1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

By letter dated December 31, 2016, NuScale Power, LLC (hereafter referred to as NuScale or the applicant), filed its application with the U.S. Nuclear Regulatory Commission (NRC or Commission) for certification of the NuScale standard plant design (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17013A229). By January 10, 2017, NuScale had submitted four technical reports (ADAMS Accession Nos. ML17007A001, ML17009A490 (containing two reports), and ML17010A433) and one topical report (ADAMS Accession No. ML17005A122) that allowed for the successful completion of the NRC's electronic processing of the application. The applicant submitted this application in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Subpart B, "Standard Design Certifications." The applicant submitted Revisions 1, 2, 3, and 4 of the design certification application (DCA) on March 15, 2018 (ADAMS Accession No. ML18086A090), October 30, 2018 (ADAMS Accession No. ML18311A006), August 22, 2019 (ADAMS Accession No. ML19241A315), and January 16, 2020 (ADAMS Accession ML20036D336), respectively. In addition, the applicant submitted updates to Revision 4 of the DCA on April 1, 2020 (ADAMS Accession No. ML20092L899), and May 20, 2020 (ADAMS Accession No. ML20141L787), followed by Revision 4.1 on June 19, 2020 (ADAMS Accession No. ML20198M392). On July 13, 2020, the applicant submitted a request for Standard Design Approval (SDA) based on the NuScale Standard Plant Design Certification Application (ADAMS Accession No. ML20195C766). The contents of the SDA are comprised of a subset of the DCA parts; FSAR Tier 1, Environmental Report, Emergency Plans, and Security Plans are not included in the applicant's SDA. Technical Specifications, Inspections, Tests, Analyses & Acceptance Criteria (ITAAC), and portions of Exemptions are considered supplementary information that is not specifically required by Subpart E but included in this SDA.).

Commented [A1]: When DCA Rev. 5 is received on the docket, this statement will be replaced with "*The staff's regulatory findings documented in this report are based on Revision [5] of the DCA, dated [month day, 2020] (Agencywide Document Access and Management System (ADAMS), Accession No. ML20xxxxxxx).*"

The NuScale DCA is the first application for a Small Modular Reactor (SMR). The DCA consists of 10 parts: Part 1, "General and Financial Information;" Part 2, Tier 1 and Tier 2, "Final Safety Analysis Report (FSAR);" Part 3, "Applicant's Environmental Report - Standard Design Certification"; Part 4, "Generic Technical Specifications;" Part 5, "Emergency Plans;" Part 6, "Security Plans;" 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 DCA (Docket No. 52-048) on March 23, 2017 (ADAMS Accession No. ML17074A087). NuScale design information and all other correspondence submitted before that date can be found in ADAMS under Project No. PROJ0769 or 99902043.

The applicant's design consists of up to 12 NuScale Power Modules (NPMs). As depicted in DCA Part 2, Tier 2, Figure 1.2-6, "Cutaway View of NuScale Power Module," and Figure 1.2-7, "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 160 megawatts thermal (MWt) (up to 1,920 MWt total for 12 NPMs), with approximately 50 megawatts electric (MWe) (up to 600 MWe total for 12 NPMs) output. Electrical output is dependent on environmental conditions. When considering house loads, the total net output is approximately 570 MWe for a 12-NPM facility.

The NuScale DCA Part 2 is divided into two categories, denoted as Tier 1 and Tier 2. Tier 1 means the portion of the generic design-related information that is proposed for approval and certification, including, among other things, the inspections, tests, analyses, and acceptance criteria (ITAAC). Tier 2 means the portion of the generic design-related information proposed for approval but not for certification. Tier 2 information includes, among other things, a description of the facility design required for an FSAR by 10 CFR 52.47, “Contents of applications; technical information.” NuScale DCA Part 2 contains no Tier 2* information (for a definition of this term, see Section 1.15 of this report). To evaluate the NuScale design, the NRC staff (staff) reviewed the DCA, including all referenced technical and topical reports, and generated its final safety evaluation report (FSER) on all Tier 1 and Tier 2 information. The FSER is divided into Chapters that evaluate the matching chapters in DCA 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. The DCA Part 2, Tier 1, 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 B, 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 52.53, “Referral to the Advisory Committee on Reactor Safeguards (ACRS).”

DCA Part 2, Tier 2, Section 1.1, “Introduction,” and Section 1.2, “General Plant Description,” summarize the NuScale SMR design. DCA Part 2, Tier 2, Section 1.3, “Comparison with Other Facilities,” compares the NuScale SMR design with other facilities. DCA Part 2, Tier 2, Section 1.4, “Identification of Agents and Contractors,” identifies the agents and contractors that provided design services to the applicant or other support for the design. DCA Part 2, Tier 2, Section 1.5, “Requirements for Additional Technical Information,” describes the requirements for additional technical information. DCA Part 2, Tier 2, Section 1.6, “Material Referenced,” provides an account of referenced topical reports and technical reports. DCA Part 2, Tier 2, Section 1.7, “Drawings and Other Detailed Information,” discusses the drawings and other detailed information for the NuScale SMR design. DCA Part 2, Tier 2, Section 1.8, “Interfaces with Certified Design,” addresses NuScale SMR design interfaces with certified designs and lists combined license (COL) information items in Table 1.8-2, “Combined License Information Items.” DCA Part 2, Tier 2, Chapter 1, “Introduction and General Description of the Plant,” describes eight COL items across six sections. DCA Part 2, Tier 2, Section 1.9, “Conformance with Regulatory Criteria,” describes the NuScale SMR design’s conformance with regulatory criteria. DCA Part 2, Tier 2, Section 1.10, “Nuclear Power Plants To Be Operated on Multi-Unit Sites,” provides stipulations for a COL applicant referencing the NuScale Power Plant design certification with regard to nuclear power plants to be operated on multiunit sites.

This report includes additional sections not listed in DCA Part 2 but added by the staff to clarify certain applicationwide considerations. Section 1.14, “Index of Exemptions,” identifies the exemptions listed in DCA Part 7. Section 1.15, “Index of Tier 2* Information,” identifies that DCA Part 2 contains no Tier 2* information. Section 1.16, “COL Information Items,” describes how the applicant handled COL items in DCA Part 2. Section 1.17, “Requests for Additional Information,” describes the nomenclature for RAIs discussed in the FSER.

As described above, the applicant supplemented the information in the DCA by providing revisions and updates to the document. The staff has completed its review of the DCA along with the supplements and updates, as documented throughout this report. Section 1.18, “Conclusion,” documents the staff’s overall conclusion.

The NRC staff identified three issues as not resolved within the meaning of 10 CFR 52.63(a)(5); NuScale Power has provided insufficient information regarding (1) the shielding wall design in certain areas of the plant (refer to FSER Section 12.3.4.1.2 for relevant discussions); (2) the potential for containment leakage from the combustible gas monitoring system (refer to FSER Sections 12.3.4.1.3 and 15.0.3.4.3 for relevant discussions); and (3) the ability of the steam generator tubes to maintain structural and leakage integrity during density wave oscillations in the secondary fluid system, including the method of analysis to predict the thermal-hydraulic conditions of the steam generator secondary fluid system and resulting loads, stresses, and deformations from density wave oscillations reverse flow (refer to FSER Sections 3.9.1, 3.9.2, and 5.4.1 for relevant discussions).

Graded Review Approach

The staff used a graded review approach to evaluate the NuScale DCA, 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). 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, 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 DCA review reflect the staff assessment of the applicant’s SSC categorization results provided in the DCA. The staff’s assessment includes the review of the probabilistic risk assessment, 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* (FR) 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 DCA, 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 DCA 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. The staff predicated its findings on the proprietary versions of these reports, which are those primarily referenced throughout this report, and RAI responses. However, evaluations of those reports and RAI responses described in this FSER do not contain proprietary information.

1.1.3 Combined License Applicants Referencing the NuScale Design

Applicants that reference the NuScale Power Plant design certification for specific facilities will retain architect-engineers, constructors, and consultants, as needed. As part of its review of an application for a 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 DCA Part 2, the applicant identified matters to be addressed by plant-specific COL applicants as "COL items." DCA Part 2, Tier 2, Table 1.8-2, lists COL items identified in DCA Part 2 and this FSER. The list in Table 1.8-2 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) bounded by the site parameters described in DCA Part 2, Tier 2, Chapter 2, "Site Characteristics." The NuScale Power Plant is designed to accommodate up to 12 NPMs. COL Item 1.1-1 provides that a COL applicant that references the NuScale Power Plant design certification 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 design certification 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

Appendix A to this FSER provides a chronology of the principal actions, submittals, and amendments related to the processing of the NuScale Power Plant DCA. 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.

Commented [A2]: Appendices will be published at the time of completion of the DCA review final Phase, "Phase 6 - Final SER with No Open Items," targeted for September 8, 2020.

Questions on the DCA 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 <u>General Design Description</u>

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 52.47, which requires that an applicant for certification provide an essentially complete design scope, except for site-specific elements. Therefore, the scope of the NuScale SMR design must include all of the plant SSCs that can affect the safe operation of the plant, except for its site-specific elements. DCA Part 2, Tier 2, Section 1.8, describes the NuScale SMR standard design scope, including the site-specific elements that are either partially or wholly

outside of the standard design scope. The applicant also described interfaces with the standard design (see DCA Part 2, Tier 2, Table 1.8-1, “Summary of NuScale Certified Design Interfaces with Remainder of Plant”) and representative conceptual designs.

1.2.2 Summary of the NuScale Small Modular Reactor Design

The NuScale SMR is an integrated pressurized water reactor (PWR). DCA Part 2, Tier 2, Section 1.1.4, “Power Output,” identifies the power output for the NuScale SMR.

The NuScale NSSS is a passive NuScale-designed small modular PWR. This design encompasses an integral 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 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 12 NPMs and the associated balance of plant support systems and structures. A NuScale SMR facility can consist of up to 12 NPMs that can operate within a single RXB. The information provided in DCA Part 2 includes the design of an individual NPM, as well as plant design and interfaces between the site and the design for a 12-NPM facility. However, DCA Part 2 in general describes a single NPM. DCA Part 2, Tier 2, Chapter 21, “Multi-Module Design 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 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

DCA Part 2, Tier 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 provides optimum workload management. 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

DCA Part 2, Tier 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 DCA Part 2, Tier 2, Chapter 2. COL Item 1.1-1 in DCA Part 2, Tier 2, Section 1.1.1, states, "A COL applicant that references the NuScale Power Plant design certification 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

DCA Part 2, Tier 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 are simple in design and operation and are not reliant on electrical power to fulfill their safety functions
- 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

DCA Part 2, Tier 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 steel containment operated at a vacuum, eliminating the need for insulation on the RPV
- passive safety systems that are not reliant on electrical power

DCA Part 2, Tier 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 sits 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 a standard 17x17 PWR fuel assembly with 24 guide tube locations for control rods and a central instrument tube. The only significant differences are that the fuel assembly is nominally half the height of a standard fuel assembly, and it is supported by five spacer grids. The fuel is uranium dioxide (UO_2), with gadolinium oxide (Gd_2O_3) as a burnable absorber homogeneously mixed within the fuel in select rod locations. The uranium-235 enrichment is less than 4.95 percent. DCA Part 2, Tier 2, Table 4.2-1, "Fuel Design Parameters," lists fuel 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 consists of an approximately cylindrical steel vessel with an inside diameter of approximately 2.74 meters (m) (9 feet (ft)) and an overall height of approximately 17.68 m (58 ft) that is designed for an operating pressure of approximately 12.755 megapascals (MPa) (1,850 psia). The upper and lower heads are torispherical, and the lower portion of the vessel has a flange to provide access for refueling.

The RPV consists of three sections: the RPV head section, the upper section, and the lower section. The RPV head is welded to the top of the upper section, and the upper and lower sections are flanged together using bolts.

The torispherical RPV head supports the control rod drive mechanisms (CRDMs) and includes penetrations ranging from 5.08 to 20.32 centimeters (cm) (2 to 8 in) in diameter for pressurizer spray, reactor vent valves, reactor safety valves, reactor high point degasification, I&C instrument channels, and the CRDM nozzles.

The RPV upper section is cylindrical, approximately 2.74 m (9 ft) in diameter with slightly thicker sections at the feedwater inlet and steam outlet areas. The upper section includes penetrations ranging from 5.72 to 63.5 cm (2.25 to 25 in) in diameter for the main steam piping nozzles, main steam access ports, pressurizer heaters, feedwater piping nozzles, feedwater access ports, reactor recirculation valves, CVCS, and pressure instrumentation.

The RPV lower section is cylindrical, approximately 2.74 m (9 ft) in diameter, and includes a torispherical lower head that is welded in place. The lower section of the RPV has no penetrations.

A steel pressurizer baffle plate integral with the RPV provides a barrier between the saturated water in the pressurizer and the RCS. The pressurizer baffle plate is integrated with the upper steam plenums, has flow holes to allow surges of water into and out of the pressurizer, and acts as a thermal barrier.

1.2.2.3.6 Containment Vessel

The CNV is a cylindrical steel pressure vessel housing the RPV, CRDMs, and associated NSSS piping and components. The CNV has an overall height of approximately 23.16 m (76 ft) and an outside diameter of approximately 4.57 m (15 ft) and consists of an upper CNV section with a welded torispherical top head and a lower CNV section with a welded head. The upper and lower CNV sections are flanged together using bolts. The flange connection permits the CNV to be separated to provide access to the RPV for refueling and maintenance.

The safety functions of the CNV are to contain the release of radioactive material following postulated accidents and to provide heat rejection to the reactor pool following ECCS actuation. The CNV also provides support for the RPV.

Manways provide access to components located inside the CNV. Penetrations on the CNV upper head are provided for process piping, electrical power, and instrumentation.

Support lugs located slightly below the steam plenum elevation and support skirt attached to the CNV lower head provide lateral support for the CNV. The support skirt also provides vertical support for the CNV. Internal to the CNV, the RPV is laterally and vertically supported by four support plates located slightly below the steam plenum elevation and is laterally supported at the center of the lower RPV head.

The CNV is partially immersed in the reactor pool, which provides a passive heat sink for containment heat removal. The CNV is designed to withstand the external environment of the reactor pool, as well as the internal pressure and temperature of a design-basis accident.

The CNV is maintained at a vacuum under normal operating conditions. The benefits of maintaining a vacuum in the CNV include the following:

- minimizes moisture content that could impact the reliability and contribute to corrosion of components within the CNV
- facilitates detection of leakage from the reactor coolant pressure boundary (RCPB)
- eliminates convective heat transfer and, therefore, the need for RPV insulation, which reduces potential debris generated in the CNV
- limits the initial amount of oxygen in containment (severe accident combustible gas consideration)

Following an actuation of the ECCS, steam is vented from the RPV through the reactor vent valves. 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 it has achieved an improvement in safety over existing plants through simplicity of design, reliance on passive safety systems, and small fuel inventory. The integrated 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.

The reactor core has a small radioactive source term compared to a conventional 1,000-MWe nuclear reactor. Based on the smaller fuel inventory, the amount of radioactive material available for release during a postulated accident is reduced. DCA Part 2, Tier 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

DCA Part 2, Tier 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. DCA Part 2, Tier 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 three independent reactor vent valves and two independent reactor recirculation valves (see DCA Part 2, Tier 2, Figure 1.2-9, “Emergency Core Cooling System”). All five valves are closed during normal operation.

During ECCS operation, the reactor vent valves vent steam from the RPV into the CNV, where the steam condenses and collects in the bottom of the containment. The reactor recirculation valves allow water to reenter the RPV and circulate through the core. When reactor coolant temperature is reduced to below the boiling point, core cooling continues by conduction directly into the reactor pool. The cooling function of the ECCS is entirely passive, with heat conducted through the CNV wall to the reactor pool. DCA Part 2, Tier 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 toxic or 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. DCA Part 2, Tier 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. DCA Part 2, Tier 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. DCA Part 2, Tier 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 DCA Part 2, Tier 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
- plant protection system (PPS)—monitors and controls systems that are common to all NPMs and are not specific to an individual NPM
- health physics network—provides the permanently installed communications infrastructure necessary to support a licensee-implemented radiation protection program
- 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. DCA Part 2, Tier 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 ordc electrical power for actuation.

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

- The 13.8-kilovolt (kV) and switchyard system provides power from the turbine generators and the auxiliary AC power source to the 13.8-kV AC buses and connects the onsite AC system to the switchyard.
- A medium-voltage AC electrical distribution system provides power at 4,160 volts AC to buses servicing medium voltage loads.
- A low-voltage AC electrical distribution system provides power at 120 volts AC and 480 volts AC to buses servicing low voltage loads.

- A highly reliable dc power system provides a failure-tolerant source of 125-volt dc power to plant loads including emergency lighting, MPS, PPS, and postaccident monitoring loads.

- A normal dc power system provides power to nonsafety control and instrumentation loads.

- Backup power is provided for onsite AC power. The backup diesel generators provide power at the 480-volt AC level and the auxiliary AC power source provides power at the 13.8-kV AC level.

1.2.5 Power Conversion System

The power conversion systems associated with an NPM consist of a main steam system, a turbine generator set, a standard condenser and cooling tower arrangement, and a condensate and feedwater system, as shown in DCA Part 2, Tier 2, Figure 1.2-3, “Schematic of a Single NuScale Power Module and Associated Secondary Equipment.” With multiple NPMs at each plant, individual NPMs can be placed into service incrementally to meet construction schedules and grid demand as permitted by a potential future site license. NPMs can also be taken off line individually for refueling outages and maintenance.

1.2.6 Fuel Handling and Storage Systems

The fuel handling and reactor maintenance areas are located in the west end of the RXB and include space for the spent fuel pool (SFP), refueling pool, and dry dock. DCA Part 2, Tier 2, Figure 1.2-16, “Reactor Building 100'-0" Elevation,” shows the pools.

1.2.7 Plant Cooling Water Systems

The plant cooling water systems include several 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. The reactor component cooling water system provides cooling to the CRDMs, the nonregenerative heat exchangers for each CVCS, and the primary sampling system coolers (DCA Part 2, Tier 2, Section 9.2.2, “Reactor Component Cooling Water System”).

- The reactor pool cooling system and the SFP cooling system are nonsafety-related, closed-loop systems that transfer heat from the associated pool to the site cooling water system (DCA Part 2, Tier 2, Section 9.1.3, “Spent Fuel Pool Cooling and Cleanup System”).

- The circulating water system is an open-loop system that provides a continuous supply of cooling water to the plant turbine condensers. Circulating water pumps draw water from a common basin to provide cooling water flow for up to six condensers in the turbine generator building. Heated circulating water from the outlet of the condensers flows to a set of mechanical-draft cooling towers where excess heat is removed as the water gravity flows back to the common basin (DCA Part 2, Tier 2, Section 10.4.5, “Circulating Water System”).

- The site cooling water system is an open-loop system that provides a continuous supply of cooling water to the chilled water system, the balance of plant component cooling water system, the SFP cooling system, the reactor pool cooling system, the reactor component cooling water system, and the condenser air removal system. Site cooling water pumps draw water from a common basin to provide cooling water flow to the systems serviced. Heated site cooling water from the outlet of the individual system heat exchangers continues to a dedicated set of mechanical-draft cooling towers where excess heat is removed as the water flows via gravity back to the common basin (see DCA Part 2, Tier 2, Section 9.2.7, "Site Cooling Water System").

1.2.8 Radioactive Waste Management System

DCA Part 2, Tier 2, Chapter 11, "Radioactive Waste Management," discusses the radioactive waste management system in detail. DCA Part 2, Tier 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. DCA Part 2, Tier 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

DCA Part 2, Tier 2, Figure 1.2-2, "NuScale Functional Boundaries," presents the layout of a NuScale Power Plant. DCA Part 2, Tier 2, Section 1.2.2, "General Arrangement of Major Structures and Equipment," describes the following structures and equipment:

- Reactor Building (RXB)—DCA Part 2, Tier 2, Figure 1.2-5, "Cutaway Illustration of 12 Module Configuration," and Figures 1.2-10 through 1.2-20 provide RXB drawings. The RXB houses the NPMs and systems and components required for plant operation and shutdown. The RXB is a seismic Category I, reinforced concrete structure with design considerations for the effects of aircraft impact, environmental conditions, postulated design-basis accidents (internal and external), and design-basis threats. The RXB also provides radiation protection to plant operations and maintenance personnel. Each NPM is in the common reactor pool in its own three-walled bay with the open wall facing the center of the pool. The bays are arranged into two rows with six bays per row along the north and south walls of the reactor pool at the east end of the pool. A central channel between the bays allows for movement of the NPMs between the bays and the refueling pool. The NPM, reactor pool, and SFP are below grade. The surface of the reactor pool water is approximately 1.83 m (6 ft) below grade. Also located below grade are most primary systems and some radioactive waste equipment. Hoisting and handling equipment is located above grade. Pipe fittings and electrical connections are provided above the reactor pool water level to permit manual connection and disconnection during NPM installation, refueling outages, and replacement or removal of NPMs.
  - Fuel handling and reactor maintenance areas are in the west end of the RXB and include space for the SFP, refueling pool, and dry dock. DCA Part 2, Tier 2, Figure 1.2-16, shows the pools. The refueling pool is connected directly to the reactor pool, accommodating transport of an NPM through the pool water using

the RXB crane. 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. Spent fuel assemblies removed from the reactor core are placed in spent fuel storage racks in the SFP.

– Refueling operations for an individual NPM are independent of the operating status of the remaining NPMs. During refueling, an NPM is moved from its operating bay in the reactor pool to the refueling pool using the RXB crane. In the refueling area, the NPM is set into the containment flange tool where the CNV flange is unbolted. The crane lifts the NPM, separating the lower CNV from the upper CNV with the RPV still attached and intact. Next, the crane moves the upper CNV and RPV to the reactor vessel flange tool where the RPV flange is unbolted. The crane again lifts the NPM, this time separating the upper and lower RPV, leaving the lower RPV including the reactor core in the reactor vessel flange tool. Finally, the crane transports the upper NPM (now consisting of just the upper CNV with attached upper RPV) to the module inspection rack in the dry dock. Inspection, testing, and maintenance are performed while the core is being refueled with a dedicated fuel handling machine.

- Control Building (CRB)—DCA Part 2, Tier 2, Figures 1.2-21 through 1.2-27 show that the CRB is approximately 9.14 m (30 ft) east of the RXB. The MCR and the associated spaces are below grade in the CRB. The technical support center and the associated spaces are at grade level 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. Each reactor operator monitors and controls multiple NPMs from a control room panel. DCA Part 2, Tier 2, Figure 18.7-1, “NuScale Main Control Room Layout,” provides the layout for the MCR. 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.

  – A technical support center is provided, compliant with the design requirements of NUREG-0696, “Functional Criteria for Emergency Response Facilities,” issued February 1981 (ADAMS Accession No. ML051390358). DCA Part 2, Tier 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 DCA Part 2, Tier 1, Figures 1.2-28 through 1.2-33 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 DCA Part 2, Tier 2, Figure 1.2-8, “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 large, stainless steel-lined, reinforced concrete pool located in the RXB below plant grade level. The UHS consists of the reactor pool area, the refueling pool area, and the SFP area. DCA Part 2, Tier 2, Figure 1.2-16 shows the pool areas. During normal plant operations, heat is removed from the pool through the reactor pool cooling system and rejected into the atmosphere through a cooling tower or other external heat sink. The SFP has an independent SFP cooling 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.

- The CVCS is simple in design and its operation is not credited during or after an accident. 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. DCA Part 2, Tier 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

DCA Part 2, Tier 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 DCA Part 2, Tier 2, Chapter 2.

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

1.4.2 General Plant Description

The staff took the plant, system, and component descriptions in this section in large part from the summary description in DCA Part 2, Tier 2, Chapter 1.

DCA Part 2, Tier 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; and auxiliary systems (e.g., compressed air, nonradioactive drains, water systems).

NuScale stated that each COL applicant will develop an FSAR that incorporates by reference DCA Part 2. DCA Part 2 includes COL items that identify where site-specific information must be provided. However, in some instances, representative information is necessary to provide context for interfaces with the standard design. This representative or conceptual design information (CDI) is outside the scope of the NuScale Power Plant certified design. Where provided, CDI is delineated in the DCA by double brackets ([[ ]]). DCA Part 2, Tier 2, Figure 1.2-2, shows the scope of the certified design and site-specific design. DCA Part 2, Tier 2, Figure 1.2-3, shows the basic systems associated with power generation. NuScale stated that although some components from these systems are physically located in buildings that are CDI, the system itself is not, except for the clouded portion, which identifies the CDI cooling towers and certain circulating water systems. 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

DCA Part 2, Tier 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 outside the security owner-controlled area fence.

The NuScale Power Plant is designed for 1 to 12 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, and shared external cooling water systems (shown in DCA Part 2, Tier 2, Figure 1.2-3), 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 DCA Part 2, Tier 2, Figures 1.2-1 and 1.2-2):

- <u>RXB</u>—located above and below grade, houses the following facilities, and is described in further detail in DCA Part 2, Tier 2, Section 1.2.2.1, “Reactor Building”:
    - UHS (reactor pool, refueling pool, and SFP)
    - fuel handling areas
    - primary systems
- <u>CRB</u>—located above and below grade, adjacent to the RXB, provides space for the following facilities and is described further in DCA Part 2, Tier 2, Section 1.2.2.2, “Control Building”:
    - MCR—located below grade, houses the equipment, controls, and indications for operation of the NPMs

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

- RWB—located above and below grade, provides space for HVAC equipment and radioactive waste treatment and storage equipment, and is described further in DCA Part 2, Tier 2, Section 1.2.2.3, "Radioactive Waste Building"

The applicant discussed the following structures as CDI (see DCA Part 2, Tier 2, Figures 1.2-1 and 1.2-2):

- Turbine Generator Buildings—house the turbine generators and associated equipment, as described further in DCA Part 2, Tier 2, Section 1.2.2.5.1, "Turbine Generator Building"

- Annex Building—controls access into the radiologically-controlled area and provides space for health physics facilities; servicing potentially radioactive and nonradioactive tooling, fixtures, and instrumentation; and conducting security services and various personnel services, as described further in DCA Part 2, Tier 2, Section 1.2.2.5.2, "Annex Building"

- Security Buildings—provide for controlled access into the security owner-controlled area and the protected area of the plant, as described further in DCA Part 2, Tier 2, Section 1.2.2.5.3, "Security Buildings"

- Central Utility Building—houses various equipment for the chilled water system and other ancillary equipment for balance of plant systems, as described further in DCA Part 2, Tier 2, Section 1.2.2.5.4, "Central Utility Building"

- Diesel Generator Buildings—house the backup diesel generators and associated equipment, as described further in DCA Part 2, Tier 2, Section 1.2.2.5.5, "Diesel Generator Buildings"

- Site Cooling Water System—provides cooling water to plant auxiliary systems; the details associated with location and orientation of the cooling towers as well as equipment design and operation are site specific, as described further in DCA Part 2, Tier 2, Section 1.2.1.6, "Plant Cooling Water Systems"

1.4.3 Facility Description

DCA Part 2, Tier 2, Section 1.2.1.1, "Facility Description," states that the reactor core is located in a core support assembly, which is seated in the lower RPV assembly. A central hot-leg riser is connected to the top of the core support assembly. 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 DCA Part 2, Tier 2, Figure 1.2-7.

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 the utility grid through a step-up transformer. A turbine bypass line provides up to 100 percent of the rated main steam flow directly from the associated SGs to the main condenser in a controlled manner to remove heat from the reactor following a load reduction or loss of electrical load. Steam that exits or bypasses the turbine is directed to the condenser. A shared circulating water loop removes heat and condenses the steam for up to six condensers. The condensate is pumped through condensate polishing equipment to the inlet of the variable speed feedwater pumps.

1.4.3.1 NRC Staff Review of Multimodule Design Considerations

1.4.3.1.7 Summary of NRC Staff Technical Evaluation

FSER Chapter 21, “Multi-Module Design Considerations,” identifies sections of other FSER chapters in which the staff has evaluated the interactions of systems shared among multiple NPMs and documented the findings. FSER Table 21-1, “NuScale Standard Design Shared Systems Evaluated by NRC Staff,” lists the systems shared among multiple NPMs and the SER sections that document the staff’s evaluation of the interactions of these systems.

FSER Chapter 15, “Transient and Accident Analysis,” considers and evaluates the failure of shared systems that are not safety related within the NuScale transient and accident analyses. FSER Section 15.0.0, “Classification and Key Assumptions,” contains the staff’s review of the categorization and classification of NuScale’s DBEs. FSER Section 15.0.3, “Radiological Consequences of Design Basis Accidents,” contains the staff’s review of the radiological consequences of the design-basis accidents.

FSER Section 8.3.1, “Alternating-Current Power Systems,” and FSER Section 8.3.2, “Direct Current Power Systems,” address multimodule design aspects of the NuScale SMR design as related to electrical power.

FSER Section 7.2.11, “Multi-Unit Stations,” documents the staff’s evaluation of I&C systems for multimodules. FSER Section 9.2.5, “Ultimate Heat Sinks,” documents the staff’s evaluation of NuScale UHS safety function for a limiting 12-NPM heat load.

FSER Section 19.1.4.9, “Evaluation of Multimodule Risk,” documents the staff’s discussion of multimodule risk, including internal and external events.

1.4.3.2 Applicability of Topical Report TR-0815-16497-P-A, “Safety Classification of Passive Nuclear Power Plant Electrical Systems”

1.4.3.2.1 General

DCA Part 2, Tier 2, Table 1.6-1, “NuScale Referenced Topical Reports,” references Topical Report TR-0815-16497-P-A, “Safety Classification of Passive Nuclear Power Plant Electrical Systems,” dated February 23, 2018 (ADAMS Accession No. ML18054B608 (Proprietary version); ML18054B607 (public version)). As part of the DCA, the applicant must describe how the NuScale design meets the following:

- the limitations and conditions described in TR-0815-16497-P-A, Section A, “Limitations and Conditions”

- the conditions of applicability described in TR-0815-16497-P-A, Section B, “Conditions of Applicability,” Table 3-1, Sections I and II

- the augmented provisions described in TR-0815-16497-P-A, Section B, Table 3-2

DCA Part 2, Tier 2, Chapter 8, “Electric Power,” Table 8.3-9, “FSAR Cross Reference for the Conditions of Applicability and NRC SER Limitations and Conditions for TR-0815-16497-P-A,” and Table 8.3-10, “FSAR Cross Reference for the EDSS Augmented Provisions in TR-0815-16497-P-A,” highlight the portions of DCA Part 2 that address the information necessary for referencing TR-0815-16497-P-A. The following sections of this SER provide the staff evaluations of information addressing the limitations and conditions necessary for referencing TR-0815-16497-P-A.

The information that NuScale provided can be found in Revision 4 to the DCA, dated January 16, 2020, (ADAMS Accession No. ML20036D336), and in a letter dated March 27, 2018 (ADAMS Accession No. ML18086B096).

1.4.3.2.2 Limitations and Conditions

TR-0815-16497-P-A, Section A, describes five limitations and conditions. The following sections discuss the staff’s dispositions of each.

<u>Condition 4.1</u>:

> Address the guidance in RG 1.155, Appendix A, in sufficient detail to enable the NRC staff to verify that the relevant QA program would meet or exceed the guidance in RG 1.155.

<u>Disposition of Condition 4.1</u>:

Because DCA Part 2, Tier 2, Section 8.3.2.2.2, “Onsite Direct Current Power System Conformance with Regulatory Framework,” states that an augmented quality assurance (QA) program is applied to the EDSS and the program meets the QA provisions of Regulatory Guide (RG) 1.155, “Station Blackout,” the staff finds that a relevant QA program would meet or exceed the guidance in RG 1.155, Appendix A, “Quality Assurance Guidance for Non-Safety Systems and Equipment.” The staff finds the disposition acceptable because DCA Part 2 addresses the QA provisions of RG 1.155, Appendix A. Therefore, the NRC staff finds that the applicant met Condition 4.1. FSER Chapter 8, “Electric Power,” documents the staff’s evaluation supporting this conclusion.

<u>Condition 4.2</u>:

> Confirm that the valve-regulated lead-acid (VRLA) batteries and their structures are seismic Category 1. To provide reasonable assurance that the VRLA batteries will perform as intended, an applicant that references the TR shall provide a COL action item to support that the VRLA batteries and their structures are seismic Category 1. A qualification testing plan includes environmental and seismic qualification and a technical functional requirement for VRLA batteries to show they can perform as intended.

Disposition of Condition 4.2:

Because DCA Part 2, Tier 2, Table 3.2-1, states that the highly reliable dc power system (identified as EDSS) is classified as seismic Category I, and DCA Part 2, Tier 2, Section 8.3.2.1.1, “Highly Reliable Direct Current Power System,” states that all EDSS equipment is designed to seismic Category I standards, the staff finds that the applicant has confirmed that the batteries are seismic Category I, and a COL action item is not required, as the information is contained in DCA Part 2. DCA Part 2, Tier 2, Section 8.3.2.2.2, states that the EDSS design accommodates the effects of environmental conditions by using a graded approach to applying augmented provisions for the design, qualification, and QA typically applied to Class 1E dc power systems. Furthermore, DCA Part 2, Tier 2, Section 8.3.2.1.1, states that qualification provisions are applied to the EDSS. The staff finds the disposition acceptable because DCA Part 2 addresses qualification provisions as well as seismic Category I classification for the EDSS. Therefore, the NRC staff finds that the applicant met Condition 4.2. FSER Chapter 8 documents the staff’s evaluation supporting this conclusion.

Condition 4.3:

> Demonstrate that operator actions are not necessary to ensure the performance of safety related functions for any postulated DBE (i.e., the design does not include Type A variables as defined in IEEE Std. 497-2002, as modified in RG 1.97, Regulatory Position C.4), as presented in Chapter 15 of its DCA Part 2 and the human factors analysis in Chapter 18 of its DCA Part 2.

Disposition of Condition 4.3:

The disposition of Condition 4.3 of the NRC staff’s SER for TR-0815-16497 is provided in DCA Part 2, Tier 2, Section 7.1.1.2.2, “Post-Accident Monitoring,” Section 15.0.0.6.4, “Required Operator Actions,” and Section 18.6.2.2, “Deterministically Important Human Actions.” The NRC staff reviewed the disposition and found it acceptable because the NuScale design has no Type A variables, as defined in the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 497-2002, as modified in RG 1.97, Revision 5, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” Regulatory Position C.4 (issued April 2019). The NRC staff concluded that there are no operator actions credited in any DCA Part 2, Tier 2 Chapter 15 anticipated operational occurrence (AOO), infrequent event, or accident analysis, nor the station blackout or anticipated transient without scram analysis. FSER Chapter 7 documents the NRC staff’s evaluation following the guidance of RG 1.97 for the variable selection and found that no Type A variables are required. FSER Chapter 15 documents the NRC staff’s evaluation for the required operator actions.

In addition, the NRC staff has not identified any deterministically important human actions to ensure reactivity control, core heat removal, or containment isolation and integrity. The only important human actions modeled in the probabilistic risk assessment are in response to multiple failures of automatic safety systems, and the actions are not required to meet the assumptions of any accident analysis licensing basis. FSER Chapter 18, “Human Factors Engineering,” documents the NRC staff’s evaluation of the deterministically important human actions. Therefore, the staff finds that the applicant met Condition 4.3.

Condition 4.4:

> Evaluate the frequency for which a combination of an AOO and an actuation of the NuScale ECCS is realistically expected to occur, and show that such a combination of events is not expected to occur during the lifetime of the module.

<u>Disposition of Condition 4.4</u>:

DCA Part 2, Tier 2, Table 8.3-9, references DCA Part 2, Tier 2, Section 15.0.0.6.3, “Engineered Safety Features Characteristics,” to address Condition 4.4. DCA Part 2, Tier 2, Section 15.0.0.6.3, clarifies that, even though some AOOs and infrequent events result in ECCS actuation under conservative DCA Part 2, Tier 2, Chapter 15 assumptions, the applicant conducted a realistic analysis of these events, which showed that ECCS actuation in response to an AOO or infrequent event is expected to occur much less than once in the lifetime of an NPM. Based on the results of the applicant’s analysis, the NRC staff finds that the applicant met Condition 4.4.

<u>Condition 4.5</u>:

> Demonstrate that the reactor can be brought to a safe shutdown using only safety-related equipment in the absence of electrical power following a DBE, with margin for stuck rods. Alternatively, an applicant addressing this condition may provide justification, for NRC review, for a less restrictive approach.

<u>Disposition of Condition 4.5</u>:

DCA Part 2, Tier 2, Table 8.3-9, references DCA Part 2, Tier 2, Section 4.3.1.5, “Shutdown Margin and Long Term Shutdown Capability,” Section 9.3.4.3, “Safety Evaluation,” and Section 15.0.6, “Evaluation of a Return to Power,” to address Condition 4.5. DCA Part 2, Tier 2, Section 4.3.1.5, provides the design basis for long-term shutdown capability, which the applicant defined as the amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all CRAs are fully inserted and the RCS is cooled to equilibrium conditions. DCA Part 2, Tier 2, Section 15.0.6, evaluates a scenario in which a CRA fails to insert and the CVCS is unavailable, resulting in the reactor core returning to a low power level. As described in the SER for TR-0815-16497, the regulatory basis for Condition 4.5 is 10 CFR Part 50, Appendix A, GDC 27, “Combined Reactivity Control Systems Capability.” DCA Part 7 includes a request for exemption from GDC 27. Additionally, DCA Part 7, Section 15.3.1 states the special circumstances for the exemption as (1) the application of the stuck rod assumption as part of the NuScale design basis for shutdown is not necessary to meet the underlying purpose of GDC 27 because the NuScale design can maintain a safe, stable condition with a stuck rod in the long term following an accident, and (2) an exemption to GDC 27 would result in a benefit to the public health and safety that compensates for any decrease in safety that may result from the exemption. FSER Section 15.0.6 documents the NRC staff’s evaluation of the GDC 27 exemption request, which includes an analysis of the special circumstances required by 10 CFR 50.12, “Specific exemptions.” The staff concludes in FSER Section 15.0.6 that consistent with the requirements in 10 CFR 50.12(a), the proposed exemption requested in DCA Part 7, Section 15, regarding the requirement stated in GDC 27 is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, the special circumstances in 10 CFR 50.12(a)(2)(ii) are present, in that the application of GDC 27 in the particular circumstances is not necessary to achieve the underlying purpose of this rule. Based on the staff’s analysis of the exemption requested in DCA Part 7, and the supporting information in DCA Part 2, Tier 2, Sections 4.3.1.5 and 15.0.6, the NRC staff finds that the applicant’s response addresses Condition 4.5.

1.4.3.2.3 Conditions of Applicability

TR-0815-16497-P-A, Section B, describes the conditions of applicability. In this section, the staff dispositions the conditions of applicability. The staff has delineated proprietary information with double brackets "**[[ ]]**."

TR-0815-16497-P-A, Table 3-1, "Conditions of Applicability," Section I:

1. **[[**

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a. **[[ ]]**

Disposition of Condition I.1.a:

The applicant provided the disposition of Condition I.1.a in DCA Part 2, Tier 2, Section 3.9.4, "Control Rod Drive System," and Section 8.4.2, "Station Blackout Analysis and Results." Specifically, DCA Part 2, Tier 2, Section 3.9.4.1.1, "Control Rod Drive Mechanism," states that the CRA is released and inserted by gravity into the core if electrical power is interrupted. Additionally, FSAR Section 4.6.2 clarifies that the control rod drive system (CRDS), which includes the control rod drive mechanisms and all electrical and instrumentation and controls components, facilitates reliable control by performing a reactor trip via gravity-dropping of the CRAs on a reactor trip signal or loss of power. The physical operation of the control rod drive mechanisms (CRDMs) (including the non-reliance on electrical power) is further evaluated in FSER Section 3.9.4, "Control Rod Drive Systems," and the overall evaluation of the CRDS (including the initiation of a trip on a loss of electrical power) is provided in FSER Section 4.6. Based on the design of the CRDS as described in DCA Part 2, Tier 2, Section 3.9.4 and Section 4.6, the NRC staff finds that reactor trip is assured in the absence of electrical power because the interruption of electrical power results in the insertion of the CRAs into the core by gravity. Therefore, the NRC staff finds that the applicant met Condition I.1.a.

b. **[[**

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Disposition of Condition I.1.b:

The applicant provided the disposition of Condition I.2.b in DCA Part 2, Tier 2, Section 4.3.1.5, "Shutdown Margin and Long Term Shutdown Capability," Section 8.4.2, "Station Blackout Analysis and Results," and Section 15.6, "Decrease in Reactor Coolant Inventory." The NRC staff reviewed the disposition and found it acceptable because the referenced analyses do not credit electrical power for event mitigation. Additionally, DCA Part 7 includes a

request for exemption from GDC 27, which is associated with the [negative reactivity] requirement from Condition I.1.b. FSER Section 15.0.6 documents the NRC staff's evaluation of the GDC 27 exemption request.

c. **[[**

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Disposition of Condition I.1.c:

The applicant provided the disposition of Condition I.1.c in DCA Part 2, Tier 2, Section 5.4.3.1, "Design Basis"; Section 6.3.1, "Design Basis"; Section 8.4.2; Section 15.0.0.6.3; Table 15.0-2, "Acceptance Criteria—Thermal Hydraulic and Fuel"; Table 15.0-3, "Acceptance Criteria Specific to Rod Ejection Accidents"; and Table 15.0-4, "Acceptance Criteria Specific to Loss of Coolant Accidents." DCA Part 2, Tier 2, Section 15.0.0.6.3, states that the ECCS and DHRS valves do not rely on electrical power or on not safety-related support systems for actuation. Additionally, DCA Part 2, Tier 2, Section 15.0.0.6.5, "Availability of Offsite Power," clarifies that neither AC nor dc power systems are credited to mitigate the events in DCA Part 2, Tier 2, Chapter 15. Based on the ECCS and DHRS nonreliance on electrical power, and the fact that the design basis for the NPM does not credit electrical power to mitigate events in DCA Part 2, Tier 2, Chapter 15, the NRC staff finds that the applicant met Condition I.1.c. FSER Chapter 15 documents the NRC staff's evaluation of the ECCS and DHRS performance for limiting DBEs.

d. **[[ ]]**

Disposition of Condition I.1.d:

The applicant provided the disposition of Condition I.1.d in DCA Part 2, Tier 2, Section 6.2.4.2.1, "General Description" (of the containment isolation system), and Section 8.4.3, "Station Blackout Coping Equipment Assessment." The NRC staff reviewed the disposition and found it acceptable because containment isolation is achieved and maintained in the absence of Class 1E electrical power. Therefore, the NRC staff finds that the applicant met Condition I.1.d. FSER Chapter 6, "Engineered Safety Features," documents the NRC staff's evaluation of containment isolation performance.

e. **[[**

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Disposition of Condition I.1.e:

The applicant provided the disposition of Condition I.1.e in DCA Part 2, Tier 2, Section 6.2.1, "Containment Functional Design," Section 6.2.2, "Containment Heat Removal," Section 6.2.5.1, "Design Bases," Section 8.4.3, and Table 15.0-2 (i.e., acceptance criteria confirm containment peak pressure less than design pressure). The NRC staff reviewed the disposition and found it acceptable because containment integrity is achieved and maintained in the absence of Class 1E electrical power. Therefore, the NRC staff finds that the applicant met

Condition I.1.e. FSER Chapters 6, 8, and 15 document the NRC staff's evaluation supporting this conclusion.

f. **[[**

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Disposition of Condition I.1.f:

The applicant provided the disposition of Condition I.1.f in DCA Part 2, Tier 2, Section 6.5.3, "Fission Product Control Systems," and Table 15.0-12, "Radiological Dose Consequences for Design Basis Analyses." The NRC staff reviewed the disposition and found it acceptable because active fission product control and removal systems are not needed to maintain offsite doses within applicable guidelines. Active control and removal are not needed because the containment vessel passively removes fission products by its inherent natural aerosol removal mechanisms, which include thermophoresis, diffusiophoresis, hygroscopicity, and sedimentation. Therefore, the NRC staff finds that the applicant met Condition I.1.f. FSER Chapters 6 and 15 document the NRC staff's evaluation supporting this conclusion.

g. **[[** **]]**

Disposition of Condition I.1.g:

The applicant provided the disposition of Condition I.1.g in DCA Part 2 Tier 2, Section 5.2.2.1, Section 8.4.2, and Table 15.0-2. The NRC staff finds this response acceptable because it references the design basis and analysis acceptance criteria associated with overpressure protection, and the referenced analyses take no credit for electrical power. Accordingly, the NRC staff finds that the applicant addressed Condition I.1.g. FSER Section 5.2.2 and Chapter 15 document the NRC staff's evaluation of the overpressure protection design basis and transient analyses, respectively.

2. **[[**

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a. **[[**

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Disposition of Condition I.2.a:

The disposition of Condition I.2.a is provided in DCA Part 2, Tier 2, Section 7.1.1.2.2, "Post-Accident Monitoring," and Section 8.4.2, "Station Blackout Analysis and Results." The NRC staff reviewed the disposition and found it acceptable because there are no operator actions credited in the evaluation of NuScale DBEs. After a DBE, automated actions place the NPM in a safe state, and the NPM remains in the safe-state condition for at least

72 hours without operator action. FSER Section 7.2.13.4.1, “Compliance with IEEE Std. 603-1991, Section 5.8.1,” documents the NRC staff’s evaluation of postaccident monitoring instrumentation. Therefore, the NRC staff finds that the applicant met Condition I.2.a.

b. **[[**

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Disposition of Condition I.2.b:

The disposition of Condition I.2.b is provided in DCA Part 2, Tier 2, Section 7.1.1.2.2. The NRC staff reviewed the disposition and found it acceptable because there are no postaccident monitoring Type A variables for the NuScale design, and all required protective actions by the MPS are automatic. In addition, no operator actions are credited in any DCA Part 2, Tier 2, Chapter 15 AOO, infrequent event, or accident analysis, nor the station blackout or anticipated transient without scram analysis. FSER Section 7.2.13, “Displays and Monitoring,” documents the NRC staff’s evaluation of the displays and monitoring systems. Therefore, the NRC staff finds that the applicant met Condition I.2.b.

c. **[[**

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Disposition of Condition I.2.c:

The disposition of Condition I.2.c is provided in DCA Part 2, Tier 2, Section 7.1.1.2.2. The NRC staff reviewed the disposition and found it acceptable because the NuScale design does not rely upon Type B and Type C variables for the performance of operator actions in response to a DBE. FSER Section 7.2.13 documents the NRC staff’s evaluation of the displays and monitoring systems. Therefore, the NRC staff finds that the applicant met Condition I.2.c.

3. **[[**

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Disposition of Condition I.3:

In DCA Part 2, Tier 2 Table 8.3-9, “FSAR Cross Reference for the Conditions of Applicability and NRC SER Limitations and Conditions for TR-0815-16497-P-A,” the applicant indicated that the disposition of Condition I.3 is demonstrated by the descriptions provided in DCA Part 2, Tier 2, Section 9.1.3, “Spent Fuel Pool Cooling and Cleanup System,” and Section 9.2.5, “Ultimate Heat Sink.” The

NRC staff reviewed the disposition discussed in Sections 9.1.3 and 9.2.5 and found it acceptable because the NuScale design does not credit electrical power or operator action in response to a DBE. FSER Section 9.2.5 addresses the NRC staff's evaluation of the cooling of the fuel assemblies during a DBE. Therefore, the NRC staff finds that the applicant met Condition I.3.

4. **[[**

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Disposition of Condition I.4:

In DCA Part 2, Tier 2 Table 8.3-9, the applicant indicated that the disposition of Condition I.4 is demonstrated by the descriptions provided in DCA Part 2, Tier 2, Sections 9.1.3 and 9.2.5. The NRC staff reviewed the disposition and found it acceptable because the NuScale design does not credit electrical power or operator action in response to a DBE. FSER Section 9.2.5 documents the NRC staff's evaluation of the cooling of the reactor fuel assemblies in the process of core unloading and loading during a DBE. Therefore, the NRC staff finds that the applicant met Condition I.4.

5. **[[**

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Disposition of Condition I.5:

In DCA Part 2, Tier 2, Table 8.3-9, the applicant dispositions Condition I.5 in DCA Part 2, Tier 2, Sections 6.4, and 8.4.3. The NRC staff reviewed the disposition and found it acceptable because control room habitability is actuated and maintained for a minimum of 72 hours in the absence of Class 1E electrical power. Therefore, the NRC staff finds that the applicant met Condition I.5. FSER Chapter 6, "Engineered Safety Features," and Chapter 8, "Electric Power," document the NRC staff's evaluation supporting this conclusion.

6. **[[**

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Disposition of Condition I.6:

In DCA Part 2, Tier 2, Table 8.3-9, the applicant dispositions Condition I.6 in DCA Part 2 Tier 2, Section 3.11.2.1, "Environmental Qualification of Electrical Equipment" (postaccident monitoring environmental qualification); Section 3.11.4,

"Loss of Ventilation" (72-hour loss of ventilation); Table 3C-3, "Designated Mild Environment Areas," (EDSS environment); and Sections 8.4.2, and 8.4.3. The NRC staff reviewed the disposition and found it acceptable because requirements in 10 CFR 50.63, "Loss of All Alternating Current Power," are met for a minimum of 72 hours in the absence of Class 1E electrical power. Therefore, the NRC staff finds that the applicant met Condition I.6. FSER Chapter 3, "Design of Structures, Systems, Components and Equipment," and FSER Chapter 8 document the NRC staff's evaluation supporting this conclusion.

7. **[[**

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Disposition of Condition I.7:

In DCA Part 2, Tier 2, Table 8.3-9, the applicant dispositions Condition I.7 in DCA Part 2, Tier 2, Section 6.5.3, "Fission Product Control Systems," Section 9.4.2, "Reactor Building and Spent Fuel Pool Area Ventilation System," Section 9.4.5, "Engineering Safety Feature Ventilation System," Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," and Table 15.0-12. The NRC staff reviewed the disposition and found it acceptable because active ventilation or fission product removal systems are not needed to maintain offsite doses within applicable guidelines. Therefore, the NRC staff finds that the applicant met Condition I.7. FSER Chapter 6, Chapter 9 "Auxiliary Systems," and Chapter 15 document the NRC staff's evaluation supporting this conclusion.

TR-0815-16497-P-A, Table 3-1, "Conditions of Applicability," Section II:

1. **[[**

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Disposition of Condition II.1:

DCA Part 2, Tier 2, Section 8.3.2.1.1, "Highly Reliable Direct Current Power System," states that the EDSS includes augmented design provisions for batteries. Furthermore, DCA Part 2, Tier 2, Section 8.3.2.2.2, states that the EDSS design accommodates the effects of environmental conditions by using a graded approach to apply augmented provisions for the design, qualification, and QA typically applied to Class 1E dc power systems. Further, DCA Part 2, Tier 2, Section 8.3.2.1.1, states that qualification provisions are applied to the EDSS. The NRC staff reviewed the disposition and found it acceptable because augmented design, qualification, and QA provisions are applied to the EDSS.

Therefore, the NRC staff finds that the applicant met Condition II.1. FSER Chapter 8 documents the NRC staff's evaluation supporting this conclusion.

2. **[[**

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Disposition of Condition II.2:

DCA Part 2 Tier 2, Section 8.3.2.1.1 states that an evaluation of EDSS reliability was performed using the methodology described in Condition of Applicability II.2. DCA Part 2, Tier 2, Table 8.3-7, "Highly Reliable Direct Current Power System Failure Modes and Effects Analysis," evaluates the EDSS component failures, and DCA Part 2, Section 8.3.2.1.1, states that the results show that failures do not prevent safety-related functions from being achieved and maintained. Furthermore, DCA Part 2, Tier 2, Section 8.3.2.1.1, states that an evaluation of the EDSS reliability was performed and, using the generic failure probabilities in DCA Part 2, Tier 2, Section 19.1.4.1.1.5, "Data Sources and Analysis," the EDSS supports the mission requirements of high reliability. The NRC staff reviewed the disposition and found it acceptable because a failure modes and effects analysis was performed to evaluate EDSS reliability. Therefore, the NRC staff finds that the applicant met Condition II.2. FSER Chapter 8 and Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation," document the NRC staff's evaluation supporting this conclusion.

3. **[[**

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Disposition of Condition II.3:

DCA Part 2, Tier 2, Section 9.5.3.2, describes the provisions for emergency lighting; FSER Section 9.5.3 documents the staff's review of the plant lighting systems.

1.4.3.2.4 Augmented Provisions

The staff evaluated DCA Part 2, Tier 2, Table 8.3-10, and verified that DCA Part 2, Tier 2, Chapter 3, "Design of Structures, Systems, Components and Equipment," and Chapter 8, address the augmented provisions. FSER Chapter 8 addresses and grants the exemption requests from GDC 17, "Electric Power Systems," and GDC 18, "Inspection and Testing of Electric Power Systems," with regard to the safety classification, independence, single-failure criterion, common-cause failure, protection, surveillance and testing of the EDSS, and multiunit considerations.

1.4.3.2.5 Conclusion

DCA Part 2, Tier 2, Table 1.6-1, references TR-0815-16497-P-A. In particular, DCA Part 2 Tier 2, Chapter 8, Table 8.3-9 and Table 8.3-10 identify the portions of DCA Part 2 that address information required for referencing TR-0815-16497-P-A. Based on the staff’s review of the DCA, the staff finds the information acceptable because the DCA adequately describes how the NuScale design meets (1) the limitations and conditions described in TR-0815-16497-P-A, Section A, (2) the conditions of applicability described in TR-0815-16497-P-A, Section-B (Table 3-1, Sections I and II), and (3) the augmented provisions described in TR-0815-16497-P-A, Section B (Table 3-2).

1.5 Comparison with Other Facilities

DCA Part 2, Tier 2, Table 1.3-1, “NuScale Plant Comparison with Other Facilities,” provides the major NuScale Power Plant design features and nominal parameters; the associated DCA 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. DCA Part 2, Tier 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

DCA Part 2, Tier 2, Section 1.10, “Nuclear Power Plants to be Operated on Multi-Unit Sites,” NuScale provides COL Item 1.10-1, directing a COL applicant that references the NuScale Power Plant design certification to evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk-significant SSCs of existing operating unit(s) and newly constructed operating unit(s) at the collocated site in accordance with 10 CFR 52.79(a)(31). NuScale stated that COL Item 1.10-1 is not applicable for construction activities (buildout of the facility) at an individual NuScale Power Plant with operating NPMs.

1.6 Identification of Agents and Contractors

1.6.1 General and Financial Information

DCA Part 1, “General and Financial Information,” states that the DCA is submitted by NuScale Power, LLC. All references herein to "NuScale Power" or "NuScale," as an entity, refer to NuScale Power, LLC. NuScale Power, LLC's headquarters are located at 6650 SW Redwood Lane, Suite 210, Portland, OR 97224. NuScale Power, LLC is engaged principally in the business of developing and marketing a nuclear power plant design based on the NuScale SMR.

DCA Part 1 further states that the material in the DCA is based upon work supported by the U.S. Department of Energy (DOE) under Award Number DE-NE0000633. The NuScale DCA was prepared as an account of work sponsored by an agency of the U.S. Government. The staff understands and believes that the DOE award to NuScale was based, in part, on DOE’s assessment of NuScale’s technical competency. In addition, the staff had extensive technical interactions with NuScale in 2015 and 2016 during the application preparation and readiness assessment activities. During this time window, the staff also reviewed and accepted a number of topical reports submitted by NuScale that have subsequently been referenced in the DCA.

Many of these reports have already been approved by the staff. The staff also conducted a number of pre-application audits at the NuScale facility in Corvallis, OR, and other facilities that support NuScale design development. During these technical interactions, the staff gained understanding of and confidence in NuScale's technical competence as a potential DC applicant. Further, the acceptance of the DCA in 2017 documents that the NRC staff performed an acceptance review of the NuScale DCA in accordance with 10 CFR 2.815, "Docketing and acceptance review," 10 CFR 52.46, "Contents of applications; general information," 10 CFR 52.47, "Contents of applications; technical information," and Office Instruction NRO-REG-100, "Acceptance Review Process for Early Site Permit, Design Certification, and Combined License Applications," dated December 18, 2014. The NRC staff concluded that the DCA is sufficiently complete and technically adequate to allow the staff to conduct its detailed technical review within a predictable timeframe. Accordingly, based on the above discussion, the staff finds that NuScale and its supporting companies are technically qualified to support the design certification represented in the DCA in accordance with 10 CFR 52.47(a)(7).

1.6.2 Principal Consultants and Other Participants

DCA Part 2 Tier 2, Section 1.4 "Identification of Agents and Contractors," states that NuScale has the overall design responsibility for the NuScale certified design. Fluor Corporation (Fluor) provided the balance of plant design described in the DCA.

DCA Part 2 Tier 2, Table 1.8-2, lists one COL item:

> COL Item 1.4-1: A COL applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.

The staff finds that COL Item 1.4-1 is reasonable because it supports the COL applicant's compliance with 10 CFR 52.79(a)(26), which requires a COL applicant to provide, "the applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications for operation."

1.7 Requirements for Additional Technical Information

DCA Part 2, Tier 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 DCA Part 2, Tier 2, Section 1.5, was developed to provide data to support the final safety analyses.

1.7.1 NuScale Testing Programs

The tests under the following testing programs focus on design features of the NPM for which applicable data or operational experience did not previously exist. DCA Part 2, Tier 2, Section 1.5.1.1, "Critical Heat Flux Testing—Preliminary Fuel Design," and Section 1.5.1.2, "Critical Heat Flux Testing—NuFuel HTP2™ Fuel Design," summarize tests specific to the NuScale fuel design (refer to Section 3.1.3.2, "Equivalent Grid Spacers," of the NRC staff's safety evaluation report on Topical Report TR-0116-21012, "NuScale Power Critical Heat Flux Correlations," for discussion of this test); DCA Part 2, Tier 2, Section 1.5.1.3, "Steam Generator Thermal-Hydraulic Performance Testing—Electrically Heated Facility," and Section 1.5.1.4, "Steam Generator Thermal-Hydraulic Performance Testing—Fluid-Heated Facility," summarize tests specific to the SG (refer to the NRC staff's FSER Section 5.4.2, "Steam Generator Program," for discussion of these tests); DCA Part 2, Tier 2, Section 1.5.1.5, "NuScale Integral

System Test Program," summarizes tests involving integrated system phenomena (refer to the NRC staff's FSER Section 6.2.1.1, "Containment Structure," and Section 15.6.5.2, "Long-Term Cooling after a Loss-of-Coolant Accident," Subsection 15.6.5.2.4.1, "Evaluation Model," for discussion of this test). DCA Part 2, Tier 2, Section 1.5.1.6, "Control Rod Drive Mechanism Proof Test," and Section 1.5.1.7, "Control Rod Assembly Drop and Control Rod Drive Shaft Alignment Test," summarize tests specific to the CRAs (refer to the NRC staff's FSER Section 3.9.4.4.2, "Descriptive Information," for discussion of these tests). The following NRC staff inspection reports document the results of the staff's inspection of some of the aforementioned tests: "IR 99901418-13-201 and Notice of Violation, March 4-8, 2013, NuScale Power LLC" (ADAMS Accession No. ML13098A338); "IR 99901437-13-201, on 12/9-13/2013, NuScale Power, LLC, Siet S.p.A" (ADAMS Accession No. ML14023A613); and "Nuclear Regulatory Commission Inspection Of NuScale Power LLC Inspection Report No. 99901351/2015-201 And Notice Of Violation" (ADAMS Accession No. ML15268A186).

1.7.1.1 Critical Heat Flux Testing—Preliminary Fuel Design

DCA Part 2, Tier 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 DCA Part 2, Tier 2, Section 1.5.1.2).

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 and provided data that were used to develop NuScale's NSP2 CHF correlation in support of the NuScale SMR technology. NuScale further stated that the NRC inspected this test program in accordance with Inspection Procedure (IP) 35017, "Quality Assurance Implementation Inspection," dated July 29, 2008 (ADAMS Accession No. ML081410388); IP 35034, "Design Certification Testing Inspection," dated January 27, 2010 (ADAMS Accession No. ML082140148); and IP 36100, "Inspection of 10 CFR Part 21 and Programs for Reporting Defects and Noncompliance," dated May 16, 2019 (ADAMS Accession No. ML19087A149).

1.7.1.2 Critical Heat Flux Testing—NuFuel HTP2™ Fuel Design

DCA Part 2, Tier 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 DCA Part 2, Tier 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

DCA Part 2, Tier 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

DCA Part 2, Tier 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

DCA Part 2, Tier 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-1) 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 and PIM.

The NIST-1 is a scaled representation of the NPM reactor, containment, and reactor pool systems. It is constructed of stainless steel and has a maximum operating pressure of 11.4 MPa (1,650 psia) and temperature of 332 degrees Celsius (630 degrees Fahrenheit or 605 degrees Kelvin).

NuScale stated that the following tests have been completed at the NIST-1, 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
- flow-stability tests used to validate PIM for reactor stability analyses
- non-LOCA (AOO) tests used to validate NRELAP5 for non-LOCA analyses
- long-term cooling tests used to validate NRELAP5 for long-term cooling analyses

Data obtained from the NIST-1 tests identified above have been used to validate the NRELAP5 and PIM codes for LOCA and containment, non-LOCA, flow stability, and long-term cooling applications.

1.7.1.6 Control Rod Drive Mechanism Proof Test

As stated by NuScale, the CRDM for the NPM contains features that are uncommon in conventional CRDMs: a remote disconnect mechanism and a long control rod drive shaft. A proof-of-concept testing program was conducted to demonstrate the validity of these new designs with regard to performance, reliability, and repeatability of each system. DCA Part 2, Tier 2, Section 1.5.1.7, describes additional testing to determine misalignment limits.

NuScale concluded that the tests demonstrated hardware performance, which has been extended to aid in the design of drive rod position detection circuitry. Information gained from this testing has been used as a development tool to improve the design and does not create a design basis for the final CRDM.

1.7.1.7 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:

- CRDM nozzles in the reactor vessel head
- integrated steam plenum
- five 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 AREVA 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.

1.7.1.8 NuScale Test and Inspection Plans

DCA Part 2, Tier 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

DCA Part 2, Tier 2, Table 1.6-1 and Table 1.6-2, "NuScale Referenced Technical Reports," incorporate by reference topical reports and technical reports, respectively, as part of the NuScale Power Plant DCA. Tables 1.8-1 and 1.8-2 below list the publicly available versions of these reports.

Table 1.8-1 NuScale Referenced Topical Reports

	Topical Report Number (ADAMS Accession No.)	Topical Report Title	Submittal Date	DCA Part 2, Tier 2, Chapter
1	TR-1010-859-NP-A, Revision 5 (ML20176A494)	NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant	June 2020	17
2	TR-0515-13952-NP-A, Revision 0 (ML16284A016)	Risk Significance Determination	October 2016	17, 19
3	TR-0815-16497-NP-A, Revision 1 (ML18054B607)	Safety Classification of Passive Nuclear Power Plant Electrical Systems	February 2018	8, 15
4	TR-1015-18653-NP-A, Revision 2 (ML17256A892)	Design of the Highly Integrated Protection System Platform Topical Report	September 2017	7, 15

	Topical Report Number (ADAMS Accession No.)	Topical Report Title	Submittal Date	DCA Part 2, Tier 2, Chapter
5	TR-0915-17565-NP-A, Revision 4 (ML20057G132)	Accident Source Term Methodology	February 2020	15
6	TR-0116-20825-NP-A, Revision 1 (ML18040B306)	Applicability of AREVA Fuel Methodology for the NuScale Design	February 2018	4
7	TR-0616-48793-NP-A, Revision 1 (ML18348B036)	Nuclear Analysis Codes and Methods Qualification	December 2018	4
8	TR-0516-49417-NP-A, Revision 1 (ML20078Q094)	Evaluation Methodology for Stability Analysis of the NuScale Power Module	March 2020;	4
9	TR-0516-49422-NP-A, Revision 2 (ML20189A644)	Loss-of-Coolant Accident Evaluation Model	July 2020	15
10	TR-0915-17564-NP-A, Revision 2 (ML19067A256)	Subchannel Analysis Methodology	March 2019	4
11	TR-0516-49416-NP-A, Revision 3 (ML20191A281)	Non-Loss-of-Coolant Accident Analysis Methodology	July 2020	15
12	TR-0116-21012-NP-A, Revision 1 (ML18360A632)	NuScale Power Critical Heat Flux Correlations	December 2018	4
13	TR-0716-50350-NP-A, Revision 1 (ML20168B203)	Rod Ejection Accident Methodology	June 2020	15
14	TR-0716-50351-NP-A, Revision 1 (ML20122A248)	NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces	May 2020	4

Table 1.8-2 NuScale Referenced Technical Reports

	Technical Report Number	Technical Report Title	Submittal Date	DCA Part 2, Tier 2, Section
1	TR-0116-20781-NP, Revision 1 (ML19183A485)	Fluence Calculation Methodology and Results	July 2019	4.3, 5.3
2	TR-0316-22048-NP, Revision 3 (ML20141M764)	Nuclear Steam Supply System Advanced Sensor Technical Report	May 2020	7.1, 7.2
3	TR-0516-49084-NP, Revision 3 (ML20141L808)	Containment Response Analysis Methodology Technical Report	May 2020	6.2
4	TR-0616-49121-NP, Revision 3 (ML20141M114)	NuScale Instrument Setpoint Methodology Technical Report	May 2020	7.0, 7.2
5	TR-0716-50424-NP, Revision 1 (ML19091A232)	Combustible Gas Control	March 2019	6.2
6	TR-0716-50439-NP, Revision 2 (ML19212A776)	NuScale Comprehensive Vibration Assessment Program Analysis Technical Report	July 2019	3.9, 14.2
7	TR-0816-49833-NP, Revision 1 (ML18310A154)	Fuel Storage Rack Analysis	November 2018	9.1
8	TR-0816-50796-NP, Revision 1 (ML19165A294)	Loss of Large Areas Due to Explosions and Fires Assessment	June 2019	20.2
9	TR-0816-50797 (NuScale Nonproprietary), Revision 3 (ML19302H598)	Mitigation Strategies for Loss of All AC Power Event	October 2019	20.1
10	TR-0816-51127-NP, Revision 3 (ML19353A719)	NuFuel-HTP2™ Fuel and Control Rod Assembly Designs	December 2019	4.2
11	TR-0818-61384-NP, Revision 2 (ML19212A682)	Pipe Rupture Hazards Analysis	July 2019	3.6
12	TR-0916-51299-NP, Revision 3 (ML20141L816)	Long-Term Cooling Methodology	May 2020	5.4, 6.2, 6.3, 15.0

	Technical Report Number	Technical Report Title	Submittal Date	DCA Part 2, Tier 2, Section
13	TR-0916-51502-NP, Revision 2 (ML19093B850)	NuScale Power Module Seismic Analysis	April 2019	3.7, 3.12, 3B
14	TR-0917-56119-NP, Revision 1 (ML19158A382)	CNV Ultimate Pressure Integrity	June 2019	3.8
15	TR-0918-60894-NP, Revision 1 (ML19214A248)	Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report	August 2019	3.9, 14.2
16	TR-1015-18177-NP, Revision 2 (ML18298A304)	Pressure and Temperature Limits Methodology	October 2018	5.3
17	TR-1016-51669-NP, Revision 1 (ML19211D411)	NuScale Power Module Short-Term Transient Analysis	July 2019	3.9
18	TR-1116-51962-NP, Revision 1 (ML19149A298)	NuScale Containment Leakage Integrity Assurance Technical Report	May 2019	6.2
19	TR-1116-52065-NP, Revision 1 (ML18317A364)	Effluent Release (GALE Replacement) Methodology and Results	November 2018	11.1, 11.2, 11.3
20	RP-0215-10815-NP, Revision 3 (ML19133A293)	Concept of Operations	May 2019	18.7
21	RP-0316-17614-NP, Revision 0 (ML16364A342)	Human Factors Engineering Operating Experience Review Results Summary Report	December 2016	18.2
22	RP-0316-17615-NP, Revision 0 (ML16364A342)	Human Factors Engineering Functional Requirements Analysis and Function Allocation Results Summary Report	December 2016	18.3
23	RP-0316-17616-NP, Revision 2 (ML19119A393)	Human Factors Engineering Task Analysis Results Summary Report	April 2019	18.4
24	RP-0316-17617-NP, Revision 0 (ML17004A222)	Human Factors Engineering Staffing and Qualifications Results Summary Report	December 2016	18.5
25	RP-0316-17618-NP, Revision 0 (ML17004A222)	Human Factors Engineering Treatment of Important Human Actions Results Summary Report	December 2016	18.6

	Technical Report Number	Technical Report Title	Submittal Date	DCA Part 2, Tier 2, Section
26	RP-0316-17619-NP, Revision 2 (ML19119A398)	Human Factors Engineering Human-System Interface Design Results Summary Report	April 2019	18.7
27	RP-0516-49116-NP, Revision 1 (ML16364A356)	Control Room Staffing Plan Validation Results	December 2016	18.5
28	RP-0914-8534-NP Revision 5 (ML19119A342)	Human Factors Engineering Program Management Plan	April 2019	18.1
29	RP-0914-8543-NP, Revision 5 (ML19119A372)	Human Factors Verification and Validation Implementation Plan	April 2019	18.1
30	RP-0914-8544-NP, Revision 4 (ML19331A910)	Human Factors Engineering Design Implementation Implementation Plan	November 2019	18.11
31	RP-1215-20253-NP, Revision 3 (ML16364A353)	Control Room Staffing Plan Validation Methodology	December 2016	18.5
32	ES-0304-1381-NP, Revision 4 (ML19338E948)	Human-System Interface Style Guide	December 2019	18.10
33	RP-1018-61289-NP, Revision 1 (ML19212A773)	Human Factors Engineering Verification and Validation Results Summary Report	July 2019	18.1

1.9 Drawings and Other Detailed Information

Where appropriate, DCA 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.

DCA Part 2, Tier 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 DCA Part 2 (see DCA Part 2, Tier 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 design certification to provide site-specific diagrams and legends, as applicable.

DCA Part 2, Tier 2, Table 1.7-2, lists system drawings used in DCA Part 2 (see DCA Part 2, Tier 2, Figures 1.7-3a through 1.7-3f, 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 design certification to list additional site-specific piping and instrumentation diagrams and legends as applicable.

1.10 Interfaces with Certified Designs

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

DCA Part 2, Tier 2, Table 1.8-1, “Summary of NuScale Certified Design Interfaces with Remainder of Plant,” identifies the interfaces between the NuScale certified design and the site-specific design. Table 1.10-1 below is derived from DCA Part 2 Tier 2, Table 1.8-1. There are two types of interfaces:

(1) CDI—conceptual design information provided for the noncertified portion of the plant to facilitate review of the certified design

(2) COL—site-specific design elements that are the responsibility of the COL applicant; this type of interface is identified as a COL information item

Table 1.10-1 Summary of NuScale Certified Design Interfaces with Remainder of Plant

System, Structure, or Component	Interface Type	DCA Part 2, Tier 2, Section
Turbine Generator Buildings	CDI	1.2.2
Access Building	CDI	1.2.2
Cooling towers, pump houses, and associated structures, systems, and components (e.g., cooling tower basin, circulating water pumps, cooling tower fans, chemical treatment building, etc.)	CDI	1.2.2, 10.4.5
Security Buildings	CDI	1.2.2
Central Utility Building	CDI	1.2.2
Diesel Generator Buildings	CDI	1.2.2
Offsite power transmission system, main switchyard, and transformer area	CDI	8.2
Auxiliary AC power system	CDI	8.3.1
Site cooling water system	CDI	9.2.7
Circulating water system	CDI	10.4.5

System, Structure, or Component	Interface Type	DCA Part 2, Tier 2, Section
Grounding and lightning protection system	CDI	8.3.1
Plant exhaust stack	CDI	9.4.2
Potable and sanitary water systems	COL	9.2.4
Resin tanks for the condensate polishing system	COL	10.4
Site drainage system	COL	N/A
Raw water system	COL	9.2.9
Site parameters, geographic and demographic characteristics, meteorological characteristics, nearby industrial, transportation, and military facilities, hydrologic characteristics, geology, seismology, and geotechnical characteristics, weather conditions and site topography, flooding	COL	Table 2.0-1, 2.1, 2.2, 2.3, 2.4, 2.5, 3.3, 3.4
Site-specific communications	COL	9.5.2
Turbine generators	COL	3.5-1
Operational Support Center	COL	13.3

1.11 Conformance with Regulatory Criteria

DCA Part 2, Tier 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 (TMI) requirements), operational experience (i.e., generic communications), and advanced and evolutionary light-water reactor design issues.

Conformance with Regulatory Guides

DCA Part 2, Tier 2, Table 1.9-2, "Conformance with Regulatory Guides," provides an evaluation of conformance with the NRC RGs in effect 6 months before the application docket date. This evaluation also includes an identification and description of deviations from the NRC RGs and suitable justifications for any alternative approaches proposed.

NuScale further stated that the conformance evaluation was performed on the following groups of RGs:

- Division 1, "Power Reactors"
- Division 4, "Environmental and Siting" (applies to the environmental report and should be discussed therein)
- Division 5, "Materials and Plant Protection" (applies to the security plan and should be discussed therein)
- Division 8, "Occupational Health"

1.11.1 Conformance with Standard Review Plan

DCA Part 2, Tier 2, Section 1.9.2, “Conformance with Standard Review Plan,” indicates that NuScale reviewed the SRP (NUREG-0800), including branch technical positions and guidance referenced within the SRP, and submitted a summary of this review to the NRC as NP-RT-0612-023, “Gap Analysis Summary Report,” Revision 1, in July 2014 (ADAMS Accession No. ML14212A832). The gap analysis review for applicability was directed toward the acceptance criteria of each SRP section. NuScale further stated that the review considered the relevance of subtier guidance, whether referenced in the acceptance criteria or in other portions of the SRP being reviewed. Additionally, NuScale stated that it considered conformance with the design-specific review standard that the NRC staff had developed as a supplement to the SRP for the review of the NuScale SMR design. NuScale incorporated this information into DCA Part 2, Tier 2, Table 1.9-3 “Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS).” DCA Part 2, Tier 2, Table 1.9-4, “Conformance with Interim Staff Guidance (ISG),” presents NuScale’s conformance with NRC interim staff guidance.

1.11.1.1 Generic Issues and Three Mile Island Requirements

DCA Part 2, Tier 2, Section 1.9.3, “Generic Issues,” states that in accordance with 10 CFR 52.47(a)(8), conformance is assessed against technically relevant TMI requirements identified in 10 CFR 50.34(f), except for 10 CFR 50.34(f)(1)(xii), (f)(2)(ix), and (f)(3)(v). The appropriate DCA Part 2 sections describe plant characteristics and plant programs that address relevant TMI requirements.

NuScale further stated that in accordance with 10 CFR 52.47(a)(21), proposed resolutions must be identified for any technically relevant, unresolved safety issues and medium-priority to high-priority generic safety issues identified in the version of NUREG-0933, “Resolution of Generic Safety Issues,” that is current 6 months prior to the application for design certification. Resolution and closure of generic issues is managed through the NRC Generic Issues Program. In SECY-07-0110, “Summary of Activities Related to Generic Issues Program,” dated July 6, 2007, the staff provides the most recent supplemental status report of the Generic Issues Program. As such, NuScale used Appendix B, “Applicability of NUREG-0933 Issues to Operating and Future Reactor Plants,” to NUREG-0933, Revision 21, and SECY-07-0110 to identify those generic issues applicable to the NuScale Power Plant design certification.

DCA Part 2, Tier 2, Table 1.9-5, “Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933),” identifies the applicable TMI requirements and generic issues, along with an abbreviated summary description of the NRC position for each table entry. DCA Part 2, Tier 2, Table 1.9-5, also provides a brief conformance assessment notation, including annotation of any exceptions, and a reference to the DCA Part 2 section or sections addressing the issue; those NUREG-0933 generic issues determined to be nonapplicable were eliminated from consideration in DCA Part 2, Tier 2, Table 1.9-5, based on the following:

- <u>Resolved</u>—The issue has been completely resolved and removed from the latest Generic Issues Program list of active and regulatory office implementation generic issues.

- <u>Boiling-water reactor, ice condenser containment, or other</u>—The issue applies to another nuclear power plant design concept or to the design of a nuclear facility other than a nuclear power plant.

1.11.1.2 Operational Experience (Generic Communications)

DCA Part 2, Tier 2, Section 1.9.4, “Operational Experience (Generic Communications),” states that in accordance with 10 CFR 52.47(a)(22) requirements, applicants for design certification of new plant designs describe how the design process has incorporated operational experience. NuScale stated that operational experience insights are incorporated into applicable SRP sections as they are updated. NuScale further stated that operational experience from NRC bulletins and generic letters not incorporated into the most recent applicable SRP 6 months before the application docket date is incorporated into the design unless stated otherwise. NuScale also stated that the design is an evolution of nuclear power plant designs that have operated in the United States, as addressed by 10 CFR 52.41(b)(1); therefore, the appropriate DCA Part 2 sections address NRC guidance for technically relevant operational experience issues.

DCA Part 2, Tier 2, Table 1.9-6, “Evaluation of Operating Experience (Generic Letters and Bulletins),” provides the conformance assessment relative to operational experience.

1.11.1.3 10 CFR Part 21 Notification of Failure To Comply or Existence of a Defect and Its Evaluation

DCA Part 2, Tier 2, Section 1.9.4, states that the applicant reviewed notifications under 10 CFR Part 21, “Reporting of Defects and Noncompliance,” for impact on the NuScale design as part of the supplier evaluation process. NuScale’s QA supplier evaluation program includes a review of 10 CFR Part 21 notifications for every nuclear safety-related supplier before use as an approved supplier for safety related items or services. NuScale also evaluates any 10 CFR Part 21 notifications as part of the monitoring of supplier performance by periodic annual review. There have been no 10 CFR Part 21 notifications impacting nuclear safety-related work performed by NuScale-approved safety-related suppliers for the development of the NuScale design. Therefore, all applicable 10 CFR Part 21 notification requirements have been met.

1.11.1.4 Advanced and Evolutionary Light-Water Reactor Design Issues (Commission Documents)

DCA Part 2, Tier 2, Section 1.9.5, “Advanced and Evolutionary Light-Water Reactor Design Issues,” states that guidance in SRP Chapter 1.0, “Introduction and Interfaces,” recommends that this section address the applicable licensing and policy issues the NRC developed and documented in SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 2, 1993, as supplemented by the associated staff requirements memorandum (SRM) dated July 21, 1993. DCA Part 2, Tier 2, Table 1.9-7, “Conformance with Advanced and Evolutionary Light Water Reactor Design Issues (SECYs and Associated SRMs),” lists applicable design issues identified in SECYs and their associated SRM. The table provides a conformance assessment notation, including annotation of any exceptions, for each issue. DCA Part 2, Tier 2, Table 1.9-7, also provides a cross reference from the SECY issues to the DCA Part 2 sections that address them. DCA Part 2, Tier 2, Table 1.9-8, “Conformance with SECY-93-087, ‘Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs,’” provides a separate assessment of SECY-93-087 line items pertaining to advanced light-water reactor designs.

1.11.1.5 Mitigation of Beyond-Design-Basis Events

Although recent regulations in 10 CFR 50.155, governing mitigation of beyond-design-basis events, do not apply to applicants for design certification, NuScale is voluntarily seeking the

NRC's approval of its proposal to use installed design features for mitigation of beyond-design-basis external events. These features are discussed in SECY-19-0066, "Staff Review of NuScale Power's Mitigation Strategy for Beyond- Design-Basis External Events," dated June 26, 2019 (ADAMS Accession No. ML19148A443). The following portions of the DCA discuss NuScale's approach to these issues:

- DCA Part 2, Tier 2, Section 20.1, "Mitigating Strategies for Beyond-Design-Basis External Events," discusses the mitigating strategies that address an extended loss of AC power and loss of normal access to the UHS resulting from a beyond-design-basis external event.
- DCA Part 2, Tier 2, Section 20.2, "Loss of Large Areas of the Plant due to Explosions and Fires," addresses the loss of large areas of the plant from fire or explosion.
- DCA Part 2, Tier 2, Section 20.3, "Integration with Emergency Procedures," contains COL Item 20.3-1, which directs a COL applicant referencing the NuScale Power Plant design certification to address procedure integration.

1.12 <u>Summary of Open Items</u>

As a result of the staff's review of the NuScale DCA, 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 DCA, 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 <u>Summary of Confirmatory Items</u>

The NRC staff's review of the NuScale DCA 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 DCA. All confirmatory items have been closed for the DCA review.

1.14 <u>Index of Exemptions</u>

In accordance with 10 CFR 52.48, "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 DCA. During this review, the staff recognized 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 17 exemption requests, which are provided in DCA 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, except for the exemption from 10 CFR Part 50, Appendix A, GDC 19, "Control Room," which affects a large number of FSER sections. The staff's evaluation of that exemption is documented in this chapter following Table 1.14-1.

Table 1.14-1: NuScale Design Certification Exemptions

FSER Section	Exemption
5.4.5 6.2	10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” and 10 CFR 50.34(f), “Additional TMI-related requirements.” NuScale requests an exemption from the requirements contained in 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) specifying high point vents for the RCS and RPV head.
6.2.5	10 CFR 50.44, “Combustible gas control for nuclear power reactors.” NuScale requests an exemption from the combustible gas control requirements of 10 CFR 50.44(c)(2).
7.1 7.2 15.8 19.1.9 19.2.2	10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.” NuScale requests an exemption from the portion of 10 CFR 50.62(c)(1) requiring equipment diverse and independent from the reactor trip system (RTS) to automatically initiate a turbine trip under conditions indicative of an ATWS. The portion of 10 CFR 50.62(c)(1) requiring diverse AFWS initiation is not applicable to the NuScale design, and therefore not within the scope of this exemption request.
3.1 3.2 7.1.2.2 8.2 8.3.1 8.3.2 8.4 15 19.1 19.2	10 CFR Part 50, Appendix A, GDC 17, “Electric power systems,” GDC 18, “Inspection and testing of electric power systems,” GDC 34, “Residual heat removal,” GDC 35, “Emergency core cooling,” GDC 38, “Containment heat removal,” GDC 41, “Containment atmosphere cleanup,” and GDC 44, “Cooling water.” NuScale requests an exemption from GDC 17, pertaining to electric power systems, and GDC 18, pertaining to inspection and testing of those electric power systems. In addition, NuScale requests an exemption from the related provisions of GDC 34, 35, 38, 41, and 44, which describe capabilities for specific systems with respect to electric power. For each of these GDC, exemption is sought from the phrase "for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available)." Exemption from this provision of GDC 34, 35, 38, 41, and 44 is consistent with the proposed exemptions from GDC 17 and GDC 18.
9.3.4 6.3	10 CFR Part 50, Appendix A, GDC 33, “Reactor Coolant Makeup.”. NuScale requests an exemption from the requirements of GDC 33, which requires a system to provide reactor coolant makeup for protection against small breaks in the RCPB.
13.1 18	10 CFR 50.54(m) (control room staffing). NuScale requests that minimum licensed operator staffing requirements specific to the NuScale Power Plant design be adopted as requirements applicable to licensees referencing the NuScale Power Plant design certification, in lieu of the requirements stated in 10 CFR 50.54(m).

FSER Section	Exemption
6.2.6	10 CFR Part 50, Appendix A, GDC 52, "Capability for Containment Leakage Rate Testing." NuScale requests an exemption from 10 CFR 50 Appendix A, GDC 52, capability for containment leakage rate testing at design pressure. 10 CFR 50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," specifies Type A testing directly related to GDC 52. While Appendix J is not applicable to a design certification applicant, NuScale requests that, with approval of the GDC 52 exemption, the NuScale Power Plant design certification rule include exemption from the requirements of 10 CFR 50 Appendix J Type A tests for plants referencing the NuScale design.
6.2.1 6.2.2 6.3	10 CFR Part 50, Appendix A, GDC 40, "Testing of containment heat removal system.". NuScale requests an exemption from GDC 40, periodic pressure and functional testing of the containment heat removal system.
5.4.4 6.2.4	10 CFR Part 50, Appendix A, GDC 55, "Reactor coolant pressure boundary penetrating containment," GDC 56, "Primary containment isolation," and GDC 57, "Closed system isolation valves.". NuScale requests an exemption from 10 CFR Part 50, Appendix A, • GDC 55 for the lines with penetrations CNV6, CNV7, CNV13, and CNV14 to allow the placement of two CIVs outside containment rather than locating one of the CIVs inside containment as specified in GDC 55; • GDC 56 for the lines with penetrations CNV5, CNV10, CNV11, and CNV12 to allow the placement of two CIVs outside containment rather than locating one of the CIVs inside containment as specified in GDC 56; and • GDC 57 for the lines with penetrations CNV3, CNV4, CNV22, and CNV23 to allow the use of a closed system outside containment rather than providing a CIV as specified in GDC 57.
15	10 CFR Part 50, Appendix K, "ECCS Evaluation Models." NuScale requests an exemption from the requirements of 10 CFR 50 Appendix K identified in DCA Part 7, "Exemptions," Section 10.1.2, "Regulatory Requirements."
5.4.6	10 CFR 50.34(f), "Additional TMI-related requirements." NuScale requests an exemption from the portions of 10 CFR 50.34(f)(2)(xx) requiring power from vital buses and emergency power sources for pressurizer level indication.
5.4.6	10 CFR 50.34(f), "Additional TMI-related requirements." NuScale requests an exemption from the requirements of 10 CFR 50.34(f)(2)(xiii).
5.2.5 6.2.4 9.3.6	10 CFR 50.34(f), "Additional TMI-related requirements." NuScale requests an exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the containment evacuation system (CES).

FSER Section	Exemption
4.2.4.6	10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.” NuScale requests an exemption from the requirement of 10 CFR 50.46 regarding the use of zircaloy or ZIRLO as a fuel rod cladding material. An exemption is required because 10 CFR 50.46 does not anticipate the use of fuel rods with cladding materials other than zircaloy or ZIRLO.
3.1 6.3 15 15.6.5 15.6.6 19	10 CFR Part 50, Appendix A, GDC 27, “Combined reactivity control systems capability.” NuScale requests an exemption from GDC 27 to the extent it has been implemented to require demonstration of long-term shutdown under post-accident conditions with an assumed worst rod stuck out.
12.03	10 CFR 50.34(f), “Additional TMI-related requirements.” NuScale requests an exemption from the post-accident sampling requirements of 10 CFR 50.34(f)(2)(viii).
DCA Part 2, Tier 2, Sections 1.2, 1.9, 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 DCA Part 4, Technical Specifications B3.3, B3.4	10 CFR Part 50, Appendix A, GDC 19, “Control room.” NuScale requests an exemption from GDC 19, which requires equipment outside the control room providing a potential capability for cold shutdown of the reactor through the use of suitable procedures.

Exemption from 10 CFR Part 50, Appendix A, General Design Criterion 19

NuScale requested an exemption from General Design Criterion (GDC) 19, “Control room,” and proposed to implement a design-specific Principal Design Criterion (PDC) 19 that meets the underlying purpose of the GDC 19 requirement for means to maintain the reactor in a safe condition in the event of a control room evacuation (ADAMS Accession No. ML19073A331). NuScale states that, the NuScale Power Plant design, as reflected in the Final Safety Analysis Report (FSAR) (DCA Part 2, Tier 2), conforms to proposed PDC 19, assuring the design capability for safe shutdown from equipment outside the control room, in lieu of the requirements for "design capability for prompt hot shutdown" and "potential capability for subsequent cold shutdown” as specified in GDC 19.

GDC 19 states the following:

> *Criterion 19 - Control room*. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

NuScale's proposed PDC 19 is stated in DCA Part 2, Tier 2, Section 3.1, "Conformance with U.S. Nuclear Regulatory Commission General Design Criteria," and summarized below:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.
>
> Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) as defined in 10 CFR 50.2 for the duration of the accident.
>
> Equipment at appropriate locations outside the control room shall be provided with a design capability for safe shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe shutdown condition.

Requirements associated with the review of this exemption request not specified above include the following:

- 10 CFR 52.47(a), which requires, in part, the following:

  - The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:

  (3) The design of the facility including:

    (i) The principal design criteria for the facility. Appendix A to 10 CFR Part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for watercooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units;

    (ii) The design bases and the relation of the design bases to the principal design criteria.

- 10 CFR 52.7, “Specific Exemptions,” which states the following:

  The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part. The Commission’s consideration will be governed by § 50.12 of this chapter, unless other criteria are provided for in this part, in which case the Commission’s consideration will be governed by the criteria in this part. Only if those criteria are not met will the Commission’s consideration be governed by § 50.12 of this chapter. The Commission’s consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are applicable by virtue of this part, shall be governed by the exemption requirements of those parts.

- 10 CFR 50.12(a), which states, in part, that the following two conditions that must be met for granting an exemption:

  1) Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.

  2) The Commission will not consider granting an exemption unless special circumstances are present. (Circumstances are enumerated in 10 CFR 50.12(a)(2)).

Evaluation for Meeting the Exemption Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, the Commission may, upon application by any interested person or upon its own initiative, consider exemptions from the requirements of 10 CFR Part 52. As 10 CFR 52.7 further states, the Commission’s consideration will be governed by 10 CFR 50.12, which states that an exemption may be granted when (1) the exemptions are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security, and (2) special circumstances are present. Specifically, 10 CFR 50.12(a)(2) lists six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request.

Authorized by Law

This exemption is not inconsistent with the Atomic Energy Act of 1954, as amended or the Commission’s regulations because, as stated above, 10 CFR Part 52 allows the NRC to grant exemptions. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is authorized by law.

No Undue Risk to Public Health and Safety

This exemption does not affect the performance or reliability of power operations, does not impact the consequences of any design basis event, and does not create new accident precursors. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption poses no undue risk to public health and safety.

Consistent with Common Defense and Security

The proposed exemption will not alter the design, function, or operation of any structures or plant equipment necessary to maintain a safe and secure plant status. In addition, the changes

have no impact on plant security or safeguards. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the common defense and security is not impacted by this exemption.

Special Circumstances

Underlying Purpose of the Rule

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. The underlying intent of the remote shutdown portion of GDC 19 is to provide means for operators to place and maintain the reactor in a safe condition in the event of a control room evacuation and the requirement of “cold shutdown” in GDC 19 is not necessary to achieve this purpose. For NuScale's passive advanced light water reactor design, the establishment of PDC 19 to require remote "safe shutdown" capability instead of "cold shutdown" specifically, is supported and consistent with NRC guidance, such as SECY-94-084, “Policy and Technical issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” dated March 28, 1994, which applies to passive residual heat removal systems and RG 1.189, “Fire Protection for Nuclear Power Plants,” regarding fire in the main control room. In the event of an MCR evacuation for the NuScale design, all reactors are tripped, and decay heat removal and containment isolation are initiated prior to operators evacuating the MCR. These actions result in passive cooling that achieves and maintains safe shutdown. Operators can also place the reactors in safe shutdown from outside the MCR in the module protection system equipment rooms within the reactor building. Accordingly, the NRC staff determined that the applicant has met the underlying purpose of the remote shutdown portion of GDC 19 by providing means for operators to maintain the reactor in a safe condition in the event of a control room evacuation.

In DCA Part 7, Section 17, “10 CFR 50, Appendix A, Criterion 19, Control Room,” the applicant stated that special circumstances described in 10 CFR 50.12(a)(2)(iv) associated with a benefit to public health and safety are present. However, as described in 10 CFR 50.12(a)(2), where the staff finds that special circumstances are present in accordance with 10 CFR 50.12(a)(2)(ii), a staff finding on whether special circumstances exist in accordance with 10 CFR 50.12(a)(2)(iv) is not necessary for the exemption to be granted. Because the staff finds that special circumstances are present in accordance with 10 CFR 50.12(a)(2)(ii), the staff makes no finding regarding the presence of special circumstances described in 10 CFR 50.12(a)(2)(iv).

Conclusion

The staff concludes that PDC 19 maintains the required control room and remote shutdown capabilities, but clarifies that safe shutdown is the necessary reactor condition to achieve and maintain from outside the control room. In accordance with 10 CFR 50.12(a)(1), the staff finds that the requested exemption to GDC 19 is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. The NRC has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present because application of the regulation is not necessary to achieve the underlying purpose of the rule. The staff approves granting NuScale’s proposed exemption from the requirements of GDC 19.

1.15 Index of Tier 2* Information

Tier 2* information is information that requires NRC approval before a departure is taken from the certified design, in accordance with the applicable design certification rule. NuScale has not identified any Tier 2* information pertaining to the NuScale SMR design.

1.16 Combined License Information Items

COL applicants and licensees referencing the certified NuScale Power Plant standard design must satisfy the requirements and commitments identified in DCA Part 2, which is the controlling document used in the certification of the NuScale SMR design. In addition, DCA Part 2 identifies certain general commitments as COL information items, which relate to programs, procedures, and issues that are outside the scope of the certified design review. These COL information items do not establish requirements; rather, they identify an acceptable set of information to be included in a plant-specific safety report. An applicant for a COL must address each of these items in its application. It may deviate from or omit these items, provided that the deviation or omission is identified and justified in the plant-specific FSAR.

DCA Part 2, Tier 2, Section 1.8.1, “Combined License Information Items,” identifies the COL information items developed as part of the design certification review and referenced throughout the Final Safety Analysis Report. DCA Part 2, Tier 2, Table 1.8 2, lists these COL information items and description, as well as the DCA sections where these are located. The COL applicant addresses each COL information item in the section where it is located.

The following table summarizes the COL items identified in DCA Part 2 Tier 2, Chapter 1.

Table 1.16-1 Combined License Information Items Identified in the DCA

Item No.	Description	DCA Part 2 Tier 2, Section
COL Item 1.1-1	A COL applicant that references the NuScale Power Plant design certification will identify the site-specific plant location.	1.1
COL Item 1.1-2	A COL applicant that references the NuScale Power Plant design certification will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1	A COL applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1	A COL applicant that references the NuScale Power Plant design certification will provide site-specific diagrams and legends, as applicable.	1.7
COL Item 1.7-2	A COL applicant that references the NuScale Power Plant design certification will list additional site-specific piping and instrumentation diagrams and legends as applicable.	1.7
COL Item 1.8-1	A COL applicant that references the NuScale Power Plant design certification will provide a list of departures from the certified design.	1.8
COL Item1.9-1	A COL applicant that references the NuScale Power Plant design certification will review and address the conformance with regulatory criteria in effect six months before the docket date of the COL application for the site-specific portions and operational aspects of the facility design.	1.9

Item No.	Description	DCA Part 2 Tier 2, Section
COL Item1.10-1	A COL applicant that references the NuScale Power Plant design certification will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk significant structures, systems, and components of existing operating unit(s) and newly constructed operating unit(s) at the co-located site per 10 CFR 52.79(a)(31). The evaluation will include identification of management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the limiting conditions for operation of an operating unit would not be exceeded. This COL item is not applicable for construction activities (build-out of the facility) at an individual NuScale Power Plant with operating NuScale Power Modules.	1.10

1.17 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 DCA 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 DCA 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.

1.18 Conclusion

As described above, the applicant supplemented the information in the initial DCA submission by providing revisions to the document. The staff has completed its review of the DCA, as documented throughout the FSER and, for the reasons given here, finds it to be acceptable, recognizing the three issues as not resolved within the meaning of 10 CFR 52.63(a)(5) due to the absence in the design of sufficient information regarding (1) the shielding wall design in certain areas of the plant; (2) the potential for containment leakage from the combustible gas monitoring system; and (3) the ability of the steam generator tubes to maintain structural and leakage integrity during density wave oscillations in the secondary fluid system, including the method of analysis to predict the thermal-hydraulic conditions of the steam generator secondary fluid system and resulting loads, stresses, and deformations from density wave oscillations reverse flow. Additionally, the staff has confirmed that the DCA 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.