[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: Final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is amending 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.

DATES: This final rule is effective on **February 21, 2023.** The incorporation by reference of certain publications listed in the rule is approved by the Director of the Federal Register as of **February 21, 2023.**

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

- **Federal Rulemaking Website:** 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; email: Dawn.Forder@nrc.gov. For technical questions, contact the individuals listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **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, 301-415-4737, or by email to PDR.Resource@nrc.gov. For the convenience of the reader, instructions about obtaining materials referenced in this document are provided in the "Availability of Documents" section.

- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room P1 B35, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852. To make an appointment to visit the PDR, please send an email to PDR.Resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8:00 a.m. and 4:00 p.m. (ET), Monday through Friday, except Federal holidays.

- **Technical Library**: 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 email at Library.Resource@nrc.gov between 8:00 a.m. and 4:00 p.m. (ET), Monday through Friday, except Federal holidays.

FOR FURTHER INFORMATION CONTACT: Yanely Malave, Office of Nuclear Material Safety and Safeguards, telephone: 301-415-1519, email: Yanely.Malave@nrc.gov, and

Carolyn Lauron, Office of Nuclear Reactor Regulation, telephone: 301-415-2736, email: Carolyn.Lauron@nrc.gov. Both are staff of the U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

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I. 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). 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 the NuScale power modules are partially submerged in a common safety-related pool, which is also the ultimate heat sink for up to 12 power modules. The pool portion of the reactor building is located below grade. The design utilizes several first-of-a-kind approaches for accomplishing key safety functions, resulting in no need for 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 need for control room staffing while providing safe operation of the plant during normal and post-accident operation.

II. Opportunities for Public Participation

The proposed rule and environmental assessment were published in the *Federal Register* on July 1, 2021, for a 60-day public comment period (86 FR 34999). The public comment period was scheduled to close on August 30, 2021. The NRC subsequently extended the comment period by 45 days (86 FR 47251; August 24, 2021), providing a total comment period of 105 days. The public comment period closed on October 14, 2021. The public comments informed the development of this final rule.

III. Regulatory and Policy Issues

A. Exemptions for future applicants referencing NuScale

1. 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, NuScale Power proposed that 6 licensed operators would operate up to 12 power modules from a single control room. The staffing proposal would meet the requirements of § 50.54(k) but 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), which requires

a licensed operator at the controls for each fueled unit. Absent alternative staffing requirements, future applicants referencing the NuScale design would need to request an exemption.

In DCA, Part 7, Section 6, NuScale requested that the NRC approve design-specific control room staffing requirements in lieu of the requirements in § 50.54(m). In the DCA Part 7, Section 6.2, “Justification for Rulemaking,” NuScale Power provided a technical basis for its proposed alternative control room staffing requirements. NuScale Power’s proposed approach is consistent with SECY-11-0098, “Operator Staffing for Small or Multi-Module Nuclear Power Plant Facilities,” dated July 22, 2011. For the reasons described in Chapter 18, Section 18.5.4.2, “Evaluation of the Applicant’s Technical Basis,” of the final safety evaluation report, 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 final rule includes the alternative staffing requirement provisions, staffing table, and appropriate table notes, a future applicant or licensee that references appendix G to 10 CFR part 52 will not need to request an exemption from § 50.54(m).

2. Preoperational and Periodic Testing of Primary Reactor Containment

General Design Criterion (GDC) 52, “Capability for Containment Leakage Rate Testing,” requires that the containment be designed so that periodic, integrated leakage rate testing can be conducted at containment design pressure; the underlying purpose of which is to provide design capability for testing that assures that containment leakage integrity is maintained and containment vessel leakage does not exceed allowable leakage rate values (see appendix J to 10 CFR part 50). Under 10 CFR 50.54(o), operating licenses and combined licenses for certain water-cooled power reactors must include a condition that the primary containment shall be subject to appendix J to 10 CFR part 50, “Primary Reactor Containment Leakage Testing for Water-Cooled Power

Reactors." Appendix J to 10 CFR part 50 requires that primary reactor containments meet the containment leakage test requirements to provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment (Type A) and systems and components that penetrate containment (Type B and Type C).

NuScale Power requested an exemption from GDC 52 in order to not design NuScale to include the capability for Type A testing and requested that the design certification rule exempt licensees referencing the NuScale design certification rule from the requirement for Type A testing in appendix J to 10 CFR part 50. NuScale Power's request was based on the NuScale small modular reactor design meeting the underlying purpose of the regulation through means not anticipated when the NRC issued GDC 52 and appendix J to 10 CFR part 50. NuScale Power stated that the NuScale containment has two primary features distinguishing it from containments at existing light water reactors that provide assurance that no unknown leakage pathways will be present. First, the NuScale containment is designed and would be constructed as a pressure vessel, and therefore leakage due to vessel design or fabrication flaws would be identified during a required preservice structural integrity test. In contrast to a Type A test, this test is a hydrostatic leakage test at design pressure, with no visible leakage as its acceptance criterion. Second, the containment is 100-percent inspectable, both inside and outside, whereby aging-related flaws leading to potential leakage could be observed. Containment leakage integrity assurance for NuScale is described in detail in technical report TR-1116-51962-NP, "NuScale Containment Leakage Integrity Assurance," Rev. 1 (May 2019), which this final rule incorporates by reference. NuScale Power stated that the required preservice tests and inservice inspections described in TR-1116-51962-NP, including Type B and Type C testing without Type A testing, ensure that containment leakage rates remain acceptable.

In Chapter 6, Section 6.2.6.4, "*Technical Evaluation for Exemption Request No. 7*," of the final safety evaluation report, the NRC staff concluded that granting this exemption from Type A testing, and associated design features required by GDC 52 to provide for Type A testing, is acceptable because the NuScale design relies on the preservice pressure test, successful Type B and C testing at each refueling as required in appendix J to 10 CFR part 50, periodic inservice inspections, and direct observation of the entire vessel to identify potential degradation or unknown leakage pathways for the remainder of the service life for the containment.

The NRC received a comment that the exemption from the requirement for Type A testing in appendix J to 10 CFR part 50 should have been listed in the proposed rule. The NRC agrees that the exemption should have been included in the proposed rule. The NRC's conclusion that Type A testing is not necessary for NuScale was noticed for comment as the basis for the exemption from GDC 52. The exemption from Type A testing itself was discussed in detail in the same section of final safety evaluation report that evaluated the exemption from GDC 52. Although the exemption from Type A testing was not included in the proposed rule, the change to this final rule only specifies that future licensees that reference this final rule will not be required to perform Type A testing for which NuScale is not designed or required to be capable of. Therefore, the NRC concludes that the exemption from the Type A test in appendix J to 10 CFR part 50 is a logical outgrowth of the proposed rule. In addition, because the issue of whether Type A testing is necessary for NuScale was noticed in the proposed rule and the NRC received no comments on the matter, the NRC finds that notice and comment on this exemption from Type A testing is unnecessary within the meaning of 5 U.S.C. 553(b).

Thus, Section V, "Applicable Regulations," in this final rule includes an exemption for licensees referencing appendix G to 10 CFR part 52 from the requirement of appendix J to 10 CFR part 50 to conduct Type A testing.

B. Incorporation by Reference

Section III.A, “Incorporation by reference approval,” of appendix G to 10 CFR part 52 lists documents that were approved by the Director of the Office of the Federal Register for incorporation by reference into this appendix. 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.”

The final rule has been updated to align with the Office of the Federal Register’s latest guidance for incorporation by reference, issued on March 1, 2022, as supplemented by Release 1-2022 to the Incorporation by Reference Handbook.

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 feedwater 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. An applicant is required to demonstrate penetration shielding adequate to address the following issues in the NuScale DCD: the plant radiation zones, environmental qualification dose calculations, and dose estimates for workers and the public. An 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 an applicant that references this appendix without a demonstrable impact on safety or standardization. Appendix G to 10 CFR part 52, Section VI, "Issue Resolution," clarifies that this issue is not resolved within the meaning of § 52.63(a)(5), and Section IV, "Additional Requirements and Restrictions," states 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 the 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, 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. An applicant is required under §§ 50.34(f)(2) and 52.47(a)(2) 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 would remain within applicable dose limits in 10 CFR part 20, "Standards for Protection Against Radiation." More information may be available at the 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 likely would 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 application review to determine whether there is reasonable assurance of adequate protection of public health and safety. Therefore, the NRC has determined that this information can be provided by an applicant that references this

appendix without a demonstrable impact on safety or standardization. Appendix G to 10 CFR part 52, Section VI, "Issue Resolution," clarifies that this issue is not resolved within the meaning of § 52.63(a)(5), and Section IV, "Additional Requirements and Restrictions," states 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 (ADAMS Accession No. ML18310A345), 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 (ADAMS Accession No. ML19241A431), 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." Revision 5 of this section 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 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 52.47(a)(2)(iv) and appendix A to 10 CFR part 50, 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, as described in the next paragraph, NuScale power did not provide verification and validation for its proposed method of analysis to demonstrate it is appropriate for this purpose.

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 technical report TR-1016-51669-NP 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 acknowledge 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-NP, "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 would 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 need to be established 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 an 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," clarifies that this issue is not resolved within the

meaning of § 52.63(a)(5), and Section IV, "Additional Requirements and Restrictions," states that the COL applicant is responsible for providing the design information to address this issue.

D. The Term "Multi-unit" as Applied to NuScale

In a letter response to NuScale Power dated October 25, 2016, the NRC staff explained how the staff's review of NuScale would apply the definitions for "nuclear power unit" from Appendix A to 10 CFR part 50, "General Design Criteria for Nuclear Power Plants," and "modular design" from § 52.1, "Definitions." As defined in Appendix A to 10 CFR part 50, a nuclear power unit is the combination of a nuclear reactor and the equipment for power generation. As defined in § 52.1, modular design means that the nuclear power station consists of two or more essentially identical nuclear reactors (modules) and that each module is capable of operation independent of the other modules, even if they have some shared systems.

The NuScale modular design combines one or more nuclear reactors (up to 12) with the necessary equipment for power generation, such that each separate nuclear reactor can be operated independent of the stage of completion or operating condition of any other nuclear reactor on the same site. Therefore, each reactor (i.e., power module) is a separate nuclear power unit. However, NuScale's modular design means that some multi-unit considerations are integral to the design. The NuScale DCD addresses multi-unit considerations other than construction for up to 12 power modules in a single reactor building, but the NuScale DCD does not address multi-unit issues that may arise if a NuScale facility is constructed and operated on the same site as another nuclear facility.

For previously certified or licensed power reactor designs (one nuclear power

unit per reactor building), multi-unit site considerations arose when multiple nuclear power units (in separate reactor buildings) on the same site could affect the construction or operation of another unit in a manner not previously reviewed by the NRC. However, because the NuScale design has been reviewed and is certified for multiple units in a single reactor building, issues related to multiple NuScale units in the same reactor building constructed at the same time have been resolved. Future applicants referencing the NuScale design certification will need to address multi-unit construction issues and, if applicable, multi-unit issues for a proposed NuScale facility to be constructed and operated on the same site as another nuclear facility, including adding additional NuScale modules to a previously licensed NuScale reactor building.

The NRC has added a definition of the term "nuclear power unit" to this final rule.

IV. Technical Issues Associated with the NuScale Design

The NRC identified significant technical issues associated with the following design areas that were resolved 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 Systems 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 Power'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, Systems, Components and Equipment," 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 in Section III.C.3, 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 that 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-NP, Revision 3, “Containment Response Analysis Methodology,” May 2020, which describes the conservative containment pressure and temperature safety analyses for several design-basis events related to the containment design margins. NuScale Power also submitted topical report TR-0516-49422-NP, “Loss-of-Coolant Accident Evaluation Model,” Revision 1, dated November 2019. 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. 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 Power 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 decisionmaking, 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 under § 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 under 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 meet the requirements of the 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, Systems, Components and Equipment,” 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 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 Power does not apply the single failure criterion to the IAB valve, and, on July 2, 2019, 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," 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"; 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"; and SRM-SECY-19-0036, the NRC has retained 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.

D. Conformance with General Design Criterion 27, "Combined Reactivity Control Systems 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 covers 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").

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,'" dated October 9, 2018. The NRC granted the exemption request.

E. Absence of 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 lower 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, Revision 1, “Safety Classification of Passive Nuclear Power Plant Electrical Systems,” dated February 7, 2017. The NRC reviewed topical report TR-0815-16497 and concluded that NuScale Power demonstrated that the 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, issued December 13, 2017, as described in the final safety evaluation report for Chapter 1, “Introduction and General Discussion,” and included in the approved version of the topical report, TR-0815-16497-NP-A.

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 structures, systems, and components 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."

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 that 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. 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.

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 under § 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.

The NRC staff submitted a related information paper to the Commission, SECY-19-0079, "Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application," dated August 16, 2019, describing the regulatory and technical issues raised by unique aspects of NuScale Power's 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 24,

2019. The approved version of topical report TR-0915-17565-NP-A, Revision 4, is discussed in the final 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,” Section 15.0.2, “Review of Transient and Accident Analysis Methods,” 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, “Conformance with General Design Criterion 27, ‘Combined Reactivity Control Systems 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

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 previously (see Table 1.14-1, "NuScale Design Certification Exemptions," in Chapter 1 of the final safety evaluation report):

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 50, GDC 52, "Capability for Containment Leakage Rate Testing" and Appendix J to 10 CFR part 50 (Type A 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 Models)
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 report where these exemption requests have been evaluated. The NRC granted each exemption request.

I. Differing Professional Opinion Related to Chapter 3 of NuScale

On September 17, 2020, a Differing Professional Opinion (DPO) was submitted that raised concerns related to the seismic margin evaluation of the NuScale reactor building and its structural response during the review level earthquake. An ad-hoc review panel was formed and tasked to review the DPO. The review panel subsequently issued its report to the Director of the Office of Nuclear Reactor Regulation (NRR) on

April 19, 2021. On May 19, 2021, the Director of NRR issued a decision to the DPO submitter. For the reasons described in the decision, the Director of NRR agreed with the review panel's finding that the NuScale reactor building design was complete and acceptable for the purposes of a design certification application. On June 14, 2021, the DPO submitter appealed the DPO decision to the Executive Director for Operations (EDO).

After consideration of the issues raised in the appeal, the EDO issued a decision on the DPO appeal on February 8, 2022. The EDO directed NRR to (1) document its evaluation of the stress averaging approach used in the NuScale design certification application, including, if necessary, updating the Final Safety Evaluation Report and assess whether there are any impacts to the standard design approval, and (2) evaluate and update guidance, or create knowledge management tools, on how to assess applications that use stress averaging for structural building design. On February 14, 2022, the DPO submitter responded to the EDO's DPO appeal decision. In this response, the submitter thanked the EDO for thoughtful consideration of the concerns raised and provided clarification regarding the applicability of the Probabilistic Risk Assessment-based seismic margin analysis to the reactor building. After reviewing and considering the submitter's response to the DPO appeal decision, on March 15, 2022, the EDO directed the NRC staff to review and consider the totality of the information provided by the submitter when addressing the tasks mandated in the DPO appeal decision.

In response to the EDO tasking, on May 13, 2022, the Director of NRR issued a memo to the EDO ("Response to DPO Tasking") discussing the staff's review of the items described in the tasking, documenting the staff's evaluation of the approach used in the NuScale design certification, and detailing the staff's assessment of existing related structural analysis guidance (ADAMS Accession No. ML22062A007). The

Director of NRR concluded that the staff sufficiently assessed the evaluation of the demand (force/moment) averaging approach used in the NuScale DCA; justified the acceptability to conclude that there are no impacts to the NuScale standard design approval issued in September 2020; determined that an update or supplement to the final safety evaluation report for the NuScale DCA is not necessary; and found that the existing review guidance is sufficient to review and evaluate an applicant's structural analysis/design. Details on the EDO's decision on the DPO appeal and related correspondence, and the Response to DPO Tasking are found in the information package for DPO-2020-004 (ADAMS Accession No. ML22122A116).

The NRC staff's assessment of NuScale's use of the demand (force/moment) averaging approach is documented in the Response to DPO Tasking. The Response to DPO Tasking elaborates on the reasons for, but does not change, the conclusion in the final safety evaluation report. Based on this assessment, the NRC concludes that the use of the demand (force/moment) averaging approach is acceptable, as stated in the final safety evaluation report.

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 after the Advisory Committee on Reactor Safeguards (ACRS) performed its final independent review and issued its July 29, 2020, letter to the Commission on its findings and recommendations. 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, as elaborated on by the Response to DPO Tasking, reflects that NuScale Power has resolved all technical and safety issues with the exception of the three issues discussed previously. As noted above, the Response to DPO Tasking elaborates on the reasons for, but does not change, the conclusion in the final safety evaluation report that NuScale's use of the demand (force/moment) averaging approach is acceptable as a realistic engineering practice.

In addition, the final safety evaluation report describes the portions of the design that are not receiving finality in this rule and, therefore, are not part of the certified design. The final safety evaluation report also 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.

NuScale Design Certification Final Rule

This section describes the purpose and key aspects of each section of this NuScale design certification final 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 final 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 is approved by this design certification final 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 contracts with the design certification applicant to obtain the generic DCD and supporting design information. If a COL applicant does not use the design certification applicant to provide the design information and instead uses an alternate vendor, then the COL applicant must meet the requirements in § 52.73. Second, paragraph X.A.1 requires 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 final 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 is 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 will also have to include considerations for a multi-unit site in the plant-specific DCD that were not previously evaluated as part of the design certification rule, e.g., construction impacts on operating units. 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 Part 4, "Technical Specifications," of the DCA 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 (SRM), dated February 15, 1991, on SRM-SECY-90-377, "Requirements for Design Certification under 10 CFR part 52," dated November 8, 1990.

Tier 1 information means the portion of the design-related information contained in the generic DCD that is approved and certified by this appendix. Tier 2 information means the portion of the design-related information contained in the generic DCD that is approved but not certified by this appendix. 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 final rule. The NRC is including a definition for "*Departure from a method of evaluation*" in paragraph II.F of appendix G to 10 CFR part 52, 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 applicants and licensees referencing appendix G to 10 CFR part 52. 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, 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.

In paragraph IV.C, the NRC lists design-specific regulations that apply to licenses that reference this appendix.

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 are resolved by the NRC through this final 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 are resolved by this final rule, paragraph VI.C identifies issues that are not resolved by this final 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 final 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 final 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

[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.

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.

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 provides 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, 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 license 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 required by 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 requires 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, 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*. 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 created 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 under § 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 provides 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 will 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 that 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 departures, the Commission would not have to consider whether the special circumstances for the Tier 2 departures would 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 for standardization similar to that imposed on Tier 1 departures by § 52.63(a)(4) and (b)(1) because it would unnecessarily restrict the flexibility of applicants and licensees with respect to Tier 2 information.

An applicant or licensee may 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 will 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 allows 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 is similar to the requirement in § 50.59 and applies to all of the information in Tier 2 except for the information that resolves the severe accident issues or the information required by § 52.47(a)(28) to address aircraft impacts.

Paragraph VIII.B.5.d 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 NRC's hearing requirements at § 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 or other operational requirements in the generic DCD is 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 is 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 an applicant for a COL referencing appendix G to 10 CFR part 52 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). 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 specification 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 is 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 referenced SUNSI and SGI be maintained by the applicant for this final 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, or its equivalent, is not in the generic DCD, it is required to be provided by an applicant for a license referencing appendix G to 10 CFR part 52. Only the generic DCD is identified and incorporated by reference by this final 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 this final rule incorporates 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.B 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 final 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.3. The frequency of the report submittals is set forth in paragraph X.B.3. The requirement for submitting a summary of the evaluations is 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.

Paragraph X.B.3.b also requires semi-annual submission of the reports required by paragraphs X.B.1 and X.B.2 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. Public Comment Analysis

The NRC prepared a summary and analysis of public comments received on the 2021 proposed rule, as referenced in the “Availability of Documents” section. The NRC received eight comment submissions during the public comment period that ended on October 14, 2021, and one late-filed comment submission on October 15, 2021, that the NRC was able to include in its consideration for this final rule. A *comment submission* is a communication or document submitted to the NRC by an individual or entity, with one or more individual comments addressing a subject or issue. Private citizens provided four comment submissions, nuclear industry organizations provided two comment submissions, science advocacy groups provided two comment submissions, and a labor union provided one comment submission. Of the nine comments, six were in favor of the design certification rule, one was opposed, and the other two comment submittals posed questions but stated no preference for the outcome of the rule. Six of the nine comment submissions contained questions on technical aspects of the design, corrections to the statement of considerations, and interpretation of requirements.

The public comment submittals are available on the Federal rulemaking website under Docket ID NRC-2017-0029. NRC’s response to the public comments, including a summary of how NRC revised the proposed rule in response to public input, can be found in the public comment analysis document.

VII. Section-by-Section Analysis

The following paragraphs describe the specific changes in this final rule:

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

In § 52.11, this final rule adds 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 final rule adds 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.

VIII. Regulatory Flexibility Certification

Under the Regulatory Flexibility Act (5 U.S.C. 605(b)), the NRC certifies that this rule does not have a significant economic impact on a substantial number of small entities. This final 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).

IX. Regulatory Analysis

The NRC has not prepared a regulatory analysis for this final 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.

X. Backfitting and Issue Finality

The NRC has determined that this final 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 final 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 final 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 final rule.

XI. 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).

XII. Environmental Assessment and Finding of No Significant Impact

The NRC conducted an environmental assessment and has determined under the National Environmental Policy Act of 1969, as amended (NEPA), and the NRC's regulations in subpart A of 10 CFR part 51, that this final 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.

The NRC's generic determination in this regard, as discussed in the 2007 final rule amending 10 CFR parts 51 and 52 (72 FR 49351; August 28, 2007), is based upon consideration that 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 NEPA as part of the application

for the construction and operation of a facility referencing any particular design certification rule.

Consistent with §§ 51.30(d) and 51.32(b), the NRC has prepared an environmental assessment 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." 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 the site parameters of the representative site 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 representative site. As noted in the environmental assessment, SAMDA candidates for multi-unit sites are evaluated in the context of multiple NuScale reactor buildings, each with up to 12 power

modules at the same site. Site-specific SAMDAs, multi-unit aspects, procedural and training SAMDAs, and the design element details of the reactor building crane will need to be assessed when an application for a specific site is submitted to construct and operate 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 XVIII of this document.

XIII. Paperwork Reduction Act

This final rule contains new or amended collections of information subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). The collections of information were approved by the Office of Management and Budget, approval number 3150-0151.

The burden to the public for the information collections is estimated to average 130 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection.

The information collection is being conducted to fulfill the requirements of a future applicant that references the design certification to maintain records of changes to and departures from the generic DCD, which are to be reflected in the plant-specific DCD. This information will be used by the NRC to fulfill its responsibilities in the licensing of nuclear power plants. Responses to this collection of information are mandatory. Confidential and proprietary information submitted to the NRC is protected in accordance with NRC regulations at §§ 9.17(a) and 2.39(b).

You may submit comments on any aspect of the information collections, including suggestions for reducing the burden, by the following methods:

- **Federal rulemaking website:** Go to https://www.regulations.gov 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; email: oira_submission@omb.eop.gov.

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.

XIV. Congressional Review Act

This final rule is a rule as defined in the Congressional Review Act (5 U.S.C. 801-808). However, the Office of Management and Budget has not found it to be a major rule as defined in the Congressional Review Act.

XV. Agreement State Compatibility

Under the “Agreement State Program Policy Statement” approved by the Commission on October 2, 2017, and published in the *Federal Register* on October 18, 2017 (82 FR 48535), this 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 AEA or the provisions of title 10 of the *Code of Federal Regulations*, and although an Agreement State may not adopt program elements reserved to the NRC, it may wish to inform its licensees of certain requirements via a mechanism that is consistent with a particular State’s administrative procedure laws, but does not confer regulatory authority on the State.

XVI. 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 final rule, the NRC certifies 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.

XVII. 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.

Documents Related to NuScale Design Certification Rule

DOCUMENT	ADAMS ACCESSION NO./WEB LINK/ *FEDERAL REGISTER* CITATION
SECY-22-0062, "Final Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029)," July 1, 2022	ML22004A002
SECY-21-0004, "Proposed Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029)," January 14, 2021	ML19353A003
Staff Requirements Memorandum for SECY-21-0004, "Proposed Rule: NuScale Small Modular Reactor Design Certification (RIN 3150-AJ98; NRC-2017-0029)," May 6, 2021	ML21126A153
Annotated Comment Submissions on Proposed Rule: NuScale Small Modular Reactor Design Certification (NRC-2017-0029; RIN 3150-AJ98), June 2022	ML22045A213
Final Rule Comment Response Document for NuScale Small Modular Reactor Design Certification (public comment analysis document), July 2022	ML22216A015
NuScale Power, LLC, Submittal of the NuScale Standard Plant Design Certification Application, Revision 5, July 2020	ML20225A071
NuScale Standard Design Certification Application, Part 3, "Applicant's Environmental Report - Standard Design Certification," Revision 5, July 2020	ML20224A512
NuScale Power, LLC, Submittal of the NuScale Standard Plant Design Certification Application, Revision 4.1, June 19, 2020	ML20205L562
NuScale Power, LLC, Submittal of the NuScale Standard Plant Design Certification Application, Part 2, Tier 2, Revision 3, August 2019	ML19241A431
NuScale Power, LLC, Submittal of the NuScale Standard Plant Design Certification Application, Part 2, Tier 2, Revision 2, October 2018	ML18310A345
NuScale Power, LLC, Topical report TR-0915-17565, Revision 3, Accident Source Term Methodology, April 21, 2019	ML19112A172

Proposed Rule for the NuScale Small Modular Reactor Design Certification, July 1, 2021	86 FR 34999
Extension of Comment Period for the Proposed Rule, August 24, 2021	86 FR 47251
Docketing Notice for the NuScale Power, LLC, Design Certification Application (DCA), March 30, 2017	82 FR 15717
Notification of Receipt of the NuScale Power, LLC, Design Certification Application (DCA), February 22, 2017	82 FR 11372
NuScale Power, LLC, Submittal of the NuScale Standard Plant Design Certification Application (NRC Project No. 0769), Revision 0, December 2016	ML17013A229
NuScale Power, LLC, Submittal of NuScale Preliminary Concept of Operations Summary and Response to NRC Questions on Control Room Activities, September 15, 2015	ML15258A846
Information on Differing Professional Opinion (DPO) 2020-004, May 13, 2022	ML22122A116
Final Safety Evaluation Report and Supporting Documents	
NuScale DCA Final Safety Evaluation Report, August 2020	ML20023A318
NRC Safety Evaluation for NuScale Power, LLC, Topical Report, TR-0516-49422, “Loss-of-Coolant,” Revision 1, November 2019	ML20044E199
NRC Safety Evaluation for NuScale Power, LLC, Topical Report, TR-0815-16497, Revision 1, “Safety Classification of Passive Nuclear Power Plant Electrical Systems,” December 13, 2017	ML17340A524
NRC Safety Evaluation for NuScale Power, LLC, Topical Report, TR-0915-17565, Rev. 3, “Accident Source Term Methodology,” October 24, 2019	ML19297G520
NRR Response to Taskings in EDO DPO Appeal Decision Concerning DPO-2020-004, May 13, 2022	ML22062A007
Environmental Reviews	
Final Environmental Assessment by the U.S. Nuclear Regulatory Commission Relating to the Certification of the NuScale Standard Design, July 2022	ML22216A014
Environmental Assessment by the U.S. Nuclear Regulatory Commission Relating to the Certification of the NuScale Standard Design, January 14, 2021	ML19303C179
Staff Technical Analysis in Support of the NuScale Design Certification Environmental Assessment, August 4, 2020	ML19302E819
Commission Papers, Staff Requirement Memoranda, and Other Supporting Documents	
SECY-11-0098, “Operator Staffing for Small or Multi-Module Nuclear Power Plant Facilities,” July 22, 2011	ML111870574

SECY-17-0075, “Planned Improvements in Design Certification Tiered Information Designations,” dated July 24, 2017	ML16196A321
SECY-18-0099, “NuScale Power Exemption Request from 10 CFR Part 50, Appendix A, General Design Criterion 27, ‘Combined Reactivity Control Systems Capability,’” dated October 9, 2018	ML18065A431
SECY-19-0079, “Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application,” August 16, 2019	ML19107A455
SECY-77-439, “Single Failure Criterion,” August 17, 1977	ML060260236
SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” April 2, 1993	ML003708021
SRM-SECY-19-0036, “Staff Requirements—SECY-19-0036—Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves,” July 2, 2019	ML19183A408
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,” June 30, 1994	ML003708098
SRM-SECY-90-377, “Requirements for Design Certification under 10 CFR part 52,” February 15, 1991	ML003707892
Response to NuScale Power, LLC Key Issue Resolution Letter, Supplemental Response Regarding Multi-Module Questions, October 25, 2016	ML16229A522
Advisory Committee on Reactor Safeguards (ACRS) Letter, “Report on the Safety Aspects of the NuScale Small Modular Reactor,” July 29, 2020	ML20211M386
American Society of Mechanical Engineers Standard QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants,” 2007	https://webstore.ansi.org/standards/asme/ansiasmeqme2007
NRC Regulatory Guide 1.100, Rev. 3, “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” September 2009	ML091320468
NRC Regulatory Guide 1.206, Rev. 1, “Applications for Nuclear Power Plants,” October 2018	ML18131A181
NRC Agreement State Program Policy Statement, October 18, 2017	82 FR 48535
Final Rule for Licenses, Certifications, and Approvals for Nuclear Power Plants (10 CFR parts 51 and 52), August 28, 2007	72 FR 49351
Office of the Federal Register (OFR) Final Rule for Incorporation by Reference, November 7, 2014	79 FR 66267

Presidential Memorandum, “Plain Language in Government Writing,” June 10, 1998	63 FR 31883
Regulatory History of Design Certification, April 2000[2]	ML003761550
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 [3]	ML16364A342
RP-0316-17615-NP, Human Factors Engineering Functional Requirements Analysis and Function Allocation Results Summary Report, Rev. 0, December 2016 [3]	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 [3]	ML17004A222
RP-0316-17618-NP, Human Factors Engineering Treatment of Important Human Actions Results Summary Report, Rev. 0, December 2016 [3]	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 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

[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.

[3] The duplicate ADAMS Accession Nos. ML16364A342 and ML17004A222 are intentional and indicate when multiple reports are part of a single submittal.

TR-0116-20825-NP-A, Applicability of AREVA Fuel Methodology for the NuScale Design, Rev. 1, June 2016	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, November 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, April 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, January 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, February 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, May 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, May 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, 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 website at https://www.regulations.gov under Docket ID NRC-2017-0029. In addition, the Federal rulemaking website allows members of the public to receive alerts when changes or additions occur in a docket folder. To subscribe: 1) navigate to the docket folder (NRC-2017-0029); 2) click the "Subscribe" link; and 3) enter an email address and click on the "Subscribe" link.

XVIII. Incorporation by Reference—Reasonable Availability to Interested Parties

The NRC is incorporating 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. On November 7, 2014, the OFR adopted changes to its regulations governing incorporation by reference (79 FR 66267). The OFR regulations require an agency to discuss, in the preamble of the final rule, the ways that the materials it incorporates by reference are reasonably available to interested parties and how interested parties can obtain the materials. The discussion in this section complies with the requirement for final rules 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[4] 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; email: Library.Resource@nrc.gov. In addition, as described in Section XVIII of this document, documents related to this final 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 final 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.

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[4] 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.

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 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 is amending 10 CFR part 52 as follows:

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), remove the phrase "appendices A, B, C, D, E, F, and N of this part" and add, in its place, the phrase "appendices A, B, C, D, E, F, G, and N of this part".

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 the NuScale design (hereinafter referred to as NuScale), in accordance with 10 CFR part 52, subpart B. The applicant for this standard design certification NuScale is NuScale Power, LLC*.*

II. Definitions

A. *Generic design control document (generic DCD)* means the documents containing the Tier 1 and Tier 2 information (including the technical and topical reports referenced in Chapter 1) and generic technical specifications that are 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 combined license (COL) final safety analysis report (FSAR) 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. 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 action items (COL license information) identify certain matters that must be addressed in the site-specific portion of the FSAR by an applicant who references this appendix. These items constitute information requirements but are not the only acceptable set of information in the FSAR. An applicant may depart from or omit these items, provided that the departure or omission is identified and justified in the FSAR. After issuance of a construction permit or COL, these items are not requirements for the licensee unless such items are restated in the FSAR.

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. *Nuclear power unit*, as applied to this certified design, means a nuclear power module and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

H. 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.

1. Certain material listed in paragraph III.A.2 of this appendix is incorporated by reference into this appendix G with the approval of the Director of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR part 51. All approved incorporation by reference (IBR) material in paragraph III.A.2 of this appendix may be obtained from NuScale Power, LLC, 6650 SW Redwood Lane, Suite 210, Portland, Oregon, 97224, telephone: 1-971-371-1592, email: RegulatoryAffairs@nuscalepower.com, and can be inspected as follows:

a. Contact the U.S. Nuclear Regulatory Commission at: U.S. Nuclear Regulatory Commission, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852; telephone: 301-415-7000; email: Library.Resource@nrc.gov; https://www.nrc.gov/reading-rm/pdr.html.

b. Access ADAMS and view the material online in the NRC Library at

https://www.nrc.gov/reading-rm/adams.html. In ADAMS, search under ADAMS Accession No. ML20225A071. The material is available in the ADAMS Public Documents collection.

c. 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 email at PDR.Resource@nrc.gov.

d. For information on the availability of this material at the National Archives and Records Administration, visit www.archives.gov/federal-register/cfr/ibr-locations.html or email: fr.inspection@nara.gov.

2. Material incorporated by reference.

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

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

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

ii. Chapter Two, Site Characteristics and Site Parameters.

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

iv. Chapter Four, Reactor.

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

vi. Chapter Six, Engineered Safety Features.

vii. Chapter Seven, Instrumentation and Controls.

viii. Chapter Eight, Electric Power.

ix. Chapter Nine, Auxiliary Systems.

x. Chapter Ten, Steam and Power Conversion System.

xi. Chapter Eleven, Radioactive Waste Management.

xii. Chapter Twelve, Radiation Protection.

xiii. Chapter Thirteen, Conduct of Operations.

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

xv. Chapter Fifteen, Transient and Accident Analyses.

xvi. Chapter Sixteen, Technical Specifications.

xvii. Chapter Seventeen, Quality Assurance and Reliability Assurance.

xviii. Chapter Eighteen, Human Factors Engineering.

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

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

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

c. DCA Part 4, Volume 1, Revision 5.0, Generic Technical Specifications, NuScale Nuclear Power Plants, Volume 1: Specifications.

d. DCA Part 4, Volume 2, Revision 5.0, Generic Technical Specifications, NuScale Nuclear Power Plants, Volume 2: Bases.

e. ES-0304-1381-NP, Human-System Interface Style Guide, December 2019, Revision 4.

f. RP-0215-10815-NP, Concept of Operations, May 2019, Revision 3.

g. RP-0316-17614-NP, Human Factors Engineering Operating Experience Review Results Summary Report, December 7, 2016, Revision 0.

h. RP-0316-17615-NP, Human Factors Engineering Functional Requirements Analysis and Function Allocation Results Summary Report, December 2, 2016, Revision 0.

i. RP-0316-17616-NP, Human Factors Engineering Task Analysis Results Summary Report, April 2019, Revision 2.

j. RP-0316-17617-NP, Human Factors Engineering Staffing and Qualifications Results Summary Report, December 2, 2016, Revision 0.

k. RP-0316-17618-NP, Human Factors Engineering Treatment of Important Human Actions Results Summary Report, December 2, 2016, Revision 0.

l. RP-0316-17619-NP, Human Factors Engineering Human-System Interface Design Results Summary Report, April 2019, Revision 2.

m. RP-0516-49116-NP, Control Room Staffing Plan Validation Results, December 2, 2016, Revision 1.

n. RP-0914-8534-NP, Human Factors Engineering Program Management Plan, April 2019, Revision 5.

o. RP-0914-8543-NP, Human Factors Verification and Validation Implementation Plan, April 2019, Revision 5.

p. RP-0914-8544-NP, Human Factors Engineering Design Implementation Plan, November 2019, Revision 4.

q. RP-1018-61289-NP, Human Factors Engineering Verification and Validation Results Summary Report, July 2019, Revision 1.

r. RP-1215-20253-NP, Control Room Staffing Plan Validation Methodology, December 2, 2016, Revision 3.

s. TR-0116-20781-NP, Fluence Calculation Methodology and Results, July 2019, Revision 1.

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

u. TR-0116-21012-NP-A, NuScale Power Critical Heat Flux Correlations, December 2018, Revision 1.

v. TR-0316-22048-NP, Nuclear Steam Supply System Advanced Sensor Technical Report, May 2020, Revision 3.

w. TR-0515-13952-NP-A, Risk Significance Determination, October 2016, Revision 0.

x. TR-0516-49084-NP, Containment Response Analysis Methodology Technical Report, May 2020, Revision 3.

y. TR-0516-49416-NP-A, Non-Loss-of-Coolant Accident Analysis Methodology, July 2020, Revision 3.

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

aa. TR-0516-49422-NP-A, Loss-of-Coolant Accident Evaluation Model, July 2020, Revision 2.

ab. TR-0616-48793-NP-A, Nuclear Analysis Codes and Methods Qualification, November 2018, Revision 1.

ac. TR-0616-49121-NP, NuScale Instrument Setpoint Methodology Technical Report, May 2020, Revision 3.

ad. TR-0716-50350-NP-A, Rod Ejection Accident Methodology, June 2020, Revision 1.

ae. 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.

af. TR-0716-50424-NP, Combustible Gas Control, March 2019, Revision 1.

ag. TR-0716-50439-NP, NuScale Comprehensive Vibration Assessment Program Analysis Technical Report, July 2019, Revision 2.

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

ai. TR-0816-49833-NP, Fuel Storage Rack Analysis, November 2018, Revision 1.

aj. TR-0816-50796-NP, Loss of Large Areas Due to Explosions and Fires Assessment, June 2019, Revision 1.

ak. TR-0816-50797, Mitigation Strategies for Loss of All AC Power Event [NuScale Nonproprietary], October 2019, Revision 3.

al. TR-0816-51127-NP, NuFuel-HTP2™ Fuel and Control Rod Assembly Designs, December 2019, Revision 3.

am. TR-0818-61384-NP, Pipe Rupture Hazards Analysis, July 2019, Revision 2.

an. TR-0915-17564-NP-A, Subchannel Analysis Methodology, February 2019, Revision 2.

ao. TR-0915-17565-NP-A, Accident Source Term Methodology, February 2020, Revision 4.

ap. TR-0916-51299-NP, Long-Term Cooling Methodology, May 2020, Revision 3.

aq. TR-0916-51502-NP, NuScale Power Module Seismic Analysis, April 2019, Revision 2.

ar. TR-0917-56119-NP, CNV Ultimate Pressure Integrity, June 2019, Revision 1.

as. TR-0918-60894-NP, NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report, August 2019, Revision 1.

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

au. TR-1015-18177-NP, Pressure and Temperature Limits Methodology, October 2018, Revision 2.

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

aw. TR-1016-51669-NP, NuScale Power Module Short-Term Transient Analysis, July 2019, Revision 1.

ax. TR-1116-51962-NP, NuScale Containment Leakage Integrity Assurance, May 2019, Revision 1.

ay. TR-1116-52065-NP, Effluent Release (GALE Replacement) Methodology and Results, November 2018, Revision 1.

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 wholly 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 action 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 requirements of 10 CFR 52.47(a)(2)(iv) and General Design Criterion (GDC) 4 and GDC 31 of appendix A to 10 CFR part 50 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.

C. A licensee referencing the NuScale design certification is exempt from portions of the following regulation:

1. Paragraph (m) of 10 CFR 50.54—Minimum Staffing. 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 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 identified in Table 1:

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.

2. Appendix J to 10 CFR part 50, Type A testing — Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.

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 **February 21, 2023**, 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:

1. Paragraph (f)(2)(vi) of 10 CFR 50.34 and 10 CFR 50.46a—High point venting for the reactor coolant system and reactor pressure vessel head.

2. Paragraph (f)(2)(viii) of 10 CFR 50.34—Post-accident sampling of the reactor coolant system and containment.

3. Paragraph (f)(2)(xiii) of 10 CFR 50.34—Power supplies for pressurizer heaters.

4. Paragraph (f)(2)(xiv)(E) of 10 CFR 50.34—Automatic closing of containment isolation systems on a high radiation signal.

5. Paragraph (f)(2)(xx) of 10 CFR 50.34—Power from vital buses and emergency power sources for pressurizer level indication.

6. Paragraph (c)(2) of 10 CFR 50.44—Combustible gas control.

7. Paragraph (a)(1)(i) of 10 CFR 50.46—Applicability limited to reactor designs that use zircaloy or ZIRLO fuel rod cladding material.

8. Paragraph (c)(1) of 10 CFR 50.62—Diverse equipment to initiate a turbine trip under conditions indicative of an anticipated transient without scram.

9 Appendix A of 10 CFR part 50—Electric Power Systems GDCs:

a. GDC 17—Electric power systems for safety-related functions;

b. GDC 18—Design to permit periodic inspection and testing of electric power systems;

c. GDC 34—Electric power systems for residual heat removal;

d. GDC 35—Electric power systems for emergency core cooling;

e. GDC 38—Electric power systems for containment heat removal;

f. GDC 41—Electric power systems for containment atmosphere cleanup; and

g. GDC 44—Electric power systems for cooling.

10. Appendix A to 10 CFR part 50, GDC 19—Equipment outside the control room with capability for cold shutdown of the reactor.

11. Appendix A to 10 CFR part 50, GDC 27—Demonstration of long-term shutdown under post-accident conditions with an assumed worst rod stuck out.

12. Appendix A to 10 CFR part 50, GDC 33—Reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary.

13. Appendix A to 10 CFR part 50, GDC 40—Periodic pressure and functional testing of containment heat removal system.

14. Appendix A to 10 CFR part 50, GDC 52—Design to allow periodic containment leakage rate testing.

15. Appendix A of 10 CFR part 50, GDCs 55, 56, and 57—Containment Isolation:

a. GDC 55—Isolation valves for certain reactor coolant pressure boundary lines penetrating containment;

b. GDC 56—Isolation valves for certain primary containment lines; and

c. GDC 57—Isolation valves for certain closed systems lines.

16. Appendix K to 10 CFR part 50—Emergency Core Cooling System Evaluation Models:

a. Section I.A.4—Heat generation rates from radioactive decay of fission products;

b. Section I.A.5—Rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction;

c. Section I.B—Predicting cladding swelling and rupture;

d. Section I.C.1.b—Calculation of the discharge rate for all times after the discharging fluid has been calculated to be two-phase;

e. Section I.C.5.a—Post-critical heat flux correlations of heat transfer from the fuel cladding to the surrounding fluid; and

f. Section I.C.7.a—Calculation of cross-flow between the hot and average channel regions of the core during blowdown.

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 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. ML22004A006) 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 the site parameters of the representative site 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 **February 21, 2023**, 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 bears on an asserted noncompliance with an ITAAC acceptance criterion in the case of a § 52.103 preoperational hearing, or that the departure bears 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 that 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: January 11, 2023.

For the Nuclear Regulatory Commission.

/RA/

Brooke P. Clark,
Secretary of the Commission.