1.0 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

THIS NRC STAFF DRAFT SE HAS BEEN PREPARED AND IS BEING RELEASED TO SUPPORT INTERACTIONS WITH THE ACRS. THIS DRAFT SE HAS NOT BEEN SUBJECT TO FULL NRC MANAGEMENT AND LEGAL REVIEWS AND APPROVALS, AND ITS CONTENTS SHOULD NOT BE INTERPRETED AS OFFICIAL AGENCY POSITION.

1.1 Introduction

By letter dated November 21, 2022, NuScale Power, LLC (NuScale) informed the U.S. Nuclear Regulatory Commission (NRC) of its intent to submit, in stages, a standard design approval application (SDAA) for a US460 Small Modular Reactor (SMR) to the NRC by December 31, 2022 (available in the NRC Agencywide Documents Access and Management System (ADAMS) Accession No. ML22325A349). NuScale listed its intended individual electronic submissions in that letter. By letter dated December 31, 2022 (electronically submitted on January 1, 2023), NuScale transmitted the last of its listed submittals. NuScale submitted the SDAA pursuant to the requirements of Title 10 of the *Code of Federal Regulations* Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The documents that were submitted as part of NuScale's SDAA can be found in ML22339A066. On October 31, 2023, the applicant submitted SDAA Revision 1 (ML23306A033). On March 17, 2025, the applicant submitted supplemental information for SDAA, Part 2, Chapter 1 (ML25076A571).

The SDAA consists of 7 parts: Part 1, "General and Financial Information;" Part 2, "Final Safety Analysis Report (FSAR);" Part 4, "Generic Technical Specifications;" Part 7, "Exemptions;" Part 8, "License Conditions; Inspections, Tests, Analyses & Acceptance Criteria (ITAAC);" Part 9, "Withheld Information;" and Part 10, "Quality Assurance Program Description." The NRC formally accepted and docketed the SDAA (Docket No. 52-050) on July 31, 2023 (ML23198A163). NuScale design information and all other correspondence submitted before that date can be found in ADAMS under Docket No. 99902078.

The applicant's design consists of up to 6 NuScale Power Modules (NPMs). As depicted in SDAA Part 2, Figure 1.2-4, "Cutaway View of NuScale Power Module," and Figure 1.2-5, "Steam Generator and Reactor Flow," the NPM is a collection of systems, subsystems, and components that together constitute a modularized, movable nuclear steam supply system (NSSS). The NPM is composed of a reactor core, a pressurizer, and two steam generators (SGs) integrated within a reactor pressure vessel (RPV) and housed in a compact steel containment vessel (CNV). Each NPM is rated at 250 megawatts thermal (MWt) (up to 1,500 MWt total for 6 NPMs), with approximately 77 megawatts electric (MWe) (up to 462 MWe total for 6 NPMs) output.

The NuScale SDAA Part 2 information includes, among other things, a description of the facility design required for an FSAR by 10 CFR 52.137, "Contents of applications; technical information." To evaluate the NuScale design, the NRC staff (staff) reviewed the SDAA, including all referenced technical and topical reports, and generated its final safety evaluation report (FSER). The FSER is divided into chapters that evaluate the matching chapters in SDAA Part 2. Throughout the course of the review, the staff requested that the applicant submit additional information to clarify the description of the NuScale standard design. The FSER (meaning all chapters, unless stated otherwise) discusses some of the applicant's responses to these requests for additional information (RAIs). Appendix E to the FSER lists the issuance and response dates for each RAI the staff issued to the applicant and supplemental information provided by the applicant. The SDAA Part 2 information and all other pertinent information and materials are available for public inspection at the NRC Public Document Room and through the ADAMS Public Electronic Reading Room.

The FSER documents the staff's safety review of the NuScale SMR design against the requirements of 10 CFR Part 52, Subpart E, and delineates the scope of the technical details considered in evaluating the proposed design. In the FSER, the NRC staff uses the term "non-safety-related" to refer to structures, systems and components (SSCs) that are not classified as "safety-related SSCs" as described in 10 CFR 50.2, "Definitions." However, among the "nonsafety-related" SSCs, there are those that are "important to safety" as that term is used in the General Design Criteria (GDC) listed in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and others that are not considered "important to safety." Appendix F to the FSER includes a copy of the report by the Advisory Committee on Reactor Safeguards (ACRS) required by 10 CFR 52.141, "Referral to the Advisory Committee on Reactor Safeguards (ACRS)."

Graded Review Approach

The staff used a graded review approach to evaluate the NuScale SDAA, consistent with NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), "Introduction—Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Small Modular Reactor Edition," Revision 0, January 2014, (ML13207A315). As such, the staff focused review efforts based on risk significance of SSCs and other aspects of the design that contribute most to safety, thereby improving the efficiency and effectiveness of the review. The graded approach uses a safety significance categorization process that classifies SSCs in one of four review levels that correlate to safety significance:

(1) A1—safety-related and risk significant
(2) A2—safety-related and not risk significant
(3) B1—not safety-related and risk significant
(4) B2—not safety-related and not risk significant

SSCs are categorized as either safety-related or not safety-related using the criteria in 10 CFR 50.2 and as either risk significant or not risk significant using the process developed for the reliability assurance program. The SSCs within the scope of the reliability assurance program are identified by using a combination of probabilistic, deterministic, and other methods of analysis to identify and quantify risk, including probabilistic risk assessment (PRA), severe accident evaluation, assessment of industry operating experience, and expert panel deliberation. The staff received preliminary categorization results from the applicant in the preapplication phase of the staff's review. The staff also conducted preapplication meetings and audits to obtain and review the information on SSC categorization. The final SSC categorization results used for the SDAA review reflect the staff assessment of the applicant's SSC categorization results provided in the SDAA. The staff's assessment includes the review of the PRA, the reliability assurance program, and any design changes resulting from the staff review of the NuScale design.

The staff applied the most rigorous review techniques to SSCs with the highest safety significance and a progressively less detailed review to SSCs with lower assigned safety significance. For example, the staff limited its review of B2 (not safety-related and not risk significant) SSCs mainly to ensuring that the failures of these SSCs would not adversely impact any safety-related functions and to identifying appropriate program requirements to confirm and monitor SSC performance. Other techniques for B2 SSCs included the use of informed sampling methods.

The NuScale design employs a number of unique design features relative to the traditional large light-water reactor designs. The staff considered these and other factors, such as adequacy of defense-in-depth and safety margins and operational program requirements, in focusing the review effort on the most safety significant aspects of the design. In all cases, the staff conducted its review to ensure that the applicant's submittal complies with NRC regulations and that any requests for exemption from certain regulations contain adequate bases and justification.

The staff considered the graded review approach for programmatic and other non-SSC topics. While risk significance associated with these non-SSC topics is not directly quantified, the staff determined the appropriate method for demonstrating satisfaction of the regulatory requirements considering the same qualitative factors (e.g., unique design aspects, defense-in-depth, and safety margins) used for the SSC review to focus the review effort on safety significant aspects. Where applicable, the staff used the safety significance of the SSCs to inform the review focus areas for the non-SSC topics.

The overall objective of the graded review approach is to focus the review effort on those aspects that contribute the most to safety, thereby improving the effectiveness of the review.

1.1.1 Metrication

The FSER conforms to the Commission's policy statement on metrication published in the *Federal Register* on June 19, 1996. According to the policy statement, all measures are to be expressed as metric units, followed by English units in parentheses. An example of a standard conversion would be as follows: 760 millimeters (mm) of mercury and 20 degrees Celsius (C) (14.7 pounds-force per square inch absolute (psia) and 68 degrees Fahrenheit (F)). The precise parameter values in the SDAA, as reviewed by the staff, are provided by the applicant using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in the FSER to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the SDAA and not converted.

1.1.2 Proprietary Information

This FSER references several NuScale reports. Some of these reports contain information that the applicant requested be held exempt from public disclosure, as provided for by 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding." For each report, the applicant provided a nonproprietary version, similar in content except for the redaction of the proprietary information.

1.1.3 Applicants Referencing the NuScale Design

Applicants that reference the NuScale Power Plant standard design for specific facilities will retain architect-engineers, constructors, and consultants, as needed. As part of its review of an application for a combined license (COL) referencing the NuScale design, the staff will evaluate, for each plant-specific application, the technical competence of the COL applicant and its contractors to manage, design, construct, and operate a nuclear power plant. COL applicants will also be subject to the requirements of 10 CFR Part 52, Subpart C, “Combined Licenses,” and any requirements resulting from the staff’s review of this design. Throughout SDAA Part 2, the applicant identified matters to be addressed by plant-specific COL applicants as “COL items.” FSAR, Table 1.8-1, lists COL items identified in the FSAR and this FSER. The list in Table 1.8-1 includes eight COL items belonging to Chapter 1.

1.1.3.1 Plant Location and Schedule

The NuScale Power Plant is designed for use at a site with site characteristics (e.g., seismology, hydrology, meteorology, geology) described in SDAA Part 2, Chapter 2, “Site Characteristics.” The NuScale Power Plant is designed to accommodate up to 6 NPMs. COL Item 1.1-1 provides that a COL applicant that references the NuScale Power Plant US460 standard design is to identify the actual plant site location. The staff finds this COL item acceptable because it supports the COL applicant’s compliance with 10 CFR 52.79(a)(1).

COL Item 1.1-2 states that the COL applicant that references the NuScale Power Plant US460 standard design is to provide the schedules for completion of construction and commercial operation of each power module. The staff finds this COL item acceptable because it supports the COL applicant’s compliance with multiple subsections of 10 CFR 52.79(a).

1.1.4 Additional Information

(Appendices will be added in Phase D of the SDAA review) - Appendix A to this FSER provides a chronology of the principal actions, submittals, and amendments related to the processing of the NuScale Power Plant SDAA. Appendix B lists the references identified in this report. Appendix C defines the acronyms and abbreviations used throughout this report. Appendix D lists the project management and principal technical reviewers who evaluated the NuScale SMR design. Appendix E provides an index of the staff’s RAIs and the applicant’s responses. Appendix F includes a copy of the ACRS letter with the results of its review of those portions of the application that concern safety.

Questions on the SDAA and the staff’s review should be directed to the Office of Nuclear Reactor Regulation, which can be contacted by calling (301) 415-7000 or by writing to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, DC 20555-0001.

1.2 General Design Description

1.2.1 Scope of the NuScale Small Modular Reactor Design

The requirement that governs the scope of the NuScale SMR design can be found in 10 CFR 52.137, which requires that an applicant that seeks review of a major portion of a standard design need only submit the information required by this section to the extent the requirements are applicable to the major portion of the standard design for which NRC staff approval is sought. The scope of the NuScale SMR design include all of the plant SSCs that can affect the safe operation of the plant, except for its site-specific elements. Section 1.2 identifies the structures, systems, and components that are included in the US460 SDAA. Figure 1.2-1 provides a representation of the overall facility and general boundaries between the US460 standard design and site-specific design. FSAR, Section 1.8 describes interfaces with the standard design, including a complete list of information that must be provided in order to license and operate a site-specific NuScale Power Plant, but is not included in the standard approved design, that are identified throughout the FSAR as COL information items (see FSAR, Table 1.8-1, “Combined License Information Items.”

1.2.2 Summary of the NuScale Small Modular Reactor Design

The NuScale SMR is an integrated pressurized-water reactor (PWR). The NuScale NSSS is a passive NuScale-designed small modular PWR. This design encompasses a NuScale Power Module (NPM) consisting of a reactor core, two SG tube bundles, and a pressurizer contained within a single reactor vessel, along with the CNV that immediately surrounds the reactor vessel. This design eliminates the need for external piping to connect the SGs and pressurizer to the RPV. Natural circulation provides the driving force for reactor coolant system (RCS) flow.

The NuScale CNV is an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class MC (steel) containment that is designed, analyzed, fabricated, inspected, tested, and stamped as an ASME Boiler and Pressure Vessel Code Class 1 pressure vessel. The CNV internal pressure is maintained at a vacuum during normal operations. The CNVs are mounted to the reactor building (RXB) module compartment walls and at the bottom within the RXB pool.

The term “NuScale Power Plant” refers to the entire site, including up to 6 NPMs and the associated balance of plant support systems and structures. A NuScale SMR facility can consist of up to 6 NPMs that operate within a single RXB. The information provided in SDAA Part 2 includes the design of an individual NPM, as well as plant design and interfaces between the site and the design for a 6-NPM facility. SDAA Part 2, Section 1.1.4, “Multi-Module Regulatory Considerations” provides information related to multimodule facilities and shared systems.

The application describes the following NuScale design features:

- no alternating current (AC) or direct current (DC) power required for safe shutdown and cooling
- compact helical coil SGs with reactor pressure on the outside of the tubes
- high-strength steel containment partially immersed in a pool of water
- subatmospheric containment pressure during normal operation
- small core with a correspondingly small source term
- comprehensive digital instrumentation and control (I&C) monitoring and control

The design identifies these key features of a multiunit plant:

- a scalable plant design, which allows for incremental plant capacity growth
- a compact nuclear island
- ability to operate in “island mode”
- black start

SDAA Part 2, Section 1.2.3, “Plant Features of Special Interest,” states that the NuScale Power Plant design minimizes human error through fail-safe design functionality, allows multimodular control capability from a single control room with effective automation design, employs digital display design and soft control technology to enhance usability, and considers minimization of operator workload. The applicant further stated that the NuScale human factors engineering program leverages human performance and operating experience from nuclear and nonnuclear industries.

The following is a general description of the NuScale SMR design. Subsequent chapters of this FSER provide detailed descriptions and evaluations of the individual systems that make up the NuScale SMR design.

1.2.2.1 Combined License Information

SDAA Part 2, Section 1.1.1, “Plant Location,” states that the NuScale Power Plant is designed to be located on a site with characteristics (e.g., seismology, hydrology, meteorology, geology, and other site-related characteristics) bounded by the site parameters described in SDAA Part 2, Chapter 2. COL Item 1.1-1 in SDAA Part 2, Section 1.1.1, states, “A COL applicant that references the NuScale Power Plant US460 standard design will identify the site-specific plant location.” The staff finds this COL item acceptable because it supports the COL applicant’s compliance with 10 CFR 52.79(a)(1).

1.2.2.2 Principal Design Criteria

SDAA Part 2, Section 1.2.1.1.1, “Principal Design Criteria,” states that the design provides a simple, safe reactor and provides the following:

- reliable, passive safety systems that do not rely on electrical power to fulfill their safety functions
- the ability to cope with DBEs without the need for operator actions
- safety features that assure a core damage frequency significantly lower than the current light-water reactor fleet
- the absence of RPV or containment penetrations below the top of the reactor core
- modularization to enable in-shop fabrication of reactor and containment components

1.2.2.3 Operating Characteristics

SDAA Part 2, Section 1.2.1.1.2, “Operating Characteristics,” states that the NPM is designed to operate up to full-power conditions using natural circulation as the means of providing reactor coolant flow, eliminating the need for reactor coolant pumps.

The NPMs are partially immersed in a reactor pool and protected by passive safety systems. Each NPM has a dedicated emergency core cooling system (ECCS) and decay heat removal system (DHRS).

Important features of the NPM include the following:

- a small, modular design

- an integral PWR NSSS that combines the reactor core, SGs, and pressurizer within the RPV, eliminating the need for external piping to connect the SGs and pressurizer to the RPV

- natural circulation that provides the driving force for reactor coolant flow, eliminating the need for reactor coolant pumps

- an RPV housed in a steel containment partially immersed in water, providing an effective passive heat sink for long-term decay heat removal

- a containment operated at a vacuum, eliminating the need for insulation on the RPV

- passive safety systems that are not reliant on electrical power

SDAA Part 2, Table 1.2-1, “Overall Characteristics of a NuScale Power Plant,” presents the overall characteristics of the NuScale Power Plant.

1.2.2.3.1 Nuclear Steam Supply System

The NSSS consists of a reactor core, two helical-coil SGs, and a pressurizer integrated within the RPV. The RPV is enclosed in a cylindrical CNV that is partially immersed in the reactor pool. The reactor core is located below the helical-coil SGs inside the RPV. Using natural circulation, the primary reactor coolant flowpath is upward through the central hot leg riser, and then downward around the outside of the SG tubes with return flow to the bottom of the core through an annular downcomer. As the reactor coolant flows across the SG tubes, heat is transferred to the secondary-side fluid inside the SG tubes. Concurrently, as the secondary-side fluid progresses up through the inside of the SG tubes, it is heated, boiled, and superheated to produce high-pressure steam for the turbine generator unit.

1.2.2.3.2 Reactor Core

The core configuration for the NPM consists of 37 fuel assemblies and 16 control rod assemblies (CRAs). The CRAs are organized into two banks: a regulating bank and a shutdown bank. The regulating bank is used during normal plant operation to control reactivity. The shutdown bank is used during normal shutdown. All 16 CRAs are inserted for scram events.

The fuel assembly design is similar to existing 17x17 designs that are used successfully in PWRs in the industry. The only significant difference is the fuel assembly length. The fuel is uranium dioxide (UO_2), with gadolinium oxide (Gd_2O_3) that may be used within the fuel assemblies when needed to support the core design. SDAA Part 2, Table 4.2-1, “Fuel Assembly Materials,” and Table 4.2-2, “Fuel Design Parameters” list fuel material and design parameters.

1.2.2.3.3 Pressurizer

The pressurizer provides the primary means for controlling RCS pressure. It is designed to maintain a stable reactor coolant pressure during operation. Reactor coolant pressure is increased by applying power to a pair of heater bundles installed above the pressurizer baffle plate. Pressure in the RCS is reduced using spray provided by the chemical and volume control system (CVCS).

1.2.2.3.4 Steam Generator

Each NPM uses two once-through, helical-coil SGs for steam production. The SGs are in the annular space between the hot leg riser and the RPV inside diameter wall. The SG consists of tubes connected to feed and steam plenums with tube sheets. Preheated feedwater enters the lower feed plenum through nozzles on the RPV. As feedwater flows through the interior of the SG tubes, heat is transferred across the SG tube wall from the reactor coolant to the feedwater. The feedwater changes phase and exits the SG as superheated steam.

1.2.2.3.5 Reactor Pressure Vessel

The RPV is an approximately cylindrical pressure vessel with an inside diameter ranging between 8 and 9 feet. The RPV comprises a lower head and lower shell section, a mating flange, an upper shell section, and an upper head. The lower head is torispherical, while the upper head is a truncated torisphere that includes a flat sub-assembly which supports the control rod drive mechanism support structure.

The RPV upper head includes penetrations for the pressurizer spray lines, the pressurizer high point vent line, reactor vent valves, reactor safety valves, and in-core instrumentation. The RPV upper shell houses the SGs, upper vessel internals, and a portion of the pressurizer region (including the pressurizer heater penetrations, heaters, and baffle plate). The upper shell also includes the feedwater and steam plenums, penetrations for the RCS injection and discharge lines, and penetrations for the reactor recirculation valves.

The RPV lower shell houses the vessel internals core support assembly and the reactor core. The lower shell is bolted to the upper shell by a mating flange, and the mating flange allows removal of the lower shell to support refueling operations. There are no penetrations in the lower section of the RPV.

1.2.2.3.6 Containment Vessel

The CNV is a cylindrical, steel pressure vessel housing the RPV, control rod drive mechanisms, and associated NSSS piping and components.

The safety functions of the CNV are to

- provide a barrier to contain mass, energy, and fission product release by closure of the containment isolation valves (CIVs) upon containment isolation signal.
- provide a sealed containment and thermal conduction for the condensation of steam that provides makeup water to the RCS.
- transfer core heat from the reactor coolant in containment to the UHS during ECCS operation.
- provide structural support for safety-related structures, systems, and components.
- provide electrical penetration assemblies for safety-related reactor instrumentation cables through the CNV.
- provide the required pressure boundary for DHRS operation.

Following an actuation of the ECCS, steam is vented from the RPV through the RVVs. This results in an initial spike in containment pressure and temperature. Steam in contact with the inside surface of the CNV is passively cooled and condensed by conduction and convection to the reactor pool water.

1.2.2.4 Safety Considerations

NuScale states that the integral design of the NPM eliminates external coolant loop piping, which eliminates large-break loss-of-coolant accident (LOCA) scenarios. The availability of passive safety systems for decay heat removal, emergency core cooling, and control room habitability eliminates the need for external power under accident conditions. With these passive safety systems, small-break LOCAs also do not challenge the safety of the plant. The result is a design with a core damage frequency that is lower than the current light-water reactor fleet.

SDAA Part 2, Table 1.2-2, "Design Features of a NuScale Power Module," lists some of the features of the NPM.

1.2.3 Engineered Safety Features and Emergency Systems

1.2.3.1 Engineered Safety Feature Materials

SDAA Part 2, Section 6.1, "Engineered Safety Feature Materials," provides details related to the selection and fabrication methods for metallic and organic materials used in engineered safety feature (ESF) components to ensure compatibility with fluids to which the component may be exposed during normal, accident, maintenance, and testing conditions.

1.2.3.2 Containment Systems

The containment system is an integral part of the NPM and provides primary containment for the RCS. SDAA Part 2, Section 6.2, "Containment Systems," provides further information on the containment system.

1.2.3.3 Emergency Core Cooling System

The ECCS provides a passive means of decay heat removal in the event of a LOCA. The ECCS consists of two RVVs mounted on the upper head of the RPV, two reactor recirculation valves (RRVs) mounted on the side of the RPV, and associated actuators located on the upper CNV.

The RVVs and RRVs are closed during normal plant operation and open to actuate the system during applicable accident conditions. When actuated, the RVVs vent steam from the RPV into the CNV, where the steam condenses, and liquid condensate collects in the bottom of the containment. The RRVs allow the accumulated coolant to reenter the RPV for recirculation and cooling of the reactor core. Placement of the RRV penetrations on the side of the RPV is such that when the system actuates, the coolant level in the RPV remains above the core and the fuel remains covered. The cooling function of the ECCS is entirely passive, with heat conducted through the CNV wall to the reactor pool. SDAA Part 2, Section 6.3, "Emergency Core Cooling System," provides design and operational information for the ECCS.

1.2.3.4 Control Room Habitability System

The control room habitability system (CRHS) ensures that plant operators are adequately protected against the effects of accidental releases of radioactive gases. The CRHS is a passive system that provides clean, compressed, breathable air to the main control room (MCR) in the event of a radioactive release or when AC power is not available. Areas served by the

CRHS are maintained at positive pressure relative to adjacent areas. Compressed breathable air storage capacity can provide clean air to the MCR spaces for at least 72 hours following an initiating event. SDAA Part 2, Section 6.4, “Control Room Habitability,” provides design and operational information for the CRHS.

1.2.3.5 Fission Product Removal and Control Systems

The only fission product removal and control system credited in the design is the CNV in conjunction with the containment isolation system. Fission product control is inherent in the design of the NPM, wherein the CNV atmosphere is depleted through the passive process of aerosol deposition. SDAA Part 2, Section 6.5, “Fission Product Removal and Control Systems,” provides information for this ESF.

1.2.3.6 Inservice Inspection of Class 2 and Class 3 Components

The inservice inspection program includes the preservice examinations and the periodic inservice inspections and tests necessary to ensure that safety-related and risk significant SSCs are capable of fulfilling their intended safety functions. SDAA Part 2, Section 6.6, “Inservice Inspection and Testing of Class 2 and 3 Systems and Components,” provides detailed information for the inservice inspection program.

1.2.4 Instrumentation, Controls, and Electrical Systems

The I&C architectural design philosophy incorporates clear interconnection interfaces, separation between safety and nonsafety systems, and simplification of system functions. The I&C architecture primarily consists of the following systems, which are described in SDAA Part 2, Section 7.0, “Instrumentation and Controls—Introduction and Overview”:

- module protection system (MPS)—provides information from safety-related sensors monitoring temperature, flow, neutron flux, and pressure data on the NSSS
- neutron monitoring system—measures neutron flux as an indication of core power and provides safety inputs to the MPS
- module control system—a distributed control system that allows monitoring and control of module-specific plant components
- plant control system—supplies nonsafety inputs to the human system interfaces in the MCR and in other locations where necessary
- fixed area radiation monitoring system—continuously monitors in-plant radiation and airborne radioactivity levels
- safety display and indication system—provides visual display and indication in the MCR from the MPS and plant protection system (PPS)
- PPS—monitors and controls systems that are common to all NPMs and are not specific to an individual NPM
- in-core instrumentation system—monitors various parameters within the reactor core and RCS and sends the parameter values to the module control system for display and evaluation

Under normal operating conditions, the AC electrical power distribution system supplies continuous power to equipment required for plant startup, normal operation, and shutdown. SDAA Part 2, Section 8.3, “Onsite Power Systems,” states that the NuScale Power Plant does not require onsite or offsite AC electrical power to cope with design-basis events (DBEs). Safety systems are not reliant on AC or DC electrical power for actuation.

The applicant described the power systems within the plant as follows:

- The high voltage AC electrical distribution system provides power from the turbine generators and the backup power supply system to high voltage AC buses and contains the switchyard.
- The medium voltage AC electrical distribution system provides power to buses servicing medium voltage loads.
- The low voltage AC electrical distribution system provides power to buses servicing low voltage loads.
- The augmented DC power system provides DC power to select plant loads.
- The normal DC Power System provides power to nonsafety-related loads.
- The backup power supply system provides backup power for plant loads.

1.2.5 Steam Power Conversion System

The steam and power conversion system associated with an NPM primarily consists of the main steam system, the turbine generator system, the air-cooled condenser system, and the condensate and feedwater system as shown in Part 2, Figure 1.2-2. Part 2, Chapter 10 provides design and operational information for the steam and power conversion system.

1.2.6 Fuel Handling and Storage Systems

The fuel handling and reactor maintenance areas are located in the RXB and include space for the spent fuel pool (SFP), refueling pool, and dry dock. SDAA Part 2, Figure 1.2-8, “Reactor Building 25'-0" Elevation,” shows the pools.

1.2.7 Plant Cooling Water Systems

The plant cooling water systems include several nonsafety-related systems that are important to supporting plant operation. These systems include the following:

- The reactor component cooling water system is a nonsafety-related, closed-loop cooling system that transfers heat from various plant components to the site cooling water system. (SDAA Part 2, Section 9.2.2, “Reactor Component Cooling Water System”).
- The pool cooling and cleanup system contains subsystems that provide water level control and temperature control for the interconnected reactor pool, refueling pool, and SFP (SDAA Part 2, Section 9.1.3, “Pool Cooling and Cleanup System”).

- The chilled water system provides cooling for heating ventilation and air conditioning and radioactive waste equipment (SDAA Part 2, (SDAA Part 2, Section 9.2.8, “Chilled Water System”).

- The site cooling water system is a two-loop system comprising a closed-loop subsystem that interfaces with plant loads, and an open-loop system that rejects heat to the environment via cooling towers (see SDAA Part 2, Section 9.2.7, “Site Cooling Water System”).

1.2.8 Radioactive Waste Management System

SDAA Part 2, Chapter 11, “Radioactive Waste Management,” discusses the radioactive waste management system in detail. SDAA Part 2, Section 11.2, “Liquid Waste Management System,” Section 11.3, “Gaseous Waste Management System,” and Section 11.4, “Solid Waste Management System,” discuss in detail liquid, gaseous, and solid radioactive waste management systems, respectively. SDAA Part 2, Section 11.5, “Process and Effluent Radiation Monitoring Instrumentation and Sampling System,” discusses process effluent radiation monitoring and sampling systems. FSER Chapter 11, “Radioactive Waste Management,” documents the NRC staff evaluation of the applicant’s radioactive waste management system.

1.3 General Arrangement of Major Structures and Equipment

SDAA Part 2, Figure 1.2-1, “Conceptual Site Layout,” presents the layout of a NuScale Power Plant. SDAA Part 2, Section 1.2.2, “General Arrangement of Major Structures and Equipment,” describes the following structures and equipment:

- Reactor Building (RXB) - SDAA Part 2, Figure 1.2-3 and Figure 1.2-8 through Figure 1.2-17 are RXB drawings. The RXB houses the NPMs and systems and components required for plant operation and shutdown. It is designed with considerations for the effects of aircraft impact, environmental conditions, postulated DBEs (internal and external), and design-basis threats. The RXB also provides radiation protection to plant operations and maintenance personnel.

- Fuel handling and reactor maintenance areas - The fuel handling and reactor maintenance areas are located in the RXB and include space for the SFP, refueling pool, and dry dock as shown in SDAA Part 2, Figure 1.2-8. Space is provided for the operation of fuel handling equipment and accessing the upper portion of an NPM while the reactor core is being refueled.

  The refueling pool is connected directly to the reactor pool, accommodating transport of an NPM through the pool water using the RXB crane. A weir between the refueling pool and SFP provides access for fuel assembly transport under water during the refueling process. The fuel handling and maintenance areas are designed to provide radiation protection for personnel in those areas.

  A fuel receiving area and a jib crane for loading new fuel assemblies into the new fuel elevator are located near the SFP. The area provides equipment access to aid in new fuel receiving activities. Upon receipt, new fuel assemblies are inspected and temporarily stored in racks in the SFP before being placed in a reactor core. The SFP provides storage space for the accumulated spent fuel assemblies before removal for dry storage and for temporary short-term storage for new fuel assemblies.

- Control Building (CRB)—SDAA Part 2, Figures 1.2-18 through 1.2-21 show the CRB. The MCR and the technical support center are located in the in the CRB. Additional equipment located in the CRB includes the control room heating ventilating and air conditioning (HVAC) system equipment, the chilled water system equipment supporting the control room HVAC system, and an elevator machine room.

  – The MCR contains control panels for all installed NPMs. Monitoring and control of multiple NPMs can be performed from a control room panel. Digital control systems are implemented in a manner that provides independence between safety-related protection systems and nonsafety-related control systems. Each reactor control system display provides the monitoring for a specific reactor. Additional display stations, including a separate display for shared plant systems, provide control room operators with access to a wide range of plant information for trending and diagnostics.

    The MCR contains alarms, displays, and controls for monitoring and control by the operators. The control room supervisor station provides an overview of all NPMs using multiple monitors. Monitor displays are designed using human factors analysis to enhance simplicity. The display layout and design provides graphical representations of plant systems and components. The following monitoring and control activities are typical control room functions:

    - initiate NPM startup
    - initiate NPM shutdown
    - set or correct selected setpoints that control the NPM or plant functions
    - take corrective actions if an NPM or plant system does not operate as intended.

  – A technical support center is provided, compliant with the design requirements of NUREG-0696, "Functional Criteria for Emergency Response Facilities," issued February 1981 (ML051390358). SDAA Part 2, Section 13.3, "Emergency Planning," provides additional information.

- Radioactive Waste Building (RWB)—The RWB houses equipment and systems for processing radioactive gaseous, liquid, and solid waste and for preparing waste for offsite shipment (see SDAA Part 2, Figures 1.2-22 through 1.2-28 for RWB drawings). The RWB contains HVAC equipment for high-efficiency particulate air filtration of air from the RXB and RWB.

- Major Systems—

  – The DHRS provides secondary-side reactor cooling for non-LOCA events when normal feedwater is not available. The system, as shown in SDAA Part 2, Figure 1.2-6, "Decay Heat Removal System," is a closed-loop, two-phase natural circulation cooling system. Two trains of decay heat removal equipment are provided, one attached to each SG loop. Each train is capable of removing 100 percent of the decay heat load and cooling the RCS. Each train has a passive condenser immersed in the reactor pool. In the event of an SG tube failure, the affected SG is isolated and the DHRS provides cooling through the intact SG.

– The ultimate heat sink (UHS) is a pool of water located in the RXB below plant grade level. The UHS consists of the reactor pool area, the refueling pool area, and the SFP area. SDAA Part 2, Figure 1.2-8 shows the pool areas. During normal plant operations, heat is removed from the pool through the pool cooling and cleanup system. In a design-basis accident involving a sustained loss of all AC power, decay heat is removed from the NPMs through passive heat transfer to the pool, resulting in pool heatup and boiling. Water inventory in the reactor pool is adequate to cool the NPMs for at least 72 hours without adding water.

– During normal operation, the CVCS recirculates a portion of the reactor coolant through demineralizers and filters to maintain reactor coolant cleanliness and chemistry. A portion of the recirculated coolant is used to supply pressurizer spray for controlling reactor pressure. Reactor coolant inventory is controlled by injection of additional water when reactor coolant levels are low or during letdown of reactor coolant to the liquid radioactive waste system when coolant inventory is high. Additionally, during the NPM startup process, the CVCS is used in conjunction with the module heatup system to add heat to the reactor coolant to establish natural circulation flow in the RCS. SDAA Part 2, Section 9.3.4, "Chemical and Volume Control System," provides CVCS design and operational information.

1.4 Description of Site, Plant, and Facility

1.4.1 Site Description

SDAA Part 2, Section 1.1.1, states that the NuScale Power Plant is designed to be located on a site with site characteristics (e.g., seismology, hydrology, meteorology, geology, and other site-related characteristics) bounded by the site parameters described in SDAA Part 2, Chapter 2.

SDAA Part 2, Section 1.1.1, COL Item 1.1-1, states that a COL applicant that references the NuScale Power Plant US460 standard design will identify the site-specific plant location.

1.4.2 General Plant Description

SDAA Part 2, Section 1.2, provides a general description of the overall facility, which includes principal design criteria, operating characteristics, and safety considerations; ESFs and emergency systems; I&C and electrical systems; power conversion system; fuel, fuel handling, and storage systems; plant cooling water systems; radioactive waste management systems.

NuScale stated that each COL applicant will develop an FSAR that incorporates by reference SDAA Part 2. SDAA Part 2 includes COL items that identify where site-specific information must be provided. The scopes of the standard design and site-specific design are shown in SDAA Part 2, Figure 1.2-1. The basic systems associated with power generation are shown in SDAA Part 2, Figure 1.2-2. NuScale has delineated security-related information using double braces {{ }}. This information is withheld in accordance with 10 CFR 2.390(d)(1).

1.4.2.1 Principal Characteristics of a NuScale Plant Site

SDAA Part 2, Figure 1.2-1, "Conceptual Site Layout," presents a conceptual layout of the overall site. The majority of the site buildings are located within the protected area and surrounded by a double fence and intrusion-detection equipment. The protected area is located within the security owner-controlled area, surrounded by an additional single fence. An administration building, training building, and a warehouse are inside the security owner-controlled area fence.

The NuScale Power Plant is designed for 1 to 6 NPMs with the associated primary and secondary systems and components necessary to produce power and maintain the facility. This includes main steam systems, turbine generator sets, condensate and feedwater systems (shown in SDAA Part 2, Figure 1.2-2), as well as module assembly equipment, fuel handling equipment, turbine maintenance equipment, and radioactive waste processing equipment.

The NuScale standard plant design includes the following structures (see SDAA Part 2, Figures 1.2-1 and 1.2-2):

- RXB—located above and below grade, houses the following facilities, and is described in further detail in SDAA Part 2, Section 1.2.2.1, "Reactor Building":

  - UHS (reactor pool, refueling pool, and SFP)
  - fuel handling areas
  - primary systems

- CRB—located adjacent to the RXB, provides space for the following facilities and is described further in SDAA Part 2, Section 1.2.2.2, "Control Building":

  - MCR—houses the equipment, controls, and indications for operation of the NPMs

  - Technical support center—located outside the radiologically controlled area, provides space to support emergency operations and personnel

- RWB—provides space for HVAC equipment and radioactive waste treatment and storage equipment, and is described further in SDAA Part 2, Section 1.2.2.3, "Radioactive Waste Building"

Other site structures include the Turbine Generator Building, Annex Building, Security Buildings, Central Utility Building, and Diesel Generator Buildings.

1.4.3 Facility Description

SDAA Part 2, Section 1.2.1.1, "Facility Description," states that the reactor core transfers heat into the reactor coolant, which flows upward through the core and lower and upper riser assemblies. The reactor core transfers heat into the reactor coolant, and the heated reactor coolant flows upward through the core and lower and upper riser assemblies. The heated coolant exits the upper riser assembly and is redirected downward, into the SG region between the vessel wall and the upper riser assembly. As the reactor coolant transfers heat to the SGs, it cools and becomes denser, which drives the natural circulation flow. The coolant returns to the bottom of the vessel through the downcomer and back into the reactor core, where the cycle begins again, as shown in SDAA Part 2, Figure 1.2-5.

On the secondary side, preheated feedwater is pumped into the tube side of the SGs where it boils. As the steam flows upward in the tubes, it is continually heated to produce superheated steam before exiting the top of the SGs.

The superheated steam is directed to a dedicated steam turbine. A generator, driven by the turbine, creates electric power that is delivered to a microgrid, utility grid, or service load. Steam that exits or bypasses the turbine is directed to the air-cooled condenser, which removes heat and condenses the steam. The condensate is pumped through condensate polishing equipment to the inlet of the feedwater pumps. A small amount of steam is extracted from turbine stages to preheat the feedwater and increase plant efficiency.

1.4.4 Multimodule Design Considerations

SDAA Part 2, Section 1.1.4, “Multi-Module Design Considerations,” describes Compliance with GDC 5 of 10 CFR Part 50, Appendix A and Multi-Module Considerations During Phased Construction and Startup.

Compliance with GDC 5 of 10 CFR Part 50, Appendix A - The Applicant states that the design complies with GDC 5. Other than the UHS, safety-related systems are functionally independent and are not shared among NPMs. The UHS is designed to perform its required safety-related functions during a DBE in one NPM and a controlled shutdown of the remaining NPMs.

Portions of the RXB and Control Building (CRB) are Seismic Category I structures that contain and support safety-related systems for multiple NPMs. The RXB and CRB structures are not adversely impacted during DBEs. Operation of the nonsafety-related shared systems, including credible failures of these systems during DBEs, does not adversely affect safety-related NPM functions.

The operating configurations include from 1 to 6 NPMs in the operating modes permitted by the technical specifications. Operating configurations are considered in the design and no restrictions are required to ensure plant safety. The shared systems have been evaluated for interface requirements and system interactions. The plant design provides protection of safety systems in the event of failures in shared systems and the performance of safety-related functions is ensured during DBEs.

Multi-Module Considerations During Phased Construction and Startup -

The design relies on passive safety-related systems that are module-specific and a shared safety-related UHS. With the exception of the UHS, the shared systems are nonsafety-related and non-risk significant, and shared system interactions do not result in a loss of NPM safety-related functions. Construction and phased expansion of NPMs do not result in operating configurations that are materially different than that assumed in the safety analysis and the independence of NPM safety-related systems is maintained.

The analysis of shared system interactions described in SDAA Part 2, Section 19.1 applies to the operating NPMs during installation of subsequent NPMs. Consequently, restrictions in operating configurations or interface requirements are not necessary to ensure the safe operation of operating NPMs during installation, testing, or startup of subsequent NPMs.

1.5 Comparison with Other Facilities

SDAA Part 2, Table 1.3-1, “NuScale Plant Comparison with Other Facilities,” provides the major NuScale Power Plant US460 standard design features and nominal parameters; the associated SDAA Part 2 sections further discuss the design features and parameters. These NuScale features and values are compared with a typical PWR plant design. All values are nominal and are provided for comparison only. The typical PWR values presented are representative of the standardized nuclear unit power plant system design. SDAA Part 2, Table 1.3-2, “Safety Systems and Components Required to Protect the Reactor Core - NuScale Comparison with Other Facilities,” compares the safety systems and components required to protect the reactor core of the NuScale Power Plant versus those for a typical PWR plant.

1.5.1 Nuclear Power Plants to Be Operated on Multiunit Sites

In SDAA Part 2, Section 1.10, “Sites With Multiple Nuclear Power Plants,” NuScale provides COL Item 1.10-1, directing a COL applicant that references the NuScale Power Plant US460 standard design will evaluate the potential hazards resulting from construction activities of the new NuScale facility to an operating nuclear power plant on a co-located site per 10 CFR 52.79(a)(31).

1.6 Identification of Agents and Contractors

SDAA Part 2, Section 1.4, “Identification of Agents and Contractors,” states that NuScale Power, LLC (NuScale), has the overall design responsibility for the NuScale Power Plant US460 standard design. In SDAA Part 2, Section 1.4.1, “Principal Consultants,” NuScale identifies Fluor Corporation (Fluor) that provided the balance of plant design described in the Standard Design Approval Application. In COL Item 1.4-1, NuScale directs an applicant that references the NuScale Power Plant US460 standard design to identify the prime agents or contractors for the construction and operation of the nuclear power plant.

1.7 Requirements for Additional Technical Information

SDAA Part 2, Section 1.5, “Requirements for Additional Technical Information,” describes the verification and confirmation tests of unique design features that support the safety analysis for the NuScale Power Plant. NuScale states that the testing program described in SDAA Part 2, Section 1.5, was developed to provide data to support the final safety analyses.

1.7.1 NuScale Testing Programs

1.7.1.1 Critical Heat Flux Testing—Preliminary Fuel Design

SDAA Part 2, Section 1.5.1.1, summarizes the critical heat flux (CHF) testing for the preliminary fuel design. NuScale states that the NPM employs a fuel design for heat generation that is similar to a standard PWR, with the exception of the fuel assembly height and the reactor coolant driving force. The NuScale fuel is approximately half the height of standard PWR fuel and features low-flow natural circulation of primary coolant rather than pump-driven primary coolant flow. In order to meet fuel licensing requirements, two CHF test programs were conducted: (1) a test program for the preliminary fuel design (described in this section), and (2) a second test program for the final fuel design (described in SDAA Part 2, Section 1.5.1.2).

NuScale stated that testing for the preliminary fuel design CHF test series was performed at Stern Laboratories in Ontario, Canada using their existing high-pressure flow loop, electrical power supplies, heat rejection devices, water conditioning system, and data acquisition system. The test section and fuel simulators were designed and manufactured to represent the preliminary NuScale fuel design. Fuel assembly simulators with different axial power shapes were manufactured along with two sets of spacer grids that allowed testing of a 5x5 fuel assembly simulation bundle with or without the center fuel rod replaced by a guide tube. The approach of using a 5x5 representation of the larger 17x17 fuel assembly is an industry-accepted practice for CHF testing of PWR fuel bundles. The fuel assembly simulation bundles were mounted within a square flow channel and installed in the instrumented test section. This allowed testing of three separate configurations of fuel assembly simulation bundles with different axial flux shapes and fuel assembly subchannels as described in NuScale Power Critical Heat Flux Correlations topical report LTR-00116-21012-A, "NuScale Power Critical Heat Flux Correlations," Revision 1-A (ML18360A633).

NuScale stated that tests were performed for a variety of thermal conditions using representative 5x5 fuel assembly simulations with a 2-m (6.56-ft) heated length, differing axial power profiles, with and without a simulated guide tube. The testing investigated the effects of shorter fuel length and low-flow natural circulation of the primary coolant.

1.7.1.2 Critical Heat Flux Testing—NuFuel HTP2™ Fuel Design

SDAA Part 2, Section 1.5.1.2, summarizes the CHF testing for the NuFuel HTP2™ fuel design. The primary objective for this test program was to obtain CHF data for the NuScale fuel design that employs AREVA HMP™/HTP™ spacer grid technology (designated as NuFuel HTP2™) to augment the existing database that was previously obtained for NuScale's preliminary fuel design (described in SDAA Part 2, Section 1.5.1.1). In addition, this test allowed NuScale to obtain bundle subchannel exit temperatures to determine mixing coefficients and to collect single-phase and two-phase pressure-drop characteristics of the assembly for a range of bundle powers and hydraulic conditions.

NuScale stated that the CHF test employed an electrically heated test section that consisted of a 5x5 simulated fuel bundle built to prototypic geometry and employed AREVA HTP™/HMP™ grid technology. The fuel assembly simulators with different power shapes were tested using a 5x5 fuel bundle with and without the center fuel rod replaced by a guide tube. The testing was conducted by flowing water through the test section at specified flow rates over a range of hydraulic conditions of the NPM. At each test point, the loop was configured for a specified flow rate, inlet temperature, and exit pressure conditions, and the bundle power was increased until CHF was detected over a range of operating conditions and axial power shapes for vertical 5x5 fuel assembly configurations. The occurrence of CHF was indicated by an excursion of the fuel simulator temperatures.

1.7.1.3 Steam Generator Thermal-Hydraulic Performance Testing—Electrically Heated Facility

SDAA Part 2, Section 1.5.1.3, summarizes the SG thermal-hydraulic performance testing for an electrically heated facility. NuScale stated that the NPM incorporates two collocated SGs housed within the RPV. The SGs provide heat transfer to and from the primary system for both normal and off-normal conditions. Through natural circulation, the RCS transfers the core power to the SG converting feedwater into steam. Unlike current PWR designs, the reactor coolant flows around the outside of the SG tubes (primary side), and the feedwater and main steam flow through the inside of the tubes (secondary side). Because these design aspects of the helical SGs are different from those used in the nuclear fleet, operational experience is not available, and large-scale experimental data were needed for validation of NuScale's thermal-hydraulic systems and design computer codes, as well as determination of SG performance characteristics.

Types of testing carried out included adiabatic testing, diabatic testing, transient testing, and density wave oscillation testing. The objective of this testing was to determine the secondary-side (inside tube) thermal-hydraulic performance of individual helical tubes representative of those used in the NPM SG design.

Dynamic pressure measurements were recorded during test runs, which supported the development of power spectral density spectra that may be used to support the evaluation of the potential for internal two-phase (boiling) pressure fluctuations to contribute to flow-induced vibration of SG tubes.

1.7.1.4 Steam Generator Thermal-Hydraulic Performance Testing—Fluid-Heated Facility

SDAA Part 2, Section 1.5.1.4, summarizes SG thermal-hydraulic performance testing for a fluid heated facility. This set of SG tests was conducted using a 252-tube bundle array that was fluid heated on the exterior of the tubes to more accurately represent primary side SG conditions.

The test facility included a large pressure vessel, which was able to accommodate the tube bundle test section and allowed for testing at elevated pressures and temperatures. Types of testing carried out included adiabatic, diabatic, transient, density wave oscillation, and fluid-elastic instability tests. In these tests, thermocouples, pressure transducers, mass flow rate instruments, and strain gauges were used to collect temperature, pressure, flow rate, and vibration data at several locations on the primary and secondary sides of the SG. These data have been used to benchmark NuScale thermal-hydraulic design and systems computer codes and to define steam outlet conditions as a function of primary-fluid heating and secondary-side conditions.

1.7.1.5 NuScale Integral System Test Program

SDAA Part 2, Section 1.5.1.5, summarizes the NuScale integral system test program. NuScale stated that the purpose of the NuScale integral system test program was to generate thermal-hydraulic data for system characterization and safety code validation using a scaled representation of the NPM design.

NuScale stated that the NuScale Integral System Test Facility (NIST) allows NuScale to replicate the integrated thermal-hydraulic phenomena occurring in the RCS, containment, safety systems, and reactor pool. Data collected provide system characterization data required for the validation of safety-related software NRELAP5 code. NRLAP5, which is based on a commercial version of RELAP5-3D, is a thermal-hydraulic analysis code developed at NuScale for use in the thermal-hydraulic design, safety analysis, and licensing of the NPM.

The NIST is a scaled representation of the NPM reactor, containment, and reactor pool systems. The NIST volumes, lengths, and areas are obtained by multiplying the respective NPM design volumes, lengths, active heat transfer areas, and flow areas by the applicable scale factors determined through a detailed scaling analysis. This process ensures the NIST properly captures the important thermal-hydraulic phenomena and processes that would occur in the plant.

NuScale stated that the following tests have been completed at the NIST, located on the Oregon State University campus in Corvallis, OR:

- facility characterization tests used to develop the NRELAP5 model of the NIST-1
- LOCA tests used to validate NRELAP5 for LOCA and containment analyses
- non-LOCA (AOO) tests used to validate NRELAP5 for non-LOCA analyses
- long-term cooling tests used to validate NRELAP5 for extended passive cooling analyses

Data obtained from the NIST tests identified above have been used to successfully validate the NRELAP5 code for LOCA and containment peak pressure, non-LOCA, and extended passive cooling applications.

1.7.1.6 Control Rod Assembly Drop and Control Rod Drive Shaft Alignment Test

The NPM is designed with control rod drive shafts that are longer than in conventional PWR designs and have the capability to be remotely disconnected. The control rod drive shafts are aligned using the following multiple-support features:

- control rod drive mechanism penetrations in the reactor vessel head
- integrated steam plenum
- upper control rod drive shaft supports in the upper riser section
- a control rod drive shaft alignment cone located at the top of the CRA guide tube

The design uses a CRA and fuel-assembly design similar to, but shorter than, that of traditional operating reactors.

Testing was performed at the Framatome Technical Center in Erlangen, Germany, and was configured as an ambient pressure and temperature test. The ambient test configuration used a full-length control rod drive shaft coupled with an NPM CRA and fuel assembly, as well as the control rod drive shaft support structures and a CRA guide tube assembly. The CRA guide tube assembly and fuel assembly were immersed in the water under ambient conditions with no coolant flow.

Test results confirmed the operability of the control rod drive shafts for a range of potential component conditions and distortions. Test results also confirmed CRA drop time and CRA impact force at end of drop.

1.7.1.7 Emergency Core Cooling System Valve Demonstration Testing

SDAA Part 2, Section 1.5.1.7 describes the NPM design ECCS valves testing program. The RVVs are located on the reactor vessel pressurizer, and the reactor recirculation valves (RRVs) are located on the reactor vessel downcomer. The RVVs and RRVs are functionally similar in design; however, the RVVs are larger than the RRVs. Each of the ECCS valves is pilot operated by remote-mounted trip and reset solenoid valves located at the reactor containment boundary. An inadvertent actuation block (IAB) feature is included in the system, located inside containment on each RRV main valve. Test programs were developed to demonstrate and evaluate functional performance of the ECCS valve system design including the unique aspects of the design, such as the configuration of the ECCS valve components and the IAB feature.

1.7.1.8 Emergency Core Cooling System Supplemental Boron Dissolution Testing

SDAA Part 2, Section 1.5.1.8 describes the test program developed for the ECCS supplemental boron (ESB) system. The ESB system provides boron to the recirculating coolant in containment during ECCS actuation. The boron addition is provided to add negative reactivity to keep the reactor subcritical during some DBEs. This boron addition is achieved passively by dissolving solid boron oxide staged in two supplemental boron dissolvers located on the inside of the CNV wall. During ECCS actuation a portion of condensate generated on the CNV inner wall is captured by condensate channels and redirected through the dissolver. The dissolution of the boron oxide creates a boric acid and water solution that is added to the recirculating coolant.

NuScale developed a test program to measure fundamental characteristics of boron oxide dissolution. This information was used to determine conservative assumptions for the ESB boron oxide dissolution rate which is used as part of the boron transport methodology. The boron transport methodology demonstrates that sufficient boron is transported back to the RPV to meet subcriticality requirements.

1.7.1.9 NuScale Test and Inspection Plans

SDAA Part 2, Section 14.2, "Initial Plant Test Program," provides information on NuScale's test and inspection plans related to plant startup testing.

1.8 Material Referenced

SDAA Part 2, Table 1.6-1, "NuScale Topical Reports," and Table 1.6-2, "NuScale Technical Reports," identify sections of the topical report and technical report that are incorporated by reference as part of the NuScale Power Plant SDAA. Tables 1.8-1 and 1.8-2 below list the publicly available versions of these reports.

Table 1.8-1 NuScale Topical Reports

Topical Report Number	Topical Report Title	Specific Report Sections Incorporated by Reference	FSAR Section
TR-0920-71621-A	Building Design and Analysis Methodology for Safety-Related Structures, Revision 1-A	Section B, Sections 3.0 - 7.0	3.5, 3.7, 3.8, 3B
TR-0118-58005-A	Improvements in Frequency Domain Soil-Structure-Fluid Interaction Analysis, Revision 2-A	Section B, Sections 3.0 - 8.0	3.7, 3A
TR-0716-50351-A	NuScale applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces, Revision 1-A	Section B, Sections 3.0 - 5.0	4.2
TR-0616-48793-A	Nuclear Analysis Codes and Methods Qualification, Revision 1-A	Section B, Sections 2.0 - 8.0	4.3, SDAA Part 4
TR-0116-20825-A	Applicability of AREVA Fuel Methodology for the NuScale Design, Revision 1-A	All	4.2, 4.3, 4.4, Exemption 14
TR-108553-A	Framatome Fuel and Structural Response Methodologies Applicability to NuScale, Revision 0-A (Supplement 1 to TR-0116-20825-P-A, Revision 1; Supplement 1 to TR-0716-50351-P-A, Revision 1)	All	4.2, 4.3, 4.4, Exemption 14
TR-00116-21012-A	NuScale Power Critical Heat Flux Correlations, Revision 1-A	All	4.4, 15.0, 15.6

Table 1.8-1 NuScale Topical Reports (Continued)

Topical Report Number	Topical Report Title	Specific Report Sections Incorporated by Reference	FSAR Section
TR-107522	Applicability Range Extension of NSP4 Critical Heat Flux Correlation, Revision 1 (Supplement 1 to TR-0116-21012-P-A, Revision 1)	All	4.4, 15.0, SDAA Part 4
TR-0915-17564-A	Subchannel Analysis Methodology, Revision 2-A	All	4.3, 4.4, 15.0, 15.4, 15.6, SDAA Part 4
TR-108601	Statistical Subchannel Analysis Methodology, Revision 4 (Supplement 1 to TR-0915-17564-P-A, Revision 2)	All	4.3, 4.4, 15.0, 15.4, 15.6, SDAA Part 4
TR-1015-18653-A	Design of the Highly Integrated Protection System Platform, Revision 2-A	All	7.0, 7.1, 7.2, 15.8, Exemption 3
TR-0915-17565-A	Accident Source Term Methodology, Revision 4-A	Section B, Sections 3.0 - 5.0	2.3, 3.11, 3C, 12.2, 15.0, SDAA Part 4
TR-0516-49416	Non-Loss-of-Coolant Accident Analysis Methodology, Revision 4	All	5.2, 6.2, 7.1, 15.0, SDAA Part 4
TR-124587	Extended Passive Cooling and Reactivity Control Methodology, Revision 0	All	5.4, 6.2, 15.0, 15.6, SDAA Part 4
TR-0716-50350	Rod Ejection Accident Methodology, Revision 3	All	4.3, 15.0, 15.4
TR-0516-49422	Loss-of-Coolant Accident Evaluation Model, Revision 3	All	4.4, 5.2, 6.2, 6.3, 15.0, 15.6
TR-0516-49417-A	Evaluation Methodology for Stability Analysis of the NuScale Power Module, Revision 1-A	Section B, Sections 3.0 - 10.0	4.4, 5.4, 15.0, 15.9
TR-0420-69456-A	NuScale Control Room Staffing Plan, Revision 1-A	None	18.5, SDAA Part 4
TR-0515-13952-A	Risk Significance Determination, Revision 0-A	Section D, Section 3.0	17.4, 18.6, 19.1, 19.3
MN-122626-A	NuScale Power, LLC Quality Assurance Program Description, Rev.2	All	17.5

Table 1.8-2 NuScale Technical Reports

Technical Report Number	SDAA Technical Reports	Specific Report Sections Incorporated by Reference	FSAR Section
TR-121516, Revision 1	Containment Vessel Ultimate Pressure Integrity	All	3
TR-123242, Revision 1	Effluent Release (Gale Replacement) Methodology and Results	All	11
TR-118976, Revision 0	Fluence Calculation Methodology and Results	All	5 - (5.3), 4 - (4.3)
TR-130418, Revision 0	Human Factors Engineering Design Implementation Plan	None	18
TR-124333, Revision 0	Human Factors Engineering Functional Requirements Analysis and Function Allocation Implementation Plan	None	18
TR-130417, Revision 0	Human Factors Engineering Human System Interface Design Implementation Plan	None	18
TR-130409, Revision 0	Human Factors Engineering Operating Experience Review Implementation Plan	None	18
TR-130414, Revision 0	Human Factors Engineering Program Management Plan	None	18
TR-130412, Revision 0	Human Factors Engineering Staffing and Qualification Results Summary Report	None	18
TR-130413, Revision 0	Human Factors Engineering Task Analysis Implementation Plan, TR-130413	None	18
TR-130416, Revision 0	Human Factors Engineering Treatment of Important Human Actions Results Summary Report	None	18
TR-130415, Revision 0	Human Factors Engineering Verification and Validation Implementation Plan	None	18
TR-117605, Revision 0	NuFuel-HTP2™ Fuel and Control Rod Assembly Designs, TR-117605	All	4 - (4.2, 4.3)
TR-121353, Revision 2	NuScale Comprehensive Vibration Assessment Program Analysis Technical Report	All	3 - (3.9), 5- (5.4), 14 - (14.2)
TR-121354, Revision 1	NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report	All	3 - (3.9), 14 - (14.2)

Table 1.8-2 NuScale Technical Reports (Continued)

Technical Report Number	SDAA Technical Reports	Specific Report Sections Incorporated by Reference	FSAR Section
TR-123952, Revision 0	NuScale Containment Leakage Integrity Assurance, TR-123952, Revision 0 (proprietary)	None	3, 6, Exemption 7
TR-122844, Revision 0	NuScale Instrument Setpoint Methodology Technical Report, TR-122844	All	7 - (7.0, 7.2)
TR-121517, Revision 1	NuScale Power Module Short-Term Transient Analysis	All	3
TR-121507, Revision 0	Pipe Rupture Hazards Analysis, TR-121507	All	3
TR-130877, Revision 1	Pressure and Temperature Limits Methodology, TR-130877	All	5
TR-101310, Revision 1	Technical Specifications Development, TR-101310	None	16
TR-121515, Revision 0	US460 NuScale Power Module Seismic Analysis	All	3
TR-130721, Revision 0	Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel	All	5
TR-118318, Revision 1	NuScale Design of Physical Security System	All	13 - (13.6), 14 - (14.2)
TR-169856, Revision 0	NuScale US460 Statistical Subchannel Critical Heat Flux Analysis Probabilistic Uncertainties	All	4

1.9 Drawings and Other Detailed Information

Where appropriate, SDAA Part 2 provides simplified I&C, electrical, or mechanical drawings as figures. These figures are used in conjunction with the written text in the associated section to provide visual clarification of design information. Component position indications shown on these figures do not represent a specific operational state unless noted.

SDAA Part 2, Table 1.7-1, “Instrumentation and Controls Functional and Electrical One-Line Diagrams,” lists I&C functional diagrams and electrical one-line diagrams used in SDAA Part 2 (see SDAA Part 2, Figure 1.7-1a, “Electrical Symbols,” Figure 1.7-1b, “Electrical Symbols,” and Figure 1.7-2, “Instrumentation and Controls Symbol Legend,” for the legends of the symbols and characters used in electrical and I&C diagrams).

In COL Item 1.7-1, NuScale directed COL applicants that reference the NuScale Power Plant US460 standard design to provide site-specific diagrams and legends, as applicable.

SDAA Part 2, Table 1.7-2, lists system drawings used in SDAA Part 2 (see SDAA Part 2, Figures 1.7-3a through 1.7-3d, for a legend of the symbols and characters used in piping and instrumentation diagrams).

In COL Item 1.7-2, NuScale directed COL applicants that reference the NuScale Power Plant US460 standard design to list additional site-specific piping and instrumentation diagrams and legends as applicable.

1.10 Interfaces with Standard Designs

This section addresses interfaces between the NuScale standard design and the site-specific design provided in a COL application. SDAA Part 2, Section 1.2, identifies the SSCs included in the standard design; SDAA Part 2, Figure 1.2-1, provides a representation of the overall facility and general boundaries between the US460 standard design and site-specific design.

Design assumptions related to site-specific design that are the responsibility of an applicant that references the NuScale Power Plant US460 standard design are identified as a COL information item. In COL 1.8-1, NuScale directed COL applicant that references the NuScale Power Plant US460 standard design to provide a list of departures from the approved design. SDAA Part 2, Table 1.8-1, "Combined License Information Items" lists the COL information items, includes the COL information item text and identifies the section where the information item is located. The applicant addresses each COL information item in the section where it is located. Table 1.10-1 below is derived from SDAA Part 2, Table 1.8-1.

Table 1.10-1 Combined License Information Items

Item No.	Description of COL Information Item	Section
COL Item 1.1-1:	An applicant that references the NuScale Power Plant US460 standard design will identify the site-specific plant location.	1.1
COL Item 1.1-2:	An applicant that references the NuScale Power Plant US460 standard design will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1:	An applicant that references the NuScale Power Plant US460 standard design will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1:	An applicant that references the NuScale Power Plant US460 standard design will list site-specific electrical and instrumentation and control drawings, diagrams, and legends, as applicable.	1.7
COL Item 1.7-2:	An applicant that references the NuScale Power Plant US460 standard design will list additional site-specific piping and instrumentation diagrams and legends as applicable.	1.7
COL Item 1.8-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a list of departures from the approved design.	1.8
COL Item 1.9-1:	An applicant that references the NuScale Power Plant US460 standard design will review and address the conformance with regulatory criteria in effect six months before the submittal date of the application for the site-specific portions and operational aspects of the facility design.	1.9
COL Item 1.10-1:	An applicant that references the NuScale Power Plant US460 standard design will evaluate the potential hazards resulting from construction activities of the new NuScale facility to an operating nuclear power plant on a co-located site per 10 CFR 52.79(a)(31).	1.10

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 2.0-1:	An applicant that references the NuScale Power Plant US460 standard design will demonstrate that site-specific characteristics are bounded by the site parameters specified in Table 2.0-1. If site-specific values are not bounded by the values in Table 2.0-1, the applicant will demonstrate the acceptability of the site-specific values in the appropriate sections of its license application.	2.0
COL Item 2.1-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the site geographic and demographic characteristics.	2.1
COL Item 2.2-1:	An applicant that references the NuScale Power Plant US460 standard design will describe nearby industrial, transportation, and military facilities. The applicant will demonstrate that the design is acceptable for each of these potential hazards, or provide site-specific design alternatives.	2.2
COL Item 2.3-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific meteorological characteristics for Section 2.3.1 through Section 2.3.5, as applicable.	2.3
COL Item 2.4-1:	An applicant that references the NuScale Power Plant US460 standard design will investigate and describe the site-specific hydrologic characteristics for Section 2.4.1 through Section 2.4.14, except Section 2.4.8, Section 2.4.10, and Section 2.4.11.	2.4
COL Item 2.5-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific geology, seismology, and geotechnical characteristics for Section 2.5.1 through Section 2.5.5.	2.5
COL Item 3.3-1:	An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to severe and extreme (tornado and hurricane) wind loads will not collapse and adversely affect the Seismic Category I portions of the Reactor Building or of the Control Building.	3.3
COL Item 3.4-1:	An applicant that references the NuScale Power Plant US460 standard design will confirm the final location of structures, systems, and components subject to flood protection. The final routing of piping, and site-specific tanks or water source tanks are placed in locations that will not cause unanalyzed flooding to the Reactor Building or Control Building.	3.4
COL Item 3.4-2:	An applicant that references the NuScale Power Plant US460 standard design will develop the on-site program addressing the key points of flood mitigation consistent with the methodology described in Section 3.4.1. The key points to this program include the procedures for mitigating internal flooding events; development of the equipment list of structures, systems, and components subject to flood protection in each plant zone; and analysis providing assurance that the program reliably mitigates flooding to the identified structures, systems, and components consistent with flood levels identified in Table 3.4-1.	3.4
COL Item 3.4-3:	An applicant that references the NuScale Power Plant US460 standard design will develop an inspection and maintenance program to ensure that each water-tight door, penetration seal, or other “degradable” measure remains capable of performing its intended function.	3.4
COL Item 3.4-4:	An applicant that references the NuScale Power Plant US460 standard design will determine the extent of waterproofing and damp proofing needed for the underground portion of the Reactor Building, including the duct bank connection to the Reactor Building based on-site-specific conditions. Additionally, the applicant will provide the specified design life for waterstops, waterproofing, damp proofing, and watertight seals. If the design life is less than the operating life of the plant, the applicant will describe how continued protection will be ensured.	3.4
COL Item 3.4-5:	An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to external flooding will not collapse and adversely affect the Seismic Category I portions of the Reactor Building or of the Control Building.	3.4

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.5-1:	An applicant that references the NuScale Power Plant US460 standard design will demonstrate the site-specific turbine missile parameters are bounded by the standard design analysis, or provide a missile analysis using the site-specific turbine generator parameters to demonstrate that barriers adequately protect essential structures, systems, and components from turbine missiles. Parameters to verify are: limiting turbine missile spectrum (rotor and blade material properties); turbine rotor design, geometry and number of blades; final design of the Reactor Building exterior wall; and location of the turbines with respect to the Reactor Building.	3.5
COL Item 3.5-2:	An applicant that references the NuScale Power Plant US460 standard design will confirm the design-basis automobile missile parameters for the reference plant of velocity and maximum altitude of impact will not be exceeded as a result of extreme wind conditions that may occur in the vicinity of the site.	3.5
COL Item 3.5-3:	An applicant that references the NuScale Power Plant US460 standard design will evaluate site-specific hazards due to external events, such as turbine failures that can occur at nearby or co-located facilities, which may produce more energetic missiles that impact different locations than the design-basis missiles defined in Section 3.5.1.	3.5
COL Item 3.6-1:	An applicant that references the NuScale Power Plant US460 standard design will perform the pipe rupture hazards analysis (including dynamic and environmental effects) of the high- and moderate-energy lines outside the reactor pool bay in the Reactor Building (RXB). This analysis includes an evaluation of multi-module impacts in common pipe galleries, and evaluations regarding subcompartment pressurization. The as-built Pipe Rupture Hazards Analysis (PRHA) will show that the analysis of RXB piping bounds the possible effects of ruptures for the routings of lines outside of the RXB, or will perform the PRHA of the high- and moderate-energy lines outside the buildings.	3.6
COL Item 3.7-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific safe shutdown earthquake.	3.7
COL Item 3.7-2:	An applicant that references the NuScale Power Plant US460 standard design will provide site-specific time histories. In addition to the above criteria for cross correlation coefficients, time step and earthquake duration, strong motion durations, comparison to response spectra and power spectra density, the applicant will also confirm that site-specific ratios V/A and AD/V^2 (A, V, D, are peak ground acceleration, ground velocity, and ground displacement, respectively) are consistent with characteristic values for the magnitude and distance of the appropriate controlling events defining the site-specific uniform hazard response spectra.	3.7
COL Item 3.7-3:	An applicant that references the NuScale Power Plant US460 standard design will include an analysis of the performance-based response spectra established at the surface and intermediate depth(s) that take into account the complexities of the subsurface layer profiles of the site and provide a technical justification for the adequacy of vertical to horizontal (V/H) spectral ratios used in establishing the site-specific foundation input response spectra and the performance-based response spectra for the vertical direction.	3.7
COL Item 3.7-4:	An applicant that references the NuScale Power Plant US460 standard design will: • develop a site-specific strain-compatible soil profile. • confirm that the criterion for the minimum required response spectrum is satisfied. • determine whether the seismic site characteristics fall within the seismic design parameters such as soil layering assumptions used in the standard design, range of soil parameters, shear wave velocity values, and minimum soil bearing capacity.	3.7
COL Item 3.7-5:	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific analysis that assesses the effects of soil separation. The applicant will confirm that the in-structure response spectra in the soil separation cases are bounded by the in-structure response spectra described in Section 3.7.2.	3.7

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.7-6:	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific analysis that assesses the effects of non-vertically propagating seismic waves on the free-field ground motions and seismic responses of Seismic Category I structures, systems, and components.	3.7
COL Item 3.7-7:	An applicant that references the NuScale Power Plant US460 standard design will confirm that nearby structures exposed to a site-specific safe shutdown earthquake will not collapse and adversely affect Seismic Category I portions of the Reactor Building and Control Building.	3.7
COL Item 3.7-8:	An applicant that references the NuScale Power Plant US460 standard design will demonstrate that the site-specific seismic demand is bounded by the Final Safety Analysis Report capacity for an empty dry dock condition.	3.7
COL Item 3.7-9:	An applicant that references the NuScale Power Plant US460 standard design will perform a soil-structure interaction analysis of the Reactor Building and the Control Building using the NuScale ANSYS models for those structures. The applicant will confirm that the site-specific seismic demands of the standard design for critical structures, systems, and components in Appendix 3B are bounded by the corresponding design certified seismic demands and, if not, the standard design for critical structures, systems, and components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demands. Seismic demands investigated shall include forces, moments, deformations, in-structure response spectra, and seismic stability of the structures.	3.7
COL Item 3.7-10:	An applicant that references the NuScale Power Plant US460 standard design will determine the means and methods of lifting the bioshield. An applicant will demonstrate that bioshield components and connections can withstand the bioshield loads and appropriate load factors.	3.7
COL Item 3.7-11:	An applicant that references the NuScale Power Plant US460 standard design will prepare site-specific procedures for seismic instrumentation maintenance and post-earthquake activities. Administrative procedures define the maintenance and repair of the seismic instrumentation to keep the maximum number of instruments in-service during plant operations and shutdown. The procedures for post-earthquake activities must provide sufficient information to determine if the level of earthquake ground motion requiring shutdown has been exceeded and appropriate corrective actions to be taken if needed. Guidance for procedure development is in Regulatory Guide 1.12, “Nuclear Power Plant Instrumentation for Earthquakes,” Regulatory Guide 1.166, “Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake,” and EPRI Report 3002005284, “Guidelines for Nuclear Plant Response to an Earthquake” (Reference 3.7.4-1).	3.7
COL Item 3.8-1:	An applicant that references the NuScale Power Plant US460 standard design will provide the design and structural analysis of the reactor flange tool. In addition to analysis of the reactor flange tool, the applicant will provide structural analysis of the fuel in response to external forces when the reactor vessel is located in the reactor flange tool.	3.8
COL Item 3.8-2:	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific program for monitoring and maintenance of the Seismic Category I structures in accordance with the requirements of 10 CFR 50.65 as discussed in Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Monitoring is to include below grade walls, groundwater chemistry if needed, base settlements, and differential displacements.	3.8
COL Item 3.8-3:	An applicant that references the NuScale Power Plant US460 standard design will identify local stiff and soft spots in the foundation soil and address these in the design, as necessary.	3.8

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.9-1:	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific seismic analysis in accordance with Section 3.7.2. In addition to the requirements of Section 3.7, for sites where the high frequency portion of the site-specific spectrum is not bounded by the certified seismic design response spectra, the standard design of NuScale Power Module components will be shown to have appropriate margin or should be appropriately modified to accommodate the site-specific demand.	3.9
COL Item 3.9-2:	An applicant that references the NuScale Power Plant US460 standard design will complete an assessment of piping systems inside the Reactor Building to determine the portions of piping to be tested for vibration, thermal expansion, and dynamic effects. Piping systems within the scope of this testing include American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Class 1, 2, and 3 piping systems, other high-energy piping systems inside Seismic Category I structures or those whose failure would reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and Seismic Category I portions of moderate-energy piping systems located outside of containment. The applicant may select the portions of piping in the design for which vibration testing is performed while considering the piping system design and analysis, including the vibration screening and analysis results and scope of testing as identified by the Comprehensive Vibration Assessment Program.	3.9
COL Item 3.9-3:	An applicant that references the NuScale Power Plant US460 standard design will verify that evaluations are performed during detailed design of the main steam lines, using acoustic resonance screening criteria and additional calculations as necessary (e.g., Strouhal number) to determine if there is a concern. The methodology in “NuScale Comprehensive Vibration Assessment Program Analysis Technical Report,” TR-121353 is acceptable for this purpose. The applicant will update Section 3.9.2.1.1.1 to describe the results of this evaluation.	3.9
COL Item 3.9-4:	An applicant that references the NuScale Power Plant US460 standard design will provide applicable test procedures before the start of testing and will submit test and inspection results from the Comprehensive Vibration Assessment Program for the NuScale Power Module in accordance with Regulatory Guide 1.20.	3.9
COL Item 3.9-5:	An applicant that references the NuScale Power Plant US460 standard design will implement a control rod drive system Operability Assurance Program that meets the requirements described in Section 3.9.4.4 and provide a summary of the testing program and results.	3.9
COL Item 3.9-6:	An applicant that references the NuScale Power Plant US460 standard design will develop a Reactor Vessel Internals Reliability Program to address industry identified aging degradation mechanism issues.	3.9
COL Item 3.9-7:	An applicant that references the NuScale Power Plant US460 standard design will provide a summary of reactor core support structure American Society of Mechanical Engineers (ASME) service level stresses, deformation, and cumulative usage factor values for each component and each operating condition in conformance with ASME Boiler and Pressure Vessel Code Section III Subsection NG.	3.9
COL Item 3.9-8:	An applicant that references the NuScale Power Plant US460 standard design will establish Preservice and Inservice Testing Programs. These programs are to be consistent with the requirements in the latest edition and addenda of the American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code incorporated by reference in 10 CFR 50.55a.	3.9
COL Item 3.9-9:	An applicant that references the NuScale Power Plant US460 standard design will develop specific test procedures to allow detection and monitoring of power-operated valve assembly performance sufficient to satisfy periodic verification design basis capability requirements.	3.9

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 3.9-10:	An applicant that references the NuScale Power Plant US460 standard design will develop specific test procedures to allow detection and monitoring of emergency core cooling system valve assembly performance sufficient to satisfy periodic verification of design-basis capability requirements.	3.9
COL Item 3.11-1:	An applicant that references the NuScale Power Plant US460 standard design will submit a full description of the Environmental Qualification Program and milestones and completion dates for program implementation.	3.11
COL Item 3.11-2:	An applicant that references the NuScale Power Plant US460 standard design will ensure the Environmental Qualification Program cited in COL Item 3.11-1 includes a description of how equipment subject to program requirements will be monitored and managed throughout plant life. This description will include methodology to ensure equipment located in harsh or mild environments will remain qualified if an actual environment parameter, such as temperature, pressure, humidity, radiation, or chemical exposure deviates from the acceptable range for which the component is qualified.	3.11
COL Item 3.11-3:	An applicant that references the NuScale Power Plant US460 standard design will implement an Environmental Qualification Operational Program that incorporates the aspects in Section 3.11.5 specific to the environmental qualification of mechanical and electrical equipment. This program will include an update to Table 3.11-1 to include commodities that support equipment listed in Table 3.11-1.	3.11
COL Item 3.12-1:	An applicant that references the NuScale Power Plant US460 standard design may use a piping analysis program other than the programs listed in Section 3.12.4; however, the applicant will implement a benchmark program using the models for the NuScale Power Plant US460 standard design.	3.12
COL Item 3.12-2:	An applicant that references the NuScale Power Plant US460 standard design will confirm that the site-specific seismic response is within the parameters specified in Section 3.7. An applicant may perform a site-specific piping stress analysis in accordance with the methodologies described in this section, as appropriate.	3.12
COL Item 3.13-1:	An applicant that references the NuScale Power Plant US460 standard design will provide an inservice inspection program for American Society of Mechanical Engineers Class 1, 2, and 3 threaded fasteners. The program will identify the applicable edition and addenda of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI and ensure compliance with 10 CFR 50.55a.	3.13
COL Item 4.2-1:	An applicant that references the NuScale Power Plant US460 standard design and wishes to utilize non-baseload operations will provide justification for the fuel performance codes and methods corresponding to the desired operation.	4.2
COL Item 5.2-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a certified Overpressure Protection Report in compliance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Subarticles NB-7200 and NC-7200 to demonstrate the reactor coolant pressure boundary and secondary system design contains adequate overpressure protection features, including low temperature overpressure protection features.	5.2
COL Item 5.2-2:	An applicant that references the NuScale Power Plant US460 standard design will develop and implement a Strategic Water Chemistry Plan. The Strategic Water Chemistry Plan will provide the optimization strategy for maintaining primary coolant chemistry and provide the basis for requirements for sampling and analysis frequencies, and corrective actions for control of primary water chemistry consistent with the latest version of the Electric Power Research Institute Pressurized Water Reactor Primary Water Chemistry Guidelines.	5.2

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 5.2-3:	An applicant that references the NuScale Power Plant US460 standard design will develop and implement a Boric Acid Control Program that includes: inspection elements to ensure the integrity of the reactor coolant pressure boundary components for subsequent service, monitoring of the containment atmosphere for evidence of reactor coolant system leakage, the type of visual or other nondestructive inspections to be performed, and the required inspection frequency.	5.2
COL Item 5.2-4:	An applicant that references the NuScale Power Plant US460 standard design will develop site-specific preservice examination, inservice inspection, and inservice testing program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and the American Society of Mechanical Engineers Operations and Maintenance Code, and will establish implementation milestones. If applicable, an applicant that references the NuScale Power Plant US460 standard design will identify the implementation milestone for the augmented inservice inspection program. The applicant will identify the applicable edition of the American Society of Mechanical Engineers Code utilized in the program plans consistent with the requirements of 10 CFR 50.55a.	5.2
COL Item 5.2-5:	An applicant that references the NuScale Power Plant US460 standard design will establish plant-specific procedures that specify operator actions for identifying, monitoring, and trending reactor coolant system leakage in response to prolonged low leakage conditions that exist above normal leakage rates and below the technical specification limits. The objective of the methods of detecting and trending the reactor coolant pressure boundary leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached.	5.2
COL Item 5.3-1:	An applicant that references the NuScale Power Plant US460 standard design will choose the final transients to generate the reactor vessel pressure-temperature limits report and limiting conditions for operation. An applicant that references the NuScale Power Plant US460 standard design will confirm that the design geometries, final transients, and material properties of the reactor pressure vessel are bounded by (or identical to) those used in the Pressure and Temperature Limits Methodology to confirm that the example curves in the Standard Design Approval Application are applicable. Operating procedures will ensure that pressure-temperature limits for the as-built reactor are not exceeded. These procedures will be based on the limits defined in the pressure-temperature limits report and material properties of the as-built reactor vessel.	5.3
COL Item 5.4-1:	An applicant that references the NuScale Power Plant US460 standard design will develop and implement a Steam Generator Program for periodic monitoring of the degradation of steam generator components to ensure that steam generator tube integrity is maintained. The Steam Generator Program will be based on the latest revision of Nuclear Energy Institute NEI 97-06, "Steam Generator Program Guidelines," and applicable Electric Power Research Institute steam generator guidelines at the time of the application. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, shell side integrity assessment, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting. The Steam Generator Program for the NuScale Power Plant US460 will require 100 percent tube inspections at the first refueling outage. In addition to other requirements and inspections, the Steam Generator Program for the first module to undergo a refueling outage will require at least 20 percent tube inspections during each subsequent refueling outage for the first 72 effective full power months after the first refueling outage. Subsequent applicants that reference the NuScale Power Plant US460 design shall provide justification that the results of the first module's inspections are applicable to the subsequent modules in order to demonstrate that these additional inspection requirements are not applicable to the subsequent modules.	5.4

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 6.2-1:	An applicant that references the NuScale Power Plant US460 standard design will verify that the final design of the containment vessel meets the design-basis requirement to maintain flange contact pressure at accident temperature, concurrent with peak accident pressure.	6.2
COL Item 6.3-1:	An applicant that references the NuScale Power Plant US460 standard design will describe a containment cleanliness program that limits debris within containment. The program should contain the following elements: • Foreign material exclusion controls to limit the introduction of foreign material and debris sources into containment. • Maintenance activity controls, including temporary changes, that confirm the emergency core cooling system function is not reduced by changes to analytical inputs or assumptions or other activities that could introduce debris or potential debris sources into containment. • Controls that prohibit the introduction of coating materials into containment. • An inspection program to confirm containment vessel cleanliness before closing for normal power operation.	6.3
COL Item 6.4-1:	An applicant that references the NuScale Power Plant US460 standard design will comply with Regulatory Guide 1.78 Revision 2, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."	6.4
COL Item 6.6-1:	An applicant that references the NuScale Power Plant US460 standard design will develop Preservice Inspection and Inservice Inspection Program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, and will establish the implementation milestones for the program. The applicant will identify the applicable edition of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code used in the program plan consistent with the requirements of 10 CFR 50.55a. The applicant will, if needed, address the use of a single Inservice Inspection Program for multiple NuScale Power Modules, including any alternative to the code that may be necessary to implement such an Inservice Inspection Program.	6.6
COL Item 7.0-1:	Not used.	7.0
COL Item 7.2-1:	An applicant that references the NuScale Power Plant US460 standard design will implement the life cycle processes for the operation phase for the instrumentation and controls systems, as defined in IEEE Std 1074-2006 and IEEE Std 1012-2004.	7.2
COL Item 7.2-2:	An applicant that references the NuScale Power Plant US460 standard design will implement the life cycle processes for the maintenance phase for the instrumentation and controls systems, as defined in IEEE Std 1074-2006 and IEEE Std 1012-2004.	7.2
COL Item 7.2-3:	An applicant that references the NuScale Power Plant US460 standard design will implement the life cycle processes for the retirement phase for the instrumentation and controls systems, as defined in Institute of IEEE Std 1074-2006 and IEEE Std 1012-2004. The Digital I&C Software Configuration Management Plan provides guidance for the retirement and removal of a software product from use.	7.2
COL Item 9.1-1:	An applicant that references the NuScale Power Plant US460 standard design will develop plant programs and procedures for safe operations during handling and storage of new and spent fuel assemblies, including criticality control.	9.1
COL Item 9.1-2:	An applicant that references the NuScale Power Plant US460 standard design will provide the design of the spent fuel pool storage racks, including the structural dynamic and stress analyses, thermal hydraulic cooling analyses, criticality safety analysis, and material compatibility evaluation.	9.1
COL Item 9.1-3:	An applicant that references the NuScale Power Plant US460 standard design will provide the periodic testing plan for fuel handling equipment.	9.1
COL Item 9.1-4:	An applicant that references the NuScale Power Plant US460 standard design will describe the process for handling and receipt of critical loads including NPMs.	9.1

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 9.1-5:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the program governing heavy loads handling. The program should address • operating and maintenance procedures. • inspection and test plans. • personnel qualification and operator training. • detailed description of the safe load paths for movement of heavy loads.	9.1
COL Item 9.3-1:	An applicant that references the NuScale Power Plant US460 standard design will submit a leakage control program for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. The leakage control program will include an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems to as low as practical.	9.3
COL Item 9.5-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the offsite communication system, how that system interfaces with the onsite communications system, as well as how continuous communications capability is maintained to ensure effective command and control with onsite and offsite resources during both normal and emergency situations.	9.5
COL Item 10.3-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a site-specific Secondary Water Chemistry Control Program based on the latest revision of the Electric Power Research Institute Pressurized Water Reactor Secondary Water Chemistry Guidelines and Nuclear Energy Institute 97-06 at the time of the application.	10.3
COL Item 10.3-2:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the Flow-Accelerated Corrosion Monitoring Program for the steam and power conversion systems based on Generic Letter 89-08 and the latest revision of the Electric Power Research Institute NSAC-202L at the time of the application.	10.3
COL Item 10.4-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a secondary water chemistry analysis. This analysis must show that the size, materials, and capacity of the feedwater treatment system equipment and components satisfies the water quality requirements of the Secondary Water Chemistry Control Program described in Section 10.3.5, and it is compatible with the chemicals used.	10.4
COL Item 11.2-1:	An applicant that references the NuScale Power Plant US460 standard design will calculate doses to members of the public using the site-specific parameters, compare those liquid effluent doses to the numerical design objectives of 10 CFR 50, Appendix I, and comply with the requirements of 10 CFR 20.1302 and 40 CFR 190.	11.2
COL Item 11.2-2:	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific evaluation of the consequences of an accidental release of radioactive liquid from the pool surge control system storage tank in accordance with NRC Branch Technical Position 11-6.	11.2
COL Item 11.2-3:	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific evaluation using the site-specific source term and dilution flow for liquid effluent releases, and confirm that the discharge concentrations do not exceed the limits specified by 10 CFR 20, Appendix B, Table 2.	11.2
COL Item 11.2-4:	An applicant that references the NuScale Power Plant US460 standard design will perform a cost-benefit analysis as required by 10 CFR 50.34a and 10 CFR 50, Appendix I, to demonstrate conformance with regulatory requirements. This cost-benefit analysis is to be performed using the guidance of Regulatory Guide 1.110.	11.2
COL Item 11.3-1:	An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific cost-benefit analysis.	11.3

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 11.3-2:	An applicant that references the NuScale Power Plant US460 standard design will calculate doses to members of the public using the site-specific parameters, compare those gaseous effluent doses to the numerical design objectives of 10 CFR 50, Appendix I, and comply with the requirements of 10 CFR 20.1302 and 40 CFR 190.	11.3
COL Item 11.3-3:	An applicant that references the NuScale Power Plant US460 standard design will perform an analysis in accordance with Branch Technical Position 11-5 using the site-specific parameters.	11.3
COL Item 12.1-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the operational program to maintain exposures to ionizing radiation as far below the dose limits as practical, as low as reasonably achievable (ALARA).	12.1
COL Item 12.2-1:	An applicant that references the NuScale Power Plant US460 standard design will describe additional site-specific contained radiation sources that exceed 100 millicuries (including sources for instrumentation and radiography) not identified in Section 12.2.1.	12.2
COL Item 12.3-1:	An applicant that references the NuScale Power Plant US460 standard design will develop the administrative controls regarding access to high radiation areas per the guidance of Regulatory Guide 8.38.	12.3
COL Item 12.3-2:	An applicant that references the NuScale Power Plant US460 standard design will develop the administrative controls regarding access to very high radiation areas per the guidance of Regulatory Guide 8.38.	12.3
COL Item 12.3-3:	An applicant that references the NuScale Power Plant US460 standard design will specify personnel exposure monitoring hardware, specify contamination identification and removal hardware, and establish administrative controls and procedures to control access into and exiting the radiologically controlled area.	12.3
COL Item 12.3-4:	An applicant that references the NuScale Power Plant US460 standard design will develop the processes and programs necessary for the implementation of 10 CFR 20.1501 related to conducting radiological surveys, maintaining proper records, calibration of equipment, and personnel dosimetry.	12.3
COL Item 12.3-5:	An applicant that references the NuScale Power Plant US460 standard design will develop the processes and programs necessary for the use of portable airborne monitoring instrumentation, including accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.	12.3
COL Item 12.3-6:	An applicant that references the NuScale Power Plant US460 standard design will develop the processes and programs associated with Objectives 5 and 6, to work in conjunction with design features, necessary to demonstrate compliance with 10 CFR 20.1406, and the guidance of Regulatory Guide 4.21.	12.3
COL Item 12.4-1:	An applicant that references the NuScale Power Plant US460 standard design will estimate doses to construction personnel from a co-located existing operating nuclear power plant that is not a NuScale Power Plant.	12.4
COL Item 12.5-1:	An applicant that references the NuScale Power Plant US460 standard design will describe elements of the operational radiation protection program to ensure that occupational and public radiation exposures are as low as reasonably achievable in accordance with 10 CFR 20.1101.	12.5
COL Item 13.1-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the corporate or home office management and technical support organization, including a description of the qualification requirements for (1) each identified position or class of positions that provide technical support to the on-site operating organization, and (2) individuals holding management and supervisory positions in organizational units providing technical support to the on-site operating organization.	13.1

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 13.1-2:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the proposed structure, functions, and responsibilities of the on-site organization necessary to operate and maintain the plant. The proposed operating staff shall be consistent with the minimum licensed operator staffing requirements in Section 18.5.	13.1
COL Item 13.1-3:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the qualification requirements for each management, operating, technical, and maintenance position described in the operating organization.	13.1
COL Item 13.2-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a description and schedule of the Initial Training and Qualification as well as Requalification Programs for reactor operators and senior reactor operators.	13.2
COL Item 13.2-2:	An applicant that references the NuScale Power Plant US460 standard design will provide a description and schedule of the Non-Licensed Plant Staff Training Programs including initial training, periodic retraining, and qualification requirements.	13.2
COL Item 13.3-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the Emergency Response facilities for management of overall licensee Emergency Response. The facility will meet the requirements of 10 CFR 52.79.	13.3
COL Item 13.3-2:	An applicant that references the NuScale Power Plant US460 standard design will provide a comprehensive Emergency Plan in accordance with 10 CFR Part 50 and 10 CFR 52.79(a)(21).	13.3
COL Item 13.4-1:	An applicant that references the NuScale Power Plant US460 standard design will provide site-specific information, including implementation milestones, for Operational Programs: • Inservice Inspection Programs (Section 5.2, Section 5.4, and Section 6.6) • Inservice Testing Programs (Section 3.9 and Section 5.2) • Environmental Qualification Program (Section 3.11) • Preservice Testing Program (Section 3.9.6 and Section 5.4) • Preservice Testing Program (Section 3.9.6 and Section 5.2) • Containment Leakage Rate Testing Program (Section 6.2) • Fire Protection Program (Section 9.5.1) • Process and Effluent Monitoring and Sampling Program (Section 11.5) • Radiation Protection Program (Section 12.5) • Non-Licensed Plant Staff Training Program (Section 13.2) • Reactor Operator Training Program (Section 13.2) • Reactor Operator Requalification Program (Section 13.2) • Emergency Planning (Section 13.3) • Process Control Program (Section 11.4) • Security (Section 13.6) • Quality Assurance Program (Section 17.5) • Maintenance Rule (Section 17.6) • Initial Test Program (Section 14.2)	13.4
COL Item 13.5-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific procedures that provide administrative control for activities that are important for the safe operation of the facility consistent with the guidance provided in Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operation),” Revision 3.	13.5
COL Item 13.5-2:	An applicant that references the NuScale Power Plant US460 standard design will provide a plan for the development, implementation, and control of administrative procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the applicant will identify the group within the operating organization responsible for maintaining these procedures.	13.5

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 13.5-3:	An applicant that references the NuScale Power Plant US460 standard design will describe the process to manage the development, review and approval of the site-specific procedures that operators use in the main control room and locally in the plant, including normal operating procedures, abnormal operating procedures, and emergency operating procedures. The applicant will describe the classification system for these procedures, and the general format and content of the different classifications.	13.5
COL Item 13.5-4:	An applicant that references the NuScale Power Plant US460 standard design will provide a plan for the development, implementation, and control of operating procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the applicant will identify the group within the operating organization responsible for maintaining these procedures.	13.5
COL Item 13.5-5:	An applicant that references the NuScale Power Plant US460 standard design will provide a plan for the development, implementation, and control of emergency operating procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the applicant will identify the group within the operating organization responsible for maintaining these procedures.	13.5
COL Item 13.5-6:	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific maintenance and other operating procedures, including how these procedures are classified, and the general format and content of the different classifications. The categories of procedures listed below will be included: • plant radiation protection procedures • emergency preparedness procedures • calibration and test procedures • chemical-radiochemical control procedures • radioactive waste management procedures • maintenance and modification procedures • material control procedures • plant security procedures	13.5
COL Item 13.5-7:	An applicant that references the NuScale Power Plant US460 standard design will provide a plan for the development, implementation, and control of maintenance and other operating procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the applicant will identify what group or groups within the operating organization have the responsibility for maintaining and following these procedures.	13.5
COL Item 13.6-1:	An applicant that references the NuScale Power Plant US460 standard design will provide the following: • Security Plans (Physical Security, Security Training and Qualification, and Safeguards Contingency Plans) • proposed site security provisions to be implemented during construction and as modules are completed and become operational • elements of the physical security system not located within the nuclear island and structures	13.6
COL Item 13.6-2:	An applicant that references the NuScale Power Plant US460 standard design will be responsible for the requirements described in Table 5-1 of TR-118318, "NuScale Design of Physical Security Systems" (Reference 13.6-1).	13.6
COL Item 13.6-3:	An applicant that references the NuScale Power Plant US460 standard design will provide a secondary alarm station that is equal and redundant to the central alarm station.	13.6
COL Item 13.6-4:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the Access Authorization Program.	13.6
COL Item 13.6-5:	An applicant that references the NuScale Power Plant US460 standard design will provide a Cybersecurity Plan.	13.6

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 14.2-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific organizations that manage, supervise, or execute the Initial Test Program, including the associated training requirements.	14.2
COL Item 14.2-2:	An applicant that references the NuScale Power Plant US460 standard design will develop the Startup Administration Manual that will contain the administrative procedures and requirements that control the activities associated with the Initial Test Program. The applicant will provide a milestone for completing the Startup Administrative Manual and making it available for Nuclear Regulatory Commission inspection.	14.2
COL Item 14.2-3:	An applicant that references the NuScale Power Plant US460 standard design will identify the specific operator training to be conducted during low-power testing related to the resolution of Three Mile Island Action Plan Item I.G.1, as described in NUREG-0660, NUREG-0694, and NUREG-0737.	14.2
COL Item 14.2-4:	An applicant that references the NuScale Power Plant US460 standard design will provide a schedule for the Initial Test Program.	14.2
COL Item 14.2-5:	An applicant that references the NuScale Power Plant US460 standard design will provide a test abstract for the potable water system pre-operational testing.	14.2
COL Item 14.2-6:	An applicant that references the NuScale Power Plant US460 standard design will provide a test abstract for the seismic monitoring system pre-operational testing.	14.2
COL Item 14.3-1:	An applicant that references the NuScale Power Plant US460 standard design will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for emergency planning.	14.3
COL Item 14.3-2:	An applicant that references the NuScale Power Plant US460 standard design will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for structures, systems, and components within their scope.	14.3
COL Item 16.1-1:	An applicant that references the NuScale Power Plant US460 standard design will provide the final plant-specific information identified by [ ] in the generic Technical Specifications and generic Technical Specification Bases.	16.1
COL Item 16.1-2:	An applicant that references the NuScale Power Plant US460 standard design will prepare and maintain an owner-controlled requirements manual that includes owner-controlled limits and requirements described in the Bases of the Technical Specifications or as otherwise specified in the FSAR.	16.1
COL Item 16.1-3:	An applicant that references the NuScale Power Plant US460 standard design, and uses allocations for sensor response times based on records of tests, vendor test data, or vendor engineering specifications as described in the bases for Surveillance Requirement 3.3.1.3, will do so for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.	16.1
COL Item 17.4-1:	An applicant that references the NuScale Power Plant US460 standard design will describe the Reliability Assurance Program conducted during the operations phases of the plant's 60-year design life.	17.4
COL Item 17.4-2:	An applicant that references the NuScale Power Plant US460 standard design will identify site-specific structures, systems, and components within the scope of the Reliability Assurance Program.	17.4
COL Item 17.4-3:	An applicant that references the NuScale Power Plant US460 standard design will identify the quality assurance controls for the Reliability Assurance Program structures, systems, and components during site-specific design, procurement, fabrication, construction, and preoperational testing activities.	17.4
COL Item 18.5-1:	An applicant that references the NuScale Power Plant US460 standard design will address the staffing and qualifications of non-licensed operators.	18.5

Table 1.10-1 Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 18.12-1:	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the Human Performance Monitoring Program in accordance with applicable NUREG-0711 or equivalent criteria.	18.12
COL Item 19.1-1:	An applicant that references the NuScale Power Plant US460 standard design will identify and describe the use of the probabilistic risk assessment in support of licensee programs being implemented during the COL application phase.	19.1
COL Item 19.1-2:	An applicant that references the NuScale Power Plant US460 standard design will identify and describe specific risk-informed applications being implemented during the COL application phase.	19.1
COL Item 19.1-3:	An applicant that references the NuScale Power Plant US460 standard design will specify and describe the use of the probabilistic risk assessment in support of licensee programs during the construction phase (from issuance of the COL up to initial fuel loading).	19.1
COL Item 19.1-4:	An applicant that references the NuScale Power Plant US460 standard design will specify and describe risk-informed applications during the construction phase (from issuance of the COL up to initial fuel loading).	19.1
COL Item 19.1-5:	An applicant that references the NuScale Power Plant US460 standard design will specify and describe the use of the probabilistic risk assessment in support of licensee programs during the operational phase (from initial fuel loading through commercial operation).	19.1
COL Item 19.1-6:	An applicant that references the NuScale Power Plant US460 standard design will specify and describe risk-informed applications during the operational phase (from initial fuel loading through commercial operation).	19.1
COL Item 19.1-7:	An applicant that references the NuScale Power Plant US460 standard design will evaluate site-specific external event hazards (e.g., liquefaction, slope failure), screen those for risk significance, and evaluate the risk associated with external hazards that are not bounded by the standard design.	19.1
COL Item 19.1-8:	An applicant that references the NuScale Power Plant US460 standard design will confirm the validity of the “key assumptions” and data used in the standard design approval application PRA and modify, as necessary, for applicability to the as-built, as-operated PRA.	19.1
COL Item 19.2-1:	An applicant that references the NuScale Power Plant US460 standard design will develop severe accident management guidelines and other administrative controls to define the response to beyond-design-basis events.	19.2
COL Item 19.2-2:	An applicant that references the NuScale Power Plant US460 standard design will use the site-specific probabilistic risk assessment to evaluate and identify improvements in the reliability of core and containment heat removal systems as specified by 10 CFR 50.34(f)(1)(i).	19.2
COL Item 19.2-3:	Not used.	
COL Item 19.2-4:	An applicant that references the NuScale Power Plant US460 standard design will identify from Table 19.2-8 the components and their severe accident doses for cases where the severe accident dose is greater than the environmental qualification dose.	19.2
COL Item 19.3-1:	An applicant that references the NuScale Power Plant US460 standard design will identify site-specific Regulatory Treatment of Nonsafety Systems structures, systems, and components and applicable process controls.	19.3

1.11 Conformance with Regulatory Criteria

SDAA Part 2, Section 1.9, provides a guide to NuScale conformance with regulatory criteria. This includes conformance with RGs, SRPs, generic issues (including Three Mile Island requirements), operational experience (i.e., generic communications), and advanced and evolutionary light-water reactor design issues. The following FSAR tables describe conformance with regulatory criteria in effect 6 months before the docket date as listed below.

Table 1.9-1, Conformance Status Legend

Table 1.9-2, Conformance with Regulatory Guides

Table 1.9-3, Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard

Table 1.9-4, Conformance with Interim Staff Guidance

Table 1.9-5, Conformance with Three Mile Island Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)

Table 1.9-6, Evaluation of Operating Experience (Generic Letters and Bulletins)

Table 1.9-7, Conformance with Advanced and Evolutionary Light Water Reactor Design Issues (Commission papers (SECYs) and associated Staff Requirements Memoranda)

Table 1.9-8, Conformance with SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs"

In COL Item 1.9-1, NuScale directed COL applicants that reference the NuScale Power Plant US460 standard design to review and address the conformance with regulatory criteria in effect six months before the submittal date of the application for the site-specific portions and operational aspects of the facility design.

1.12 Summary of Open Items

As a result of the staff's review of the NuScale SDAA, the staff identified several issues (open items) all of which the applicant subsequently addressed through information submitted on the docket. The staff has directly evaluated the adequacy of the information submitted on the docket to address the issues, which is included in the current version of the SDAA and has closed the open items. The staff's regulatory findings documented in this FSER are based on the latest version of the application on the docket. Appendix E to this report lists the issuance and response dates for each RAI the staff issued to the applicant.

1.13 Summary of Confirmatory Items

The NRC staff's review of the NuScale SDAA identified several confirmatory items. An item is identified as confirmatory if the staff and NuScale have agreed on a resolution of a particular item, but the resolution has not yet been formally documented in the SDAA. All confirmatory items have been closed for the SDAA review.

1.14 Index of Exemptions

In accordance with 10 CFR 52.139, "Standards for Review of Applications," the staff used the current regulations in 10 CFR Part 20, "Standards for Protection against Radiation"; 10 CFR Part 50; 10 CFR Part 73, "Physical Protection of Plants and Materials"; and 10 CFR Part 100, "Reactor Site Criteria," in reviewing the NuScale SMR SDAA. During this review, the staff recognized, among other considerations, that the application of certain regulations to the NuScale SMR design would not serve the underlying purpose of the rule from which exemption is being sought or would not be necessary to achieve the underlying purpose of the rule.

NuScale submitted 18 exemption requests, which are provided in SDAA Part 7, “Exemptions,” with an introduction, justification for the request, regulatory basis, and conclusion. Table 1.14-1 below lists the FSER sections where the staff has dispositioned these exemption requests.

Table 1.14-1: NuScale Standard Design Approval Exemptions

FSER Section	Exemption
5.4.5 6.2	NuScale requests an exemption from 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi), which require high point vents for the reactor coolant system (RCS), reactor pressure vessel (RPV) head, and other systems required to maintain adequate core cooling.
6.2.5	NuScale requests an exemption from 10 CFR 50.44(c)(4) and 10 CFR 50.34(f)(2)(xvii)(C), which require the capability for monitoring combustible gases during an accident.
7.1 7.2 15.8 19.1.9 19.2.2	NuScale requests an exemption from the portion of 10 CFR 50.62(c)(1) requiring diverse equipment to initiate a turbine trip under conditions indicative of an anticipated transient without scram (ATWS).
3.1 3.2 7.1.2.2 8.2 8.3.1 8.3.2 8.4 15 19.1 19.2	NuScale requests an exemption from General Design Criterion (GDC) 17 because there are no safety-related functions in the NuScale US460 standard design that rely on electrical power. NuScale further requests exemptions from GDC 18 and from the portions of GDCs 34, 35, 38, 41, and 44 addressing electric power as conforming changes.
9.3.4 6.3	NuScale requests an exemption from General Design Criterion (GDC) 33, which requires a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary (RCPB).
5.3.4	NuScale requests an exemption from 10 CFR 50.60, which requires that light water reactors meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary (RCPB) set forth in 10 CFR Part 50, Appendices G and H.
6.2.6	NuScale requests an exemption from General Design Criterion (GDC) 52, which requires that the containment be designed so that periodic integrated leak rate testing (ILRT) can be conducted at containment design pressure.

Table 1.14-1: NuScale Standard Design Approval Exemptions (Comntinued)

FSER Section	Exemption
6.2.1 6.2.2 6.3	NuScale requests an exemption from General Design Criterion (GDC) 40, which requires design provisions for periodic pressure and functional testing of the containment heat removal system.
5.4.4 6.2.4	NuScale requests exemptions from General Design Criteria (GDCs) 55, 56, and 57, as applied to several containment penetrations in the US460 standard design. GDCs 55, 56, and 57 specify containment isolation provisions for piping system lines penetrating primary containment and generally require one isolation barrier inside containment and one outside containment.
15	NuScale requests an exemption from certain portions of 10 CFR Part 50, Appendix K regarding features that are required of the emergency core cooling system (ECCS) evaluation model (EM).
5.4.6	NuScale requests an exemption from the portions of 10 CFR 50.34(f)(2)(xx) applicable to pressurizer level indicators. 10 CFR 50.34(f)(2)(xx) specifies power requirements for pressurizer relief valves, block valves, and level indicators.
5.4.6	NuScale requests an exemption from 10 CFR 50.34(f)(2)(xiii), which requires providing power supplies for pressurizer heaters and associated motive and control interfaces to establish and maintain natural circulation in hot standby conditions.
5.2.5 6.2.4 9.3.6	NuScale requests an exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the containment evacuation system (CES). The rule requires automatic containment isolation on a high radiation signal of systems that provide a path to the environs from containment.
4.2.4.6	NuScale requests an exemption from 10 CFR 50.46 concerning zircaloy or ZIRLO as acceptable fuel rod cladding materials. The NuScale Power Module (NPM) fuel design uses Framatome's M5® zirconium alloy for the fuel rod cladding material.
5.3.5	NuScale requests an exemption from 10 CFR 50.61, which provides fracture toughness requirements to protect against pressurized thermal shock (PTS) events.
12.03	NuScale requests an exemption from 10 CFR 50.34(f)(2)(viii), which requires certain capabilities for post-accident sampling of the reactor coolant system and containment. The rule requires the capability to obtain and analyze samples without exceeding prescribed radiation dose limits to any individual.

Table 1.14-1: NuScale Standard Design Approval Exemptions (Continued)

FSER Section	Exemption
SDAA Part 2 3.1, 5.4, 6.4, 7.0, 7.1, 7.2, 9.4.1, 9.5.1, 9.5.2, 11.5, 12.3, 14.3 SDAA Part 4, Technical Specifications B3.3, B3.4	NuScale requests an exemption from the portion of General Design Criterion (GDC) 19 requiring the capability to achieve cold shutdown from equipment outside the control room.
SDAA Part 2 3.1.4.6, 6.3, 15.0.2, 15.6.5	NuScale requests an exemption from the requirement of 10 CFR 50.46(a)(1)(i) that requires “the most severe postulated loss-of-coolant accidents are calculated." This rule requires loss-of-coolant accidents (LOCAs) of different sizes, locations, and other properties be postulated to ensure the "most severe" LOCA is evaluated in the “acceptable evaluation model,” where LOCAs are breaks in pipes in the reactor coolant pressure boundary (RCPB). Using the acceptable evaluation model, an applicant must demonstrate specified emergency core cooling system (ECCS) performance criteria are met for the spectrum of postulated LOCAs.

1.15 Requests for Additional Information

The RAIs are questions the NRC staff asked of NuScale concerning the application. The staff sent questions to NuScale using an electronic RAI capture platform specifically created for the NuScale SDAA docket, and NuScale responded to the staff in letters submitted on the same docket. Appendix E to this FSER lists these RAIs, along with the ADAMS accession numbers.

The nomenclature for RAIs concerning SDAA Part 2 takes the following form:

- AA.BB.CC-DD, where AA.BB.CC is the section number within the review chapter, and DD is the question sequence number. In some cases, the staff may have used just the review chapter number and the question sequence number, such as 18-46.

1.16 Conclusion

As described above, the applicant supplemented the information in the initial SDAA submission by providing revisions to the document. The staff has completed its review of the SDAA, as documented throughout the FSER and, for the reasons given here, finds it to be acceptable. Additionally, the staff has confirmed that the SDAA contains design information that Subpart E, “Standard Design Approvals (SDA),” of 10 CFR Part 52, requires for a standard plant design; therefore, the staff finds the applicant’s request for an SDA acceptable.