Accident Mitigation Design Alternatives

4.3.1. NuScale Power Evaluation

In performing the cost-benefit analysis of the SAMDAs considered, NuScale estimated the cost of implementing the enhancement (cost of enhancement or COE) associated with potential events based on available information related to similar events and components of other nuclear power plant designs. The COE values for the NuScale SAMDAs were derived from the compilation of information from the SAMA analyses performed for the license renewal applications of the presently operating nuclear power plants, as documented in the licensees' renewal ERs and in the final supplemental environmental impact statements under NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants." From these sources of information, NuScale Power identified a minimum COE value of approximately $100,000.

4.3.2. NRC Evaluation

On the basis of the analyses performed by NuScale Power, the NRC found that the estimated potential costs for the SAMDAs (i.e., COE values) evaluated by NuScale at the design certification stage are acceptable because the sources for the information and the cost estimates reasonably apply guidance based on prior NRC reviewed license renewal SAMAs. This approach facilitates the cost-benefit comparisons founded on a screening approach when assessing the averted costs using 7-percent and 3-percent discount rates. This approach is consistent with the guidance in Section 7.2 of NEI-05-01, Revision A.

4.4. Cost-Benefit Comparison

4.4.1. NuScale Power Evaluation

The methodology used by NuScale Power is based primarily on the NRC's guidance for performing cost-benefit analysis outlined in NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," issued January 1997 (ADAMS Accession No. ML050190193) as described in NEI 05-01A. The guidance involves determining the NPV for each SAMDA according to the following formula:

NPV = (APE + AOC + AOE + AOSC) – COE

Where:

NPV = net present value of current risk ($)

APE = present value of averted public exposure ($)

AOC = present value of averted offsite property damage costs ($)

AOE = present value of averted occupational exposure ($)

AOSC = present value of averted onsite costs ($)

COE = cost of any enhancement implemented to reduce risk ($)

If the NPV of a SAMDA is negative, the cost of implementing the SAMDA is larger than the benefit associated with the SAMDA, and the SAMDA is not cost beneficial. As noted below, the applicant screened out 51 candidate SAMDAs from further analyses for this reason. If the SAMDA's benefit exceeds the estimated cost, resulting in a positive NPV, the SAMDA is potentially cost beneficial.

To represent the maximum benefit that could be provided, the maximum benefit is calculated to be the sum of the four averted cost categories, as follows:

Maximum Benefit = APE + AOC + AOE + AOSC

Table 1 summarizes NuScale Power's and the NRC's estimates for each of the associated maximum benefit cost elements, applying the 7-percent discount rate. NuScale performed a sensitivity case for a single NPM using the 3-percent discount rate, with a result of approximately $341,000 total maximum benefit. The averted costs in Table 1 are based on the applicant's numerical PRA results and conservatively represent the potential benefit associated with eliminating all severe accidents. As noted in design certification application Part 2, Tier 2, Chapter 19, Section 19.1.9.1, COL Item 19.1-8 states that a COL applicant that references the NuScale design certification will confirm the validity of the key assumptions and data used in the design certification application PRA and modify, as necessary, for applicability to the as-built, as-operated PRA.

Table 1 Calculated Total Maximum Benefit for Severe Accident Impact

Risk Category	Single NPM		12 NPMs	
	NuScale[a]	NRC	NuScale[a]	NRC
APE	$58.6	$61.4	$196	$205
AOC	$0.755	$0.859	$8.68	$9.47
AOE	$1,340	$1,360	$4,130	$4,250
AOSC	$31,300	$31,300	$131,000	$131,000
Total Maximum Benefit	$32,700	$32,735	$136,000	$136,000

[a] From Tables 5-3 and 5-4 of the NuScale ER.

It is important to note that the monetary present value estimate for each risk attribute does not represent the expected reduction in risk resulting from a single averted accident. Rather, it is the present value of potential losses extending over the projected lifetime of the facility (in this case, 60 years, which assumes a 40-year license and at least one 20-year license renewal). Therefore, it reflects the expected averted annual costs resulting from eliminating all severe accidents and the effect of discounting these potential averted future costs that may occur at any time over the licensed life to present value.

The NRC issued NUREG/BR-0058, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission,” Revision 4, in September 2004 (ADAMS Accession No. ML042820192), to reflect the agency’s policy on discount rates. NUREG/BR-0058, Revision 4, states that two sets of estimates should be developed—one at 3-percent and one at 7-percent.

As previously discussed, the applicant carried 51 SAMDAs to the next screening phase for cost benefit consideration. For each of the SAMDAs applying the 7-percent discount rate, NuScale Power evaluated the NPV and concluded whether the enhancements were cost beneficial. NuScale Power performed this part of the evaluation for the NPV by subtracting the cost of incorporating the SAMDA into the design, namely the COE value, from the maximum benefit derived based on the conservative assumption that the implementation of any SAMDA would reduce the total plant risk to zero. Through its SAMDA analyses, NuScale Power determined that the SAMDAs it considered to be within the scope of the design certification afforded no potentially cost-beneficial enhancements at the 7-percent discount rate. In NuScale ER Section 6.4, Screening Sensitivity, NuScale Power analyzed several sensitivity cases and found that only a siting sensitivity case using the Peach Bottom site (also a site analyzed in the SOARCA study) and the 3-percent discount rate sensitivity cases have the possibility to result in a SAMDA candidate that is potentially cost beneficial. The applicant further stated in NuScale ER Section 6.4, that the SAMDA candidates associated with seismic risk (i.e., SAMDA 140, 187, and 188 associated with improvements to the seismic ruggedness of certain plant components) were potentially cost beneficial due to the higher seismic activity at the Peach Bottom site compared to the Surry site. However, the applicant stated that the use of a more detailed SAMDA cost estimate, as opposed to the assumed SAMDA implementation cost used for screening purposes, would likely show that these SAMDAs are not cost beneficial. Therefore, of the SAMDAs considered within the scope of the design certification, NuScale Power concluded that no design changes would provide a positive cost benefit if included in the NuScale design.

4.4.2. NRC Evaluation

The staff selected a subset of the NuScale sensitivity cases described in NuScale ER Section 5.8, Maximum Benefit Sensitivity Study, for the staff's independent confirmatory analysis. The staff's confirmatory calculational results for the Surry site and Peach Bottom site (Sensitivity Case 3) are very similar to NuScale's results. The NuScale sensitivity analysis demonstrates how insensitive the maximum benefit is to changes in the consequence analysis that affect the averted public exposure (APE), averted offsite property damage costs (AOC), and averted occupational exposure (AOE) values as shown by the results for Sensitivity Cases 1, 2, and 4 through 13. Based on the staff's confirmatory calculations and NuScale's sensitivity analysis, the staff sees no benefit from performing confirmatory analysis on all of the other sensitivity cases. The staff recognizes, based on the above offsite consequence/risk and maximum benefit sensitivity analyses, that the APE, AOC, and AOE only have a small contribution to the total maximum benefit.

As shown in Table 1, the NRC's confirmatory analyses were in general agreement with those of NuScale Power for the offsite public exposure (i.e., APE), onsite occupational dose (i.e., AOE) averted costs, and the onsite (i.e., AOSC) averted costs. The NRC evaluation resulted in higher values than NuScale Power's evaluation for offsite property damage cost (i.e., AOC).

The frequency weighted maximum benefits for RC 1 through RC 7 are very small for the risks from either a single NPM or all 12 NPMs. Thus, the staff finds it reasonable that the costs for SAMDA candidates that would reduce the frequency of RC 1 through RC 7 will be much greater than the maximum benefits of RC 1 through RC 7 (less than $100 for the risk from a single NPM and $1,200 for the risk from 12 NPMs). Therefore, none of the SAMDA candidates for reducing the risks from RC 1 through RC 7 would be potentially cost beneficial. As discussed below, the staff cannot reach a finding on the maximum benefit and cost benefits for potential enhancements related to RC 8 and the procedural, training, siting, or design details of the RBC.

The staff notes that SAMDAs 17 and 85 would also be assessed at the COL stage, if applicable, due to their multi-unit aspects, and a COL applicant would address the site-specific external events and natural phenomena. SAMDA 197 (automate the NPM transport process) and SAMDA 199 (improve testing and maintenance procedures for the RBC crane), related to RBC risk reductions, would also be addressed at the COL stage once the detailed design of the RBC is complete. Due to the SAMDA's sensitivity to site characteristics, the staff also expects that a COL applicant would assess the costs and benefits of SAMDA 197. The COL stage is when the site characteristics are fully developed, with complete documentation in the COL application that would support a full and complete assessment of all possible SAMDA candidates for the severe accident scenario with the most significant risks.

For this reason, the staff findings on the NuScale SAMDA analysis are limited to those SAMDAs that could reasonably be evaluated at the design certification stage. Therefore, the staff recommends that the Commission not consider the 37 SAMDAs that were not required for design certification to be environmental issues resolved within the meaning of 10 CFR 52.63(a)(5). Rather, multi-unit aspects (SAMDAs 17 and 85), and the RBC design (SAMDA 197), as well as the 34 procedural and training SAMDAs, would need to be assessed when a specific site is proposed for constructing and operating a NuScale power plant.

The NRC's confirmatory analysis reached the same conclusion as NuScale Power that there were no cost beneficial design alternatives for a single NPM for SAMDAs addressing RC 1 through RC 7. Based on the NRC's review of the methodology and associated analysis, NuScale Power's assessment of SAMDAs that could be evaluated at the design certification stage adequately incorporated the cost benefit analysis. On May 11, 2020, NuScale and NRC discussed the potential need to submit a revision to the ER due to recent design changes that prevent postulated boron redistribution scenarios. The discussion was conducted as part of the NRC audit of these design changes. As a result of the meeting, NuScale submitted a letter (LO-0720-70844 – NuScale Power, LLC Submittal of Environmental Report: Revision Status (ADAMS Accession No. ML20192A326)) to document its evaluation of the potential need to revise the ER to reflect those design changes. The boron redistribution design changes have been evaluated for their effect on the ER. NRC verified NuScale's conclusion that the effect on the ER is limited to editorial changes associated with event sequence numbering and event descriptions for consistency with the Final Safety Analysis Report (FSAR) Chapter 19 changes that were included in design certification application Revision 4.1 (ADAMS Accession

No. ML20197A413). A COL applicant that references the NuScale design certification will need to provide a revised ER with the noted editorial changes.

4.5. Conclusions on Severe Accident Mitigation Design Alternatives

The NRC reviewed NuScale Power's SAMDA analysis and concludes that the methods used and the implementation of the methods are appropriate. Based on the staff's independent confirmatory evaluation as described in the previous sections, the staff finds the results of the NuScale Power risk and maximum benefit analyses for the single NPM with respect to RC 1 through RC 7 to be reasonable, with no potentially cost beneficial SAMDAs as assessed using the Surry site parameters for offsite consequence evaluation from the NuScale ER. However, the staff cannot reach a finding on the maximum benefit and cost benefits for potential enhancements related to: RC 8 and the procedural, training, siting, or design details of the RBC; SAMDAs associated with multi-unit aspects; and procedural and training-related SAMDAs. Therefore, a COL applicant that references the NuScale design certification rule will need to reevaluate the maximum benefit and assessment of SAMDA candidates addressing procedural, training, siting, or other specific aspects of the detailed design.

The NRC based its independent evaluation on a reasonable treatment of costs, benefits, and sensitivities, as previously described in Section 4.4.2 of this environmental assessment. Based on the NRC's review of NuScale Power's evaluation, including NuScale Power's response to requests for additional information, the NRC concludes that NuScale Power has adequately identified areas for which risk could potentially be reduced in a cost beneficial manner and has adequately assessed whether the implementation of the identified potential SAMDAs or candidate design alternatives would be cost beneficial for the given site parameters. This staff conclusion is based upon the site data and parameters that fall with those specified in the NuScale ER.

Accordingly, the staff findings on the NuScale Power SAMDA analysis are limited to those SAMDAs evaluated as part of this design certification and do not include SAMDAs that address procedures, training, or RBC design details. These SAMDAs are not final and are considered outside the scope of the NuScale certified design. Therefore, the staff recommends that the Commission not consider environmental issues resolved within the meaning of 10 CFR 52.63(a)(5) that concern the 37 SAMDAs associated with procedures, training, or the future design development of the RBC. In particular, a COL applicant that references the NuScale design certification rule will need to provide further SAMDA analyses once the design of the RBC is finalized and more information about procedures and training are available. SAMDAs on multi-unit aspects (SAMDAs 17 and 85) will need to be assessed if a specific multi-unit site is proposed in a future licensing action referencing the NuScale certified design.

5.0 Finding of No Significant Impact

On the basis of this environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC is not required to prepare an environmental impact statement for the proposed action.

The design certification rule and the documents referenced in the Statements of Consideration for the final rule contain further details with respect to the proposed action. Publicly available records will be accessible online in the ADAMS Public Documents collection at https://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents in ADAMS should contact the NRC's Public Document Room reference staff at 1-800-397-4209, at 301-415-4737, or by e-mail to PDR.Resource@nrc.gov.