9 AUXILIARY SYSTEMS

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (hereinafter referred to as the staff) review of Chapter 9, "Auxiliary Systems," of the NuScale Power, LLC (hereinafter referred to as the applicant), Standard Design Approval Application (SDAA), Part 2, Chapter 9 "Final Safety Analysis Report (FSAR)." The staff's regulatory findings documented in this report are based on Revision 2 of the SDAA, dated April 9, 2025 (Agencywide Documents Access and Management System Accession No. ML25099A236). The precise parameter values, as reviewed by the staff in this safety evaluation, are provided by the applicant in the SDAA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this safety evaluation to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant-provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the SDAA and not converted.

In this chapter, 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 Title 10 of the Code of Federal Regulations 10 CFR 50.2, "Definitions," and 10 CFR 52.1, "Definitions." However, among the non-safety-related SSCs 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."

9.1 Fuel Storage and Handling

9.1.1 Criticality Safety of Fresh and Spent Fuel Storage Handling

9.1.1.1 Introduction

Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling" of the FSAR discusses criticality safety of the new and spent fuel pool when storing and handling new and spent fuel in the onsite fuel storage and handling facility, which is located in the reactor building (RXB).

The staff reviewed the information in the application. The following sections of this report document the staff's evaluation of the new and spent fuel pool criticality safety design with respect to the combined license (COL) items.

9.1.1.2 Summary of Application

NuScale provided a general description of the design basis of the fresh and spent fuel pool for the US460 design in Section 9.1.1.1 of the FSAR. The FSAR states that the structures that form the fuel storage facility consist of the spent fuel pool (SFP), the stainless-steel liner in the SFP, and the Reactor Building. General Design Criterion (GDC) 62, "Prevention of criticality in fuel storage and handling"; American National Standards Institute/American Nuclear Society (ANSI/ANS) 57.1, "Design Requirements for Light Water Reactor Fuel Handling Systems"; and ANSI/ANS 57.2, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants" are considered in the design of the new and spent fuel storage facility and handling equipment. The FSAR provides no details of the fresh and spent fuel rack designs.

The application did not include specific SFP rack design information and corresponding criticality safety analyses. Instead, through the noted COL information items, NuScale deferred the SFP rack design to the COL stage. NuScale included COL Item 9.1-1 to require the COL applicant to develop plant programs for safe new and spent fuel assemblies, including criticality control, and COL Item 9.1-2, to require the COL applicant to provide the design of the spent fuel storage racks, including criticality safety analysis. Table 9.1.1-1 of this SER documents these COL information items.

NuScale stated that it considered the following requirements and guidance in the design of the fuel storage and handling facility:

- GDC 62, "Prevention of Criticality in Fuel Storage and Handling," in Appendix A to 10 CFR Part 50
- 10 CFR 50.68, "Criticality Accident Requirements"
- ANSI/ANS 57.1-1992, "Design Requirements for Light Water Reactor Fuel Handling Systems"
- ANSI/ANS 57.2-1983, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants"

The NRC staff discusses these requirements and guidance in Section 9.1.1.3 of this report.

NuScale also stated that geometrically safe configurations and plant programs and procedures prevent inadvertent criticality in the fuel storage racks and during fuel handling which will be designed and confirmed by the COL applicant per COL Item 9.1-1 and COL Item 9.1-2 as discussed above.

9.1.1.3 Regulatory Basis

The relevant requirements of the Commission regulations for criticality safety of fresh and spent fuel storage and handling are as follows:

- GDC 62, as it relates to the prevention of criticality by physical systems or processes, preferably by using geometrically safe configurations.
- 10 CFR 50.68, as it relates to preventing a criticality accident and to mitigating the radiological consequences of a criticality accident.
- 10 CFR 52.137(a)(17), which requires the applicant to provide information demonstrating how it will comply with the requirements for criticality safety as prescribed in 10 CFR 50.68(b)(2)–(b)(4).
- 10 CFR 52.137(a), which requires that the application must contain a final safety analysis report 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, or major portion thereof.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," identifies criteria that are acceptable to the staff, as summarized below. NUREG-0800, Section 9.1.1 also outlines review interfaces with other SRP sections.

The related acceptance criteria are as follows:

- The criteria for GDC 62 are specified in ANSI/ANS 57.1 and 57.2 as they relate to the prevention of criticality accidents in fuel storage and handling.
- Compliance with 10 CFR 50.68 requires that the licensee either maintain monitoring systems capable of detecting a criticality accident as described in 10 CFR 70.24, “Criticality Accident Requirements,” thereby reducing the consequences of a criticality accident, or comply with the requirements specified in 10 CFR 50.68(b), thereby reducing the likelihood of a criticality accident.

The following documents give additional criteria or guidance in support of the SRP acceptance criteria to meet the above requirements:

- Regulatory Guide (RG) 1.240, “Fresh and Spent Fuel Pool Criticality Safety Analyses,” issued March 2021
- NUREG/CR-6361, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages,” issued March 1997
- NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology,” issued January 2001

9.1.1.4 Technical Evaluation

In Section 9.1.1 and 9.1.2 of the FSAR, the applicant included two COL information items that require applicants that reference the NuScale Power Plant US460 standard design to perform criticality safety analyses to demonstrate that the fresh and spent fuel pool meets the regulatory requirements of 10 CFR 50.68(b). The staff reviewed the COL information items and finds that this approach supports the regulatory requirements of 10 CFR 52.137(a). The explicit statements in COL Items 9.1-1 and 9.1-2 that the fresh and spent fuel pool must meet the regulatory requirements of 10 CFR 50.68(b), give sufficient information for applicants who reference the US460 design to perform spent fuel and fresh fuel pool criticality safety analyses and demonstrate that their fresh and spent fuel pool design meets the regulatory requirements of 10 CFR 50.68(b) and GDC 62.

9.1.1.4.1 Fuel Assembly Modeling

The application does not include criticality safety analysis. COL applications that adequately address the requirements of COL Items 9.1-1 and 9.1-2 would give reasonable assurance that the specific design will remain subcritical to meet the requirements of 10 CFR 50.68(b) and GDC 62.

9.1.1.4.2 Storage Rack Modeling

The application does not include criticality safety analysis. COL applications that adequately address the requirements of COL Items 9.1-1 and 9.1-2 would give reasonable assurance that the specific design will remain subcritical to meet the requirements of 10 CFR 50.68(b).

9.1.1.5 Combined License Information Items

Table 9.1-1, below, gives the COL information items related to criticality safety of new and spent fuel storage and handling. The staff reviewed these proposed COL information items and finds them to be acceptable for the reasons discussed in Section 9.1.1.4 above. COL Item 9.1-1 directs a COL applicant to develop plant programs and procedures, which will supplement design features, to ensure safe handling and storage of fuel. COL Item 9.1-2 requires the applicant referencing the US460 design to provide the design of the SFP storage racks, including the structural dynamic and stress analyses, thermal-hydraulic cooling analyses, criticality safety analysis, and material compatibility evaluation. The staff concludes that the COL information items provide sufficient information for applicants referencing the US460 design to design a fresh and spent fuel pool that meets the identified regulatory requirements.

Table 9.1-1 NuScale COL Information Items for Section 9.1

COL Item No.	Description	FSAR Section
9.1-1	An applicant that references the NuScale Power Plant US460 standard design will develop plant programs and procedures for safe operations during handling and storage of new and spent fuel assemblies, including criticality control.	9.1
9.1-2	An applicant that references the NuScale Power Plant US460 standard design will provide the design of the spent fuel pool storage racks, including the structural dynamic and stress analyses, thermal hydraulic cooling analyses, criticality safety analysis, and material compatibility evaluation.	9.1

9.1.1.6 Conclusion

The staff reviewed the general description of the fresh and spent fuel pool (SFP) design basis, and the proposed COL information items as described in Sections 9.1.1 and 9.1.2 of the FSAR. Based on its review of NuScale's statements and proposed COL information items, the staff concludes that a COL applicant adequately addressing the COL Items 9.1-1 and 9.1-2 will ensure the design of the fuel storage facilities and supporting systems is consistent with the Commission's regulations in GDC 62 and in 10 CFR 50.68.

9.1.2 New and Spent Fuel Storage

9.1.2.1 Introduction

The SFP provides onsite underwater storage of spent fuel assemblies and onsite underwater storage of new fuel assemblies. The SFP is designed to include the necessary design features unique to fuel storage during initial receipt, refueling operations, and accident conditions, including maintaining cooling and limiting offsite exposure in the event of a fuel handling accident.

NuScale describes the SFP SSCs related to fuel storage in FSAR Section 9.1.2. The staff evaluates the pool cooling and cleanup system (PCWS), fuel handling system, and ventilation separately.

9.1.2.2 Summary of Application

NuScale described new and spent fuel storage in FSAR Section 9.1.2. This facility provides for the storage of new and spent fuel assemblies. Figure 9.1.2-1 of the FSAR shows a high-level general arrangement of the SFP storage facility. The system functions are to maintain the fuel assemblies in a safe and subcritical array during all storage conditions. Section 9.1.2.3.7 of the FSAR further states that the SFP must conform with the regulatory requirements of 10 CFR 50.68(b). Figure 9.1.2-1 of the FSAR shows a high level general arrangement of the SFP storage facility.

NuScale included one COL information item in section 9.1.2, COL item 9.1-2, to require the COL applicant to provide the design of the spent fuel pool storage racks, including the structural dynamic and stress analyses, thermal hydraulic cooling analyses, and criticality safety analysis, and material compatibility evaluation.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): FSAR Part 8, "Table 3.5-1 "Fuel Storage System Inspections, Tests, Analyses, and Acceptance Criteria," provides ITAAC information for the fuel storage system. The staff evaluates ITAAC in Section 14.3 of this report.

9.1.2.3 Regulatory Basis

The Design Specific Review Standard (DSRS) (ML15356A584) for the NuScale Small Modular Reactor Design, Section 9.1.2, Revision 0, "New and Spent Fuel Storage," issued June 2016, gives the relevant regulatory requirements and the associated acceptance criteria, as summarized below, as well as the review interfaces with other SRP sections for this area of review:

- GDC 2, "Design bases for protection against natural phenomena," as it relates to the capabilities of the SSCs important to safety to be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

- GDC 4, "Environmental and dynamic effects design bases," as it relates to the capabilities of SSCs important to safety to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs), and the appropriate protection of these SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids.

- GDC 5, "Sharing of structures, systems, and components," as it requires that SSCs important to safety not be shared among nuclear power modules unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

- GDC 61, "Fuel storage and handling and radioactivity control," as it relates to the requirement that the fuel storage system be designed to ensure adequate safety under normal and postulated accident conditions.

- GDC 63, "Monitoring fuel and waste storage," as it relates to appropriate systems for detecting conditions that may result in the loss of residual heat removal capabilities for spent fuel assemblies, detecting excessive radiation levels, and initiating appropriate safety actions.

- 10 CFR 20.1101(b), as it relates to radiation doses kept as low as reasonably achievable (ALARA).

9.1.2.4 Technical Evaluation

The NRC staff reviewed FSAR Section 9.1.2, against the agency's regulatory guidance to ensure that the FSAR represents the complete scope of information relating to this review topic.

The recommendation in DSRS Section 9.1.2(iii)(3), indicates that the minimum SFP storage capacity of the facility should equal or exceed the amount of spent fuel from 5 years of operation at full power plus one full-core discharge.

FSAR Section 9.1.2.2.1 indicates that the SFP is designed for a maximum storage capacity of 600 fuel assemblies. A capacity of 600 storage locations covers more than 5 years of operation for six NuScale Power Modules (NPMs). This is consistent with the recommendations in DSRS Section 9.1.2(iii)(3). However, the FSAR does not include the design of the SFP storage racks. NuScale proposed COL Item 9.1-2, which instructs a COL applicant to provide the design of the SFP storage racks, including the structural dynamic and stress analyses, thermal hydraulic cooling analyses, criticality safety analysis, and material compatibility evaluation. However, the storage capacity will be evaluated at the COL review stage.

9.1.2.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

FSAR Section 9.1.2, states that the SFP is located within the RXB, which is a seismic Category I structure designed to protect from the effects of natural phenomena, including earthquakes, tornadoes, hurricanes, floods, and external missiles. FSAR Section 9.1.2 also states that new fuel assemblies and spent fuel assemblies are stored in the SFP.

The SFP concrete structures are designed to meet seismic Category I requirements. The staff's evaluation of the structural components is discussed in Section 3.8.4 of this Report. Therefore, they meet (1) Regulatory Position C.1, "Seismic Design," in RG 1.13, Revision 2, "Spent Fuel Storage Facility Design Basis," issued March 2007, which states that all structures and equipment necessary to safely maintain the conditions needed for radiation shielding should be designed to seismic Category I requirements, and (2) RG 1.29, Revision 6, "Seismic Design Classification for Nuclear Power Plants," issued July 2021, which states that all SSCs that must remain functional following a design-basis seismic event should be designed to seismic Category I criteria.

The main function of the SFP Liner, as described in FSAR Section 3.8.4.1.4.2, is to prevent potential pool inventory leakage. The liner is classified as nonsafety-related and not risk-significant. FSAR Table 3.8.4-5: "Classification of Structures, Systems, and Components," indicates that the UHS pool liner and dry dock liner are classified as seismic Category III, except (1) where the pool liner is integrated with the walls of the RXB that are steel composite, where the pool liner plates are designed to be seismic category I, and (2) where the pool liner may impact the safety system function, the pool liner plates are designed to be seismic Category II.

FSAR Section 3.2.1.2, "Seismic Category II," also states, in part, that any SSC that does not perform a safety-related function, but whose failure or adverse interaction could degrade the functioning or integrity of a seismic Category I SSC to an unacceptable level or could result in incapacitating injury to occupants of the control room during or following a safe shutdown earthquake (SSE), is designed to meet seismic Category II requirements. The staff finds that this is in accordance with the guidance in DSRS Section 9.1.2.III.4.C.

DSRS Section 9.1.2.III.5.B states that if the SFP liner plate is not designed and constructed to seismic Category I requirements, the SFP liner plate is reviewed for whether a failure of the liner plate because of an SSE will not cause any of the following:

- significant releases of radioactivity because of mechanical damage to the fuel

- significant loss of water from the pool that could uncover the fuel and lead to release of radioactivity because of heat-up
- loss of ability to cool the fuel because of flow blockage caused by a complete section or portion of the liner plate falling on the fuel racks
- damage to safety-related equipment because of pool leakage
- uncontrolled release of significant quantities or radioactive fluids to the environs

The staff evaluated the design of the SFP walls discussed in FSAR Section 3.8.4 and found that Figure 3B-10: "SC Walls used in Design Calculations" identifies the SFP walls as steel-plate composite (SC) walls. The SC walls are designed as seismic Category I walls. The staff finds that this is in accordance with the guidance in DSRS Section 9.1.2.III.5.B. Based on the above review, the staff concludes that the SFP meets the requirements of GDC 2, because it is designed to withstand the effects of natural phenomena without a loss of capability to perform its safety function.

9.1.2.4.2 GDC 4, "Environmental and Dynamic Effects Design Bases"

Compliance with GDC 4 requires that SSCs important to safety be designed to accommodate the effects of, and be compatible with, the environmental conditions of, in part, anticipated normal operating and postulated accident conditions. This includes protection against dynamic effects, including those of missiles, pipe whipping, and discharging fluids caused by equipment failures and from events and conditions outside the nuclear power unit.

DSRS Section 9.1.2 states that, for new and spent fuel storage facilities, GDC 4 requires a controlled and protected environment for the new and spent fuel and all associated SSCs important to safety. The SFP liner, the new and spent fuel assemblies, and the fuel storage racks must be protected from dynamic effects, including turbine and tornado missiles. Adequately thick SFP walls and adequate water levels usually provide the necessary protection from dynamic effects for SSCs within the pool. The new fuel and its storage racks also must be protected from dynamic effects to provide reasonable assurance that a substantial margin to criticality is maintained.

FSAR Section 9.1.2.1, "Design Bases," states that the RXB protect the stored fuel assemblies from the effects of natural phenomena hazards, including earthquakes, hurricanes, tornadoes, floods, tsunami, seiches, and external missiles.

DSRS Section 9.1.2.III.5.C states that the essential portions of the new and spent fuel storage facilities must be protected from the effects of floods, hurricanes, tornadoes, and internally or externally generated missiles.

The staff reviewed NuScale's system description in the FSAR and found that the stored fuel assemblies are located below grade and protected by a seismic Category I structure. This protection ensures that the stored fuel is adequately protected against natural phenomena hazards.

FSAR Section 3.5.1, "Missile Selection and Description," discusses the site missile protection features and states that the RXB exterior walls protect the essential SSCs located within from turbine missile penetration. The applicant stated that there is no turbine missile that can prevent essential systems from performing their function.

The staff discusses its evaluation of the missile protection methodology (including acceptability of the barriers) in Section 3.5.1 of this report.

Following the staff acceptance of NuScale's missile protection methodology as discussed in Section 3.5.1 of this report, the staff finds that locating the SFP inside the seismic Category I RXB in an area adequately protected from turbine missiles meets the recommendation of RG 1.13, ensures that the spent fuel storage facility is protected from turbine missiles and that the storage pool will retain watertight integrity since these missiles will not be able to strike it.

Based on the missile prevention design features identified above, the staff finds that the design of the SFP meets the requirements of GDC 4, in that SSCs important to safety are protected against the effects of missiles from events and conditions outside the nuclear power unit.

9.1.2.4.3 GDC 5, "Sharing of Structures, Systems, and Components"

GDC 5 requires that SSCs important to safety not be shared among nuclear power modules unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, during an accident in one unit, an orderly shutdown and cooldown of the remaining units. GDC 5 requires that the fuel storage facility at multiple-unit sites not be shared among the units, or, if shared, the shared SSCs must be designed so that an accident at one facility will not significantly impair the ability of the remaining facility to protect new and spent fuel.

FSAR Section 9.1.2.3, states that the NPMs can share the new and spent fuel storage facility for normal and accident conditions without impairing the performance of fuel storage facility or NPM safety functions, even with a postulated accident in one NPM and allowing for the safe shutdown of the remaining NPMs.

FSAR Section 9.2.5, "Ultimate Heat Sink," describes the SFP as a safety-related pool that is part of the safety-related UHS. This pool performs its safety function passively, by retaining a large volume of water under all accident scenarios, which allows the removal of decay heat from the stored fuel assemblies and cools the NPMs. The pool is designed to perform its intended safety function during all postulated events (including accidents); therefore, an accident in one NPM will not prevent the orderly shutdown and cooldown of the remaining NPMs.

The staff evaluated the description of the SFP, which states that the SFP is designed as a passive system, separated from the refueling pool and reactor pool and the NPM by a weir. Based on the separation between the SFP and the remainder of the UHS and the staff evaluation of the safety-related function of cooling the stored fuel and the NPMs discussed in Sections 9.1.3 and 9.2.5 of this report, the staff determined that sharing the SFP between the NPMs does not impair the performance of the SFP to retain adequate water inventory during all accident scenarios. Therefore, based on the information provided above, the staff finds that the SFP design meets the requirements of GDC 5.

9.1.2.4.4 GDC 61, "Fuel Storage and Handling and Radioactivity Control"

GDC 61 requires that the fuel storage system be designed for adequate safety under anticipated operating and accident conditions. The fuel storage system must be designed with: (1) the capability to permit appropriate periodic inspection and testing of components important to safety, (2) suitable shielding for radiation protection, (3) appropriate containment, confinement, and filtering capability, (4) residual heat removal capability that reflects the safety importance of decay heat and other residual heat removal, and (5) the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

FSAR Section 9.1.2.4 states that the design of the SFP facilitates inspection and testing of the pool surfaces, and that most of the surfaces are available for inspection. The staff considered the overall pool design and the location of safety components and confirmed that the pool design facilitates inspection and testing of components important to safety.

DSRS Section 9.1.2.III.4.H.i states that the SFP design should include weirs and gates separating the spent fuel storage areas from handling areas to prevent the accidental draining of the coolant to levels inadequate for fuel cooling or radiation shielding. The bottom of any gate should be above the top of the fuel assemblies, and the adjacent pool should be designed to prevent leakage that would reduce the coolant inventory below the minimum safety limit.

FSAR Section 9.2.5, describes the spent fuel pool, in conjunction with the reactor pool and the refueling pool (RFP), as part of the UHS. A weir separates the SFP from the other pools. The pools are designed as seismic Category I components and will remain leak tight after an SSE. The bottom of the weir that leads from the SFP to the RFP is 3 meters (m) (10 feet (ft)) above the top of the stored fuel assemblies. The UHS includes a designated dry dock area separated from the rest of the pool by seismic category II gate. FSAR Section 9.2.5 indicates that (1) a failure of the dry dock gate while the dry dock is empty is not expected to occur, due to the seismic classification of the gate and its supports, and (2) the SFP/UHS water level is maintained above the minimum safety limit.

The staff evaluated the applicant's description of the design of the dry dock gate and finds that because of the seismic design of the gates, no failure of these gates needs to be postulated. The staff finds that this meets the recommendations of DSRS Section 9.1.2.III.4.H.i because the SFP coolant water level can be maintained at a safe level for cooling and shielding.

FSAR Section 9.1.3.2.1, states that pool piping penetrations, by piping location or by anti-siphon protections, ensure the pool level cannot be siphoned below the 49.5 ft pool water level, which prevents draining or siphoning of the SFP water below the safe level.

The staff evaluated the minimum elevation for pipe penetrations and anti-siphon devices and determined that this elevation complies with the recommendation in DSRS Section 9.1.2.III.4.H.ii. The staff finds that locating piping penetrations and anti-siphon devices at this elevation ensures the SFP/UHS coolant water level can be maintained at a safe level for cooling and shielding.

Based on its review of the SFP design features reviewed above, the staff finds that the SFP is designed with: (1) the capability to permit appropriate periodic inspection and testing of components important to safety, (2) suitable shielding for radiation protection, (3) appropriate containment, confinement, and filtering capability, (4) residual heat removal capability that reflects the safety importance of decay heat and other residual heat removal, and (5) the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions in the SFP. Therefore, the staff finds that the SFP design meets the requirements of GDC 61.

9.1.2.4.5 GDC 63, "Monitoring Fuel and Waste Storage"

GDC 63 requires appropriate systems for fuel storage, radioactive waste, and handling areas to detect conditions that may result in a loss of residual heat removal capability and excessive radiation levels and to initiate appropriate safety actions.

For spent fuel storage facilities, GDC 63 requires monitoring of the SFP water level, pool temperature, and pool building radiation levels to protect personnel, prevent significant offsite radiation doses, and detect conditions that could cause the loss of decay heat removal capabilities. In addition, alarms and communications systems must alert personnel and provide for communications between the fuel handling machine (FHM) and the control room. If necessary, to limit offsite dose consequences from a fuel handling accident or pool boiling, instrumentation should automatically place the spent fuel facility ventilation system in a mode to reduce the offsite release of radioactive material.

NuScale discusses the SFP level and temperature instrumentation in FSAR Section 9.1.3, "Pool Cooling and Cleanup System," and the staff evaluates it in Section 9.1.3 of this report. NuScale described normal and accident operation of the SFP area ventilation system in FSAR Section 9.4.2, "Reactor Building and Spent Fuel Pool Area Ventilation System." The staff evaluates the SFP area ventilation system in Section 9.4 of this report.

DSRS Section 9.1.2.III.4.K instructs the staff to verify that the design incorporates the detection and collection of SFP liner leaks, with the capability to collect pool liner leaks (e.g., through drains and sumps) to prevent uncontrolled releases of radioactive material to the environment and to keep radiation exposure ALARA for personnel.

FSAR Section 9.1.3, describes the NuScale pool leakage detection system (PLDS), which monitors, collects, and routes possible UHS liner leakage. The channels are sized to allow for inspection and the cleaning of buildup. The channels collect leakage from the pool liner plates and direct it to a sump or to collection header piping leading to a sump in the radioactive waste drain system (RWDS). The RWDS provides local and control room indication and associated alarms when the leakage rate from the PLDS reaches a predetermined level.

FSAR Section 9.1.2.3.5, "Monitoring," states that the SFP area is provided with radiation monitors to detect both general area radiation levels and airborne contamination levels. FSAR Section 12.3, "Radiation Protection Design Features," gives additional information on the radiation area monitors. The staff evaluation of NuScale's radiation protection design features is in Section 12.3 of this report.

Based on the design features reviewed above, the staff finds that the NuScale US460 design meets GDC 63 and provides assurance that a loss of residual heat removal capability and high radiation levels would be detected and that the release of radioactive materials to the environment would be prevented.

9.1.2.4.6 As Low as Reasonably Achievable Principle

Compliance with 10 CFR 20.1101(b) requires the licensee to use, to the extent practicable, procedures and engineering controls based on sound radiation protection principles to achieve ALARA occupational doses and doses to the public.

DSRS Section 9.1.2 describes staff positions and ANS guidance for the fuel storage facility meant to achieve radiation doses in compliance with the ALARA principle. Controlled drainage for the SFP limits the spread of contamination from leakage of the pool liner. Smooth and nonporous surfaces for all components in contact with contaminated coolant (e.g., the SFP liner) avoid unnecessary buildup of radioactive material. Appropriate shielding of spent fuel also ensures compliance with the ALARA principle.

FSAR Section 9.1.2.3.6, "Radiation, Shielding, and Maintaining Doses as Low as Reasonably Achievable," states that the PLDS limits the spread of contamination from liner leakage. The PLDS collects pool leakage and directs it to the waste collection system. Section 9.1.3 of this report discusses and evaluates the PLDS.

FSAR Section 9.1.2.1 states that the surfaces in contact with the pool water are smooth and nonporous to prevent the buildup of radioactive material. The storage racks are not part of the SDAA and have not been evaluated in this report. The SDAA proposed COL Item 9.1-2 requires the COL Applicant to provide the design of the spent fuel pool storage racks, including the material compatibility evaluation.

Section 12.1 of this report gives the staff's complete evaluation of the ALARA design and decontamination details. The staff finds that the features discussed above meet the recommendation of DSRS Section 9.1.2, "New and Spent Fuel Storage," and therefore comply with the ALARA principle.

9.1.2.5 Initial Test Program

The staff evaluates the initial test program (ITP) in Section 14.2 of this SER.

9.1.2.6 Technical Specifications

NuScale has not identified any generic technical specifications (GTS) evaluated in this section of the report. The staff reviewed the FSAR and concluded that no TS are required for the SFP, specifically, as other TS cover this body of water. Section 9.2.5 of this report evaluates GTS related to the SFP and UHS water level and temperature. GTS addressing criticality in the SFP are evaluated in Section 9.1.1 of this report. The SFP is part of the ultimate heat sink (UHS) and shares the same volume of water.

9.1.2.7 Combined License Information Items

FSAR, Section 9.1.2.2.2 "Fuel Storage Racks Design," describes COL Item 9.1-2 (Table 9.1-1), which directs a COL applicant that references the NuScale Power Plant US460 standard design to provide the design of the SFP storage racks, including the structural dynamic and stress analyses, thermal hydraulic cooling analyses, criticality safety analysis, and material compatibility evaluation.

The FSAR does not include the SFP storage rack design; therefore, a COL application must provide the design of the SFP storage racks, and the analysis identified in the COL information item. The staff evaluated the proposed COL information item in Section 9.1.1.5 of this report and found it acceptable. The staff finds that no other COL information item is needed.

9.1.2.8 Conclusion

The staff evaluated the new and spent fuel storage for the NuScale US460 Standard design in accordance with the guidance of SRP Section 9.1.2, "New and Spent Fuel Storage." For the reasons provided above, the staff finds that the SFP design meets the requirements of GDC 2, 4, 5, 61 and 63 and 10 CFR 20.1101, "Radiation Protection Programs." Section 9.1.1 of this report documents the staff's evaluation of the criticality safety evaluation of the fresh and spent fuel pool.

9.1.3 Pool Cooling and Cleanup System

9.1.3.1 Introduction

All nuclear reactor plants include an SFP for the wet storage of spent fuel assemblies. The methods used to provide cooling for the removal of decay heat from the stored assemblies vary from plant to plant, depending upon the individual design. The safety function to be performed by the system in all cases remains the same; that is, the spent fuel assemblies must be cooled and must remain covered with water during all storage conditions.

FSAR Section 9.1.3 discusses the design and performance of the pool cooling and cleanup system (PCWS). The PCWS consists of three subsystems: (1) the pool cooling subsystem, (2) the pool cleanup subsystem, and (3) the pool surge control subsystem. Each of these systems performs a different function. The PCWS is nonsafety-related and not risk-significant, provides for water level control and temperature maintenance of the reactor pool, the RFP and the SFP. The PLDS is nonsafety-related and not risk significant and provides collection, redirection, and measurement of leakage from the UHS and dry dock.

The active pool cooling system removes decay heat from the stored fuel and the NPM in the reactor pool and RFP during normal operation. The pool cooling and cleanup sub system removes impurities to reduce radiation dose rates and maintain water chemistry and clarity in the UHS pools and dry dock. The pool surge control subsystem drains the dry dock using the evacuation pumps to support maintenance and refueling activities. It transfers and stores excess water volume from the UHS to maintain the required water level in the pools during surge events.

During accident scenarios, the NuScale US460 design credits the safety-related water inventory stored in the UHS to passively remove the decay heat. The staff evaluates the UHS in Section 9.2.5 of this report.

9.1.3.2 Summary of Application

NuScale described the system in FSAR, Section 9.1.3. The SFP, RFP and the reactor pool are connected and share the volume of water of the UHS above the weir. The PCWS functions during normal operations. FSAR, Section 9.2.5, discusses the safety-related function of cooling the stored fuel during and after an accident scenario. The PCWS removes impurities from the UHS pools and the dry dock and is not a safety-related system.

Initial Test Program: Inspection and testing of the PCWS are performed before plant operation, as described in FSAR, Table 14.2-1: Test # 01 Pool Cooling and Cleanup System and Table 14.2-3: Test # 03 Pool Leakage Detection System.

Technical Specifications: SDAA Part 4, Limiting Condition for Operation (LCO) 3.5.3, "Ultimate Heat Sink," is related to the UHS (which includes the SFP) water level, maximum initial temperature, and minimum boron concentration.

9.1.3.3 Regulatory Basis

DSRS Section 9.1.3, Revision 0, "Spent Fuel Pool Cooling and Cleanup System," issued June 2016, gives the relevant regulatory requirements for this area of review and the associated acceptance criteria, as summarized below, as well as the review interfaces with other SRP sections.

- GDC 2, as it relates to SSCs important to safety being designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, and hurricanes. GDC 2 is not applicable to the cleanup portion of the system and need not apply to the cooling system if both the fuel pool makeup water system (and its source) and the auxiliary building (and its ventilation and filtration system) meet this criterion.

- GDC 4, as it relates to the requirement that SSCs important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs and dynamic effects resulting from pipe whip, missiles, and discharging fluids.

- GDC 5, as it relates to SSCs important to safety not to be shared among nuclear power units unless it can be shown that such sharing will not impair their ability to perform their safety functions

- GDC 61, as it relates to the requirement that the fuel storage system be designed to ensure adequate safety under normal and postulated accident conditions, including the capability to permit appropriate periodic inspection and testing of components important to safety; suitable shielding for radiation protection; appropriate containment, confinement, and filtering capability; residual heat removal capability that reflects the importance to safety of decay heat and other residual heat removal; and the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

- GDC 63, as it relates to monitoring systems provided to (1) detect conditions that could result in the loss of decay heat removal capability, (2) detect excessive radiation levels, and (3) initiate appropriate safety actions.

- 10 CFR 20.1101, as it relates to radiation doses being kept ALARA.

9.1.3.4 Technical Evaluation

The PCWS consists of three trains, each with an inlet strainer, a pump, and a heat exchanger. The heat exchangers are cooled with water from the site cooling water system (SCWS). The SFP is connected to the RFP and the reactor pool, forming the UHS. During normal operation, the suction and discharge lines in the SFP are on opposite corners of the SFP, the pool cooling suction and discharge in the reactor pool and RFP are located in the RFP and the common discharge header is located at each module operating bay in the reactor pool. NuScale stated that this configuration ensures proper mixing of the water in the SFP, reactor pool, and RFP.

9.1.3.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

Compliance with GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena combined with the appropriate effects of normal and accident conditions without a loss of capability to perform their safety functions. The acceptance criteria for meeting GDC 2 are based on conformance to RG 1.13, Regulatory Positions C.1, C.2, C.6, and C.8; and RG 1.29, Regulatory Position C.1, for safety-related portions of the system, and Regulatory Position C.2, for portions of the system that are not safety-related.

RG 1.13, Regulatory Position C.1, states that the spent fuel storage facility, including all structures and equipment necessary to maintain the minimum water levels needed for radiation shielding, should be designed to seismic Category I requirements. RG 1.13, Regulatory Position C.2, states that the spent fuel storage facility should be designed to (1) keep extreme winds and missiles generated by those winds from causing significant loss of watertight integrity of the fuel storage pool, and (2) keep missiles generated by extreme winds from contacting fuel within the pool. RG 1.13, Regulatory Position C.6, "Drainage Prevention," states that the drains, permanently connected mechanical or hydraulic systems, and other features that (by maloperation or failure) could reduce the coolant inventory to unsafe levels should not be installed or included in the design. RG 1.13, Regulatory Position C.8, "Makeup Water," states that a Quality Group C, seismic Category I makeup system should be provided to add coolant to the pool. RG 1.29, Regulatory Position C.1, lists SSCs, including their foundations and supports that should be designed to withstand the effects of the SSE and remain functional. RG 1.29, Regulatory Position C.2, states that any portion of SSCs that are not required to remain functional after an SSE but could still reduce the functioning of any plant feature that is required to remain functional to an unacceptable safety level or could result in incapacitating injury to occupants of the control room, should be designed and constructed so that the SSE would not cause such failure.

The pool cooling and cleanup system components are located within the RXB structure. The RXB is classified as seismic Category I and is designed to withstand the effects of earthquakes, tornadoes, hurricanes, floods, external missiles, and other natural phenomena, as described in FSAR, Section 3.3, "Wind and Tornado Loadings," through Section 3.8, "Design of Category I Structures." The UHS structural walls are designed to seismic Category I standards.

The active PCWS is not a safety-related system. The water inventory stored in the UHS performs the safety-related function of maintaining the stored fuel and the NPM cooled during design-basis events (DBEs). The staff evaluates the UHS performance in Section 9.2.5 of this report.

FSAR, Section 9.2.5, identifies the minimum safety water level needed to provide safety-related cooling to the NPMs as 14.7 m (48.2 ft) from the bottom of the pool. The staff confirmed that elevations of all pipe openings or antisiphon devices on the piping are above the minimum pool water level. FSAR, Section 9.1.3.2.1, "Pool Cooling Subsystem," states that the pool piping penetrations, by piping location or by anti-siphon protections, ensure the pool level cannot be siphoned below the 15.1 m (49.5 ft) pool water level. Therefore, the PCWS has no penetrations into the UHS/ SFP.

FSAR 9.1.3, states that the PCWS is classified as seismic Category III, However, piping or structures with the potential for adverse interactions with seismic Category I SSCs, are designed as seismic Category II.

DSRS Section 9.1.2.III.4.C states that nonsafety-related SSCs not designed to seismic Category I standards located in the vicinity of the new and spent fuel storage facilities are reviewed to determine whether their failure would cause an increase in k_{eff} to more than the maximum allowable.

The staff evaluated the system description in FSAR, Section 9.1.3, and finds that the UHS and the SFP are designed to seismic Category I standards, and all pool penetrations and antisiphon devices are located above the minimum safety water level, which ensures that a failure of these components does not adversely impact the safety function of the safety-related UHS/SFP.

9.1.3.4.2 GDC 4, "Environmental and Dynamic Effects Design Bases"

Compliance with GDC 4 requires that SSCs important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs and dynamic effects resulting from pipe whip, missiles, and discharging fluids.

FSAR, Section 9.1.3.1, states that the pool cleanup subsystem and pool cooling subsystem are located inside the seismic Category I RXB. The pool surge control subsystem is also located inside the RXB, except for the pool surge control storage tank and associated piping and valves, which are located outside. The systems are designed to be compatible with the environmental conditions expected during normal operations. For accident scenarios, these systems do not adversely impact the safety function of the safety-related UHS/SFP. All piping connections and antisiphon devices are located above the minimum safety water level.

FSAR, Section 3.7.3, states that non-seismic Category I SSCs that could adversely affect seismic Category I SSCs are categorized as seismic Category II.

The FSAR states that the fuel handling machine is designed to seismic Category I requirements. The staff evaluates the FHM and its operation in Section 9.1.4 of this report. The RXB is designed to seismic Category I requirements. The staff evaluates the overhead heavy load handling system (OHLHS) in Section 9.1.5 of this report.

The staff finds that failures of the PCWS that are not safety-related would not adversely impact safety-related SSCs. The design of the FHE and the OHLHS precludes the drop of heavy loads in the SFP area. Therefore, the staff finds that NuScale's design meets the recommendations of SRP Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," and complies with GDC 4 requirements, in that SSCs important to safety are protected against the effects of missiles from events and conditions outside the nuclear power plant.

9.1.3.4.3 GDC 5, "Sharing of Structures, Systems, and Components"

GDC 5 requires that SSCs important to safety not be shared among nuclear power modules unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, during an accident in one module, an orderly shutdown and cooldown of the remaining modules. To meet GDC 5, the fuel storage facility at multiple-unit sites must not be shared among the units or, if shared, the shared SSCs must be designed so an accident at one facility will not significantly impair the ability of the remaining facility to protect new and spent fuel.

In FSAR, Section 9.1.2.3, NuScale stated that the NPMs can share the new and spent fuel storage facility for normal and accident conditions without impairing the performance of fuel storage facility or NPM safety functions, even with a postulated accident in one NPM and allowing for the safe shutdown of the remaining NPMs.

FSAR, Section 9.2.5, describes the SFP as a safety-related pool that is part of the safety-related UHS. These pools perform their safety function passively, by retaining a large volume of water under all accident scenarios which allows the removal of decay heat from the stored fuel assemblies and provides cooling of the NPMs. These pools are designed to perform their intended safety function during all postulated events (including accidents); therefore, an accident in one NPM would not prevent the orderly shutdown and cooldown of the remaining NPMs. The staff evaluates the safety-related function of cooling the stored fuel and the NPMs in Section 9.2.5 of this report. The staff determined that sharing the SFP between the NPMs does not impair the performance of the SFP in retaining adequate water inventory during all accident scenarios because no single failure will impair the system from performing its safety functions. Therefore, the staff finds that the design meets the requirements of GDC 5.

9.1.3.4.4 GDC 61, "Fuel Storage and Handling and Radioactivity Control"

Compliance with GDC 61 requires that the fuel storage system be designed to ensure adequate safety under normal and postulated accident conditions. SRP Section 9.1.3 specifies that in order to meet GDC 61, the system shall be designed with the following attributes:

- the capability to permit appropriate periodic inspection and testing of components important to safety
- suitable shielding for radiation protection
- appropriate containment, confinement, and filtering capability
- residual heat removal that reflects the importance to safety of decay heat and other residual heat removal
- the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions

FSAR, Section 9.1.3, states that major trains or pieces of equipment in the PCWS are provided with isolation valves that are located to allow for systematic inservice inspections, periodic maintenance, repairs, and functional testing. NuScale stated that laydown spaces are provided for pump and heat exchanger disassembly and maintenance. Pull spaces are also provided for the heat exchanger tube bundles and head removal. The leakage channels in the PLDS are accessible for inspection.

FSAR, Section 14.3 discusses initial plant testing for the PCWS. These tests ensure that the systems are capable of performing their design functions. As discussed above, the staff finds that the design features discussed in FSAR, Section 9.1.3.4, "Inspection and Testing," ensure that pool support systems are designed to permit periodic inspection and testing of their components.

DSRS Section 9.1.3.III.3 instructs the NRC reviewer to verify the functional performance requirements of the pool cooling systems to confirm that they address the minimum system heat transfer and system flow requirements for normal plant operation.

FSAR, Section 9.1.3, discusses the pool cooling and cleanup subsystem, which consists of three trains, cooled with water from the SCWS. The pool cooling subsystem is not a safety-related system, and during normal plant operation, it runs continuously using two trains with the third one on standby. The suction and discharge lines are on opposite corners of the SFP, with intakes in the north and south walls of the RFP, and discharge into each of the NPM bays in the reactor pool.

FSAR, Section 9.1.3.3.4, states that the pool cooling subsystem is designed to maintain the pool bulk temperature below 48.9 degrees Celsius (°C) (120 degrees Fahrenheit (°F)). NuScale described the pool cooling capability under several scenarios. NuScale stated that the pool cooling system is capable of maintaining the pool water temperature at or below the normal temperature (37.8 °C (100 °F)), with at least two trains during normal operation. During off-normal heat load conditions (a full core offload), the heat load increases, and two pool cooling sets of equipment are required to maintain the pools at or below 48.9 °C (120 °F).

The staff evaluated the description of the PCWS and its performance requirements, discussed in FSAR Section 9.1.3.3.4 and Table 9.1.3-1 “Equipment Parameters for the Pool Cooling Subsystem” and determined that the PCWS is design with adequate heat transfer and system flow requirements for normal plant operation.

DSRS Section 9.1.3.III.3.D recommends that the pool cooling subsystem should retain at least half of its full heat removal capacity assuming a single active failure. This minimum heat removal capacity shall provide reasonable assurance that the pool temperature will remain within design bounds for the structure during full core discharges to the SFP when the forced-circulation cooling system is in operation and ensures that significant heat removal capacity will remain available. The staff evaluated the heat loads of the various scenarios in the FSAR and the subsystem design capability and found that the pool cooling subsystem has sufficient heat removal capability to ensure adequate pool cooling during normal operation.

DSRS Section 9.1.3.III.3.B states that the cooling loop may be constructed to nonseismic Category I requirements, provided the SFP water makeup system and the building ventilation and filtration system (1) are designed to Quality Group C and seismic Category I requirements, (2) are protected from the effects of tornadoes, and (3) meet the single-failure requirements.

Where the cooling loop is constructed to nonseismic Category I requirements, the ventilation system provides the capability to vent steam and moisture to the atmosphere to protect safety-related components from the effects of boiling in the SFP. If necessary to limit the offsite dose consequences of SFP boiling, the ventilation and filtration system should also meet the guidelines of RG 1.52, “Design, Inspection and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light -Water-Cooled Nuclear Power Plants.”

FSAR, Section 9.2.5.2.2, stated that to prevent over-pressurization in the UHS area of the RXB during abnormal conditions, over-pressurization vents are included in the RXB system. If power is available, NuScale credited the RXB heating, ventilation, and air conditioning (HVAC) system (RBVS), which will filter and control the release of airborne radioactive material from inside the

RXB. Once pressure in the RXB reaches the setpoint, the passive over-pressurization vent (OPV) opens and releases the RXB pressure and prevents over-pressurization of the building.

The staff evaluated NuScale's description of the over-pressurization vents in FSAR Table 3.9-17, "Valve Inservice Test Requirements per ASME OM Code," which describes it as a non-reclosing pressure relief device (rupture disk). The staff finds that the passive over pressurization vent conforms to the guidance in DSRS Section 9.1.3.III.3.B.

DSRS Section 9.1.3.III.3.E states that the pool cooling systems should be designed so that in the event of failure of inlets, outlets, piping, or drains, the pool level will not be inadvertently drained below the minimum safety water level. Pipes or external lines extending into the pool that are equipped with siphon breakers, check valves, or other devices to prevent drainage are acceptable as a means of implementing this requirement.

In FSAR, Section 9.2.5, the applicant identified the minimum safety water level needed to provide safety-related cooling to the NPMs as 14.7 m (48.2 ft) from the bottom of the pool. The staff confirmed that elevations of all pipe openings or antisiphon devices on the piping are above the minimum pool water level. FSAR, Section 9.1.3.2.1, "Pool Cooling Subsystem," states that the pool piping penetrations, by piping location or by anti-siphon protections, ensure the pool level cannot be siphoned below the 15.1 m (49.5 ft) pool water level.

Section 9.2.5 of this report presents the staff evaluation of the accident scenarios that determined the 14.7 m (48.2 ft) water level as the minimum safety limit. Based on the discussion above, the staff finds that the SFP design prevents any failure from inlets, outlets, piping, or drains to lower the SFP level below the minimum safety water level.

DSRS Section 9.1.3.III.3.F states that a seismic Category I, Quality Group C, makeup system and a backup method should add coolant to the SFP. The backup system should also be installed permanently, physically separate and independent from the primary makeup system, and designed to seismic Category I, Quality Group C, standards. The minimum makeup capacity for each system should exceed the larger of the pool leakage rate (assuming SFP liner perforation resulting from a dropped fuel assembly) or the maximum evaporating rate.

The staff evaluated FSAR, Section 9.1.3, which describes the makeup sources available for the SFP. The demineralized water system (DWS) is not safety-related and supplies normal makeup water to the PCWS, with the liquid radioactive waste system (LRWS) providing alternate makeup. For an accident condition that disables the normal makeup supply and the active cooling systems, the large volume of water in the UHS is designed to maintain sufficient inventory of cooling water, such that no makeup water is needed for at least 30 days.

FSAR, Sections 9.1.3.3.5 and 9.2.5, describe a single seismic Category I makeup line from the outside of the building into the SFP. This 10.2-centimeter (4-inch) diameter line is sloped and has the capacity to provide several times the required water makeup flow.

The staff evaluated the design of the NuScale US460 SFP, which is different from that of a typical large, pressurized water reactor. The total volume of water available for passive cooling of the stored fuel provides assurance that makeup water is not needed for at least 30 days. This allows sufficient time for the operator to assess the availability of the makeup sources to the SFP. The SFP is provided with a single seismic Category I makeup line from the outside. Therefore, based on the large inventory of safety-related water available to cool the stored fuel, the fact that no makeup is required for at least 30 days, and the availability of a seismic

Category I makeup line from the outside, the staff finds that NuScale's US460 design of the SFP/UHS has adequate capability to prevent a significant reduction in the fuel storage coolant inventory.

Based on the discussion above, the staff finds that the PCWS design (1) is capable of providing suitable shielding for radiation protection and appropriate containment, confinement, and residual heat removal and (2) has the capability to prevent a significant reduction in fuel storage coolant inventory under normal conditions. Therefore, the staff finds that the PCWS conforms to the applicable requirements of GDC 61 and is therefore acceptable.

The Pool Cleanup Subsystem

DSRS Section 9.1.3.III.8 states that the cleanup system should have the capacity and capability to remove corrosion products, radioactive materials, and impurities so that water clarity and quality will enable safe operating conditions in the pool.

The staff evaluated the description of the pool cleanup subsystem, which is not a safety-related system. The pool cleanup subsystem maintains water chemistry and removes particulates from the UHS. The pool cleanup subsystem has a design function that is not safety-related to clean up impurities in the UHS. Cleanup of the UHS ensures that plant operations, such as movement of power modules or fuel assemblies, can be conducted with minimal radiation exposure and without particulates obscuring the vision of personnel or operators.

In accordance with DSRS Section 9.1.3, the staff reviewed the capacity and capability of the cleanup system to ensure safe operating conditions for the pool. The applicant described the capacity of the pool cleanup subsystem in FSAR, Table 9.1.3-2, "Equipment Parameters for the Pool Cleanup Subsystem."

The pool cleanup subsystem has the capability to process the full volume of the UHS every 2 months. The three trains in the pool cleanup subsystem ensures that processing the resin beds or other operations that may make a single train of the pool cleanup subsystem inoperable would not impact normal or refueling operations.

FSAR, Table 9.1.3-4: "Water Chemistry Parameters Monitored for the Ultimate Heat Sink Pools," defines the requirements for the pool cleanup subsystem. The staff reviewed the UHS water chemistry parameters and found the chloride, fluoride, and sulfate values to be consistent with Appendix B.7, "Spent Fuel Pool Cooling and Cleanup System," to the Electric Power Research Institute (EPRI) 3002000505," Pressurized Water Reactor Primary Water Chemistry Guidelines," dated April 24, 2014 (EPRI Guidelines). The applicant provided values for silica, suspended solids, and gamma isotopic activity, which are suggested parameters for monitoring in the EPRI Guidelines.

Based on the system features discussed above, the staff concludes that the pool cleanup subsystem has sufficient capability and capacity to remove corrosion products, radioactive materials, and impurities from the UHS. The staff also finds that the pool cleanup subsystem has the capability to maintain the UHS water chemistry to prevent corrosion of the spent fuel assemblies, the SFP, and the NuScale modules. The staff finds that the design of the pool cleanup subsystem will be able to perform the functions described in the FSAR that are not safety-related (but are important to safety) associated with the removal of impurities and maintaining SFP and RFP water chemistry and clarity.

9.1.3.4.5 GDC 63, “Monitoring Fuel and Waste Storage”

Compliance with GDC 63 requires, in part, that appropriate systems be provided in the fuel storage area to detect conditions that may result in the loss of residual heat removal capability or excessive radiation levels and to initiate appropriate safety actions.

FSAR, Section 9.1.3.3.6, “Monitoring Cooling Capability and Area Radiation Levels,” states that radiation monitors are provided for detecting excessive radiation levels in the SFP area of the RXB. FSAR, Section 12.3.4, “Area Radiation and Airborne Radioactivity Monitoring Instrumentation,” contains additional information on the monitors.

DSRS Section 9.1.3.III.5 states that the cooling system should include features to contain radioactivity. The system should include means to detect, identify, and notify the staff of system leakage (i.e., sumps, collection, intersystem leakage identification) and isolation capabilities.

The FSAR states that the PLDS collects water leaking from the pool liner and routes the leaked water to the RWDS. In Section 9.1.2 of this report, the staff evaluates the design’s capability to detect liner leakage.

The RWDS provides local and control room alarms and indications of the presence of liner leakage. Additionally, the pool cooling system heat exchangers are cooled with water from the SCWS. The applicant stated that the design incorporates the means to detect intersystem leakage (i.e., radiation monitors and conductivity monitors).

The staff evaluated the system description in FSAR, Section 9.1.3, and determined that the cooling systems incorporate the means to identify leakage from the pools and to inform the operating staff of system conditions. The staff found these features to be in accordance with the recommendation in DSRS Section 9.1.3.III.5.

The staff finds that the system features discussed above demonstrate that the pool support systems are provided with monitoring and detection capabilities to ensure that the systems are capable of performing their intended safety function and, therefore, meet the requirements of GDC 63.

9.1.3.4.6 10 CFR 20.1101, “Radiation Protection Programs”

Compliance with 10 CFR 20.1101(b) requires that the licensee use, to the extent practicable, procedures and engineering controls based on sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.

As discussed previously in this SER, the applicant’s design incorporates the PLDS, which collects and identifies pool liner leakage. This prevents long-term unidentified pool leakage and accumulation of radioactive fluids. The design also includes the means to identify intersystem leakage that could contaminate systems that are not typically radioactive. The majority of the pool support systems are located in the RXB, where the RWDS collects any leakage and routes it to the LRWS for further processing.

The pool surge control storage tank is located outside the RXB in the plant yard. The secondary containment tank around the pool surge control storage tank has sufficient volume to store the pool surge control storage tank volume plus the contents of related piping. The applicant indicated that the secondary containment tank is designed to prevent leakage to the environment and is connected to a sump with valves and piping to direct collected fluids to the LRWS.

The system description in FSAR, Section 9.1.3.2.3, indicates that a guard pipe is provided where the pool surge control storage tank piping is embedded underground or in a yard area pipe chase. The leakage from a pipe into the guard pipe is detected with periodic surveillance of PCWS piping.

The staff evaluated FSAR, Section 9.1.3, for features that ensure the applicant is implementing sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA. The staff determined that the applicant’s design includes the means to reduce the level of contamination and propagation of contaminated fluids, detect leakages, and implement pipe guard systems. These provisions are in accordance with the recommendation in DSRS Section 9.1.3; therefore, the staff determines that the applicant’s design of the pool support systems meets the requirements of 10 CFR 20.1101(b).

9.1.3.5 Initial Test Program

FSAR, Section 14, Table 14.2-1: Test # 01 “Pool Cooling and Cleanup System," and Table 14.2-3: Test # 03 “Pool Leakage Detection System,” list the preoperational test requirements for the PCWS. The test ensures the PCWS is capable of performing its intended functions identified in FSAR, Section 9.1.3.

Section 14.2 of this report presents the staff’s evaluation of the ITP.

9.1.3.6 Technical Specifications

No TS requirements are directly associated with the PCWS. US460 SDAA Part 4, Volume 1, TS 3.5.3 “Ultimate Heat Sink” addresses the maximum UHS/SFP bulk temperature and minimum water level. Section 9.2.5 of this report discusses this TS.

Based on a graded approach commensurate with the safety significance of the SSCs, the staff concludes that TS are not required for the PCWS because this system does not meet the criteria for assigning an LCO in 10 CFR 50.36(c)(2). Therefore, the staff finds this acceptable.

9.1.3.7 Combined License Information Items

The applicant did not propose any COL information item for the PCWS or PLDS. These reviews of SDAA, Section 9.1.3, did not identify a need for any additional COL information item for this system.

9.1.3.8 Conclusion

The staff evaluated the PCWS and the PLDS for the NuScale US460 design in accordance with the guidance of SRP Section 9.1.3. Based on the staff’s evaluation discussed above, the staff finds that the design of the system meets the requirements of GDC 2, 4, 5, 61, and 63 and 10 CFR 20.1101 and is, therefore, acceptable.

9.1.4 Light Load Handling System (Related to Refueling)

9.1.4.1 Introduction

The fuel handling equipment (FHE) handles, moves, and stores fuel assemblies and control rod assemblies during fuel transfer operation. The FHE system is an integrated system of equipment and tools for refueling, handling, and storing fuel assemblies from receipt of the new fuel shipping container to shipment of the spent fuel cask.

9.1.4.2 Summary of Application

FSAR, Section 9.1.4, "Fuel Handling Equipment," gives the design bases, description, and safety evaluation of the FHE, the RFP, and the refueling floor. The major components of the FHE system are the FHM, the new fuel jib crane (NFJC), and the new fuel elevator (NFE). The areas of the facility associated with the FHE are the SFP, the RFP, and the new fuel staging areas, which are all enclosed within the RXB. FSAR, Table 9.1.4-1, lists design information for the three major components. FSAR, Section 9.1.4.2.2, "Major Component Description," describes each major component.

ITAAC: FSAR Part 8, Section 3.4, "Fuel Handling Equipment System" Table 3.4-1, "Fuel Handling Equipment System Inspections, Tests, Analyses, and Acceptance Criteria," specifies the ITAAC for the FHE. Section 14.3 of this report evaluates these ITAAC.

Initial Test Program: FSAR, Table 14.2-44, Test # 44 "Fuel Handling Equipment" describes the performance testing for the FHE system. Section 14.2 of this report evaluates the ITP.

9.1.4.3 Regulatory Basis

SRP Section 9.1.4, Revision 4, "Light Load Handling System and Refueling Cavity Design," issued July 2014, gives the relevant regulatory requirements for this area of review and the associated acceptance criteria, as summarized below, as well as the review interfaces with other SRP sections:

- GDC 2, as it relates to SSCs important to safety being capable of withstanding the effects of earthquakes

- GDC 5, as it relates to the sharing of equipment and components important to safety among multiple operating units at one single site, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, during an accident in one unit, an orderly shutdown and cooldown of the remaining units

- GDC 61, as it relates to fuel storage and handling systems being designed to ensure adequate safety under normal and postulated accident conditions.

- GDC 62, as it relates to preventing criticality in the fuel storage and handling system, preferably by use of geometrically safe configurations.

9.1.4.4 Technical Evaluation

The NRC staff reviewed FSAR, Section 9.1.4, against SRP 9.1.4 to ensure that the FSAR represents the complete scope of information relating to this review topic. The FHE system is used in the SFP, the RFP, and the refueling flooring the RXB. FSAR, Figure 9.1.4-1, "Refueling Floor Layout," gives the design layout of the FHE system.

The FHM is a traveling bridge and trolley crane that rides on rails with hard stops to prevent bridge wheels from moving past the end of the rails. It is used to transport fuel assemblies between the open lower reactor vessel in the RFP and the fuel assembly storage racks in the SFP during the refueling outage of one NPM. The FHM is also used to transport new fuel assemblies between the new fuel elevator and their storage locations within the SFP. In addition, the FHM is used to move spent fuel assemblies from their storage locations to the spent fuel cask in the RFP. FSAR, Section 9.1.5, "Overhead Heavy Load Handling Systems," describes the design and operation of the reactor building crane (RBC). Section 9.1.5 of this report documents the associated staff evaluation.

The NFJC is a jib crane mounted to the refueling floor and has a hoist that moves across a jib beam on the refueling floor that rotates around the stationary base of the crane. It handles new fuel in the new fuel staging area. The crane is used to remove new fuel assemblies from their shipping containers, support the assemblies during subsequent inspections, and move the assemblies to the new fuel elevator. The NFJC is also used to transport new control rod assemblies and other light load components that are placed in the SFP.

The NFE includes fixed rails mounted to the side of the SFP, a removable basket, and a drive system. It is used to lower a new fuel assembly from the operating floor level to the bottom of the SFP to allow the FHM to access it. A new fuel assembly is loaded into the NFE by the NFJC at the operating floor level before being lowered to the bottom of the pool. The NFE can also handle spent fuel assemblies for inspection or repairs and has a vertical travel limit to ensure adequate shielding of spent fuel assemblies.

9.1.4.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

The staff reviewed the FHE system for compliance with the requirements of GDC 2, with respect to its design for protection against the effects of earthquakes. Compliance with the requirements of GDC 2 is based on adherence to the guidance in RG 1.29, Regulatory Positions C.1 and C.2. This provision provides guidance on determining which SSCs shall be classified as seismic Category I.

FSAR, Section 9.1.4.3 states that the FHE is located inside the RXB, which protects the FHE from the effects of natural phenomena. The electrical power to the FHE is designed to interrupt in the event of a seismic event.

The NuScale US460 design classifies the FHE as a system that is not safety-related or risk significant. The FHM and the NFJC are designed to meet seismic Category I requirements, and the NFE is designed to meet seismic Category II requirements. In addition, the applicant stated that the FHM is also designed as a single-failure-proof crane in accordance with the guidelines of American Society of Mechanical Engineers (ASME) NOG-1, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)," for a Type 1 crane.

The NRC endorsement of ASME Std. NOG-1–2020 in RG 1.244 updates the guidance in NUREG-0554. NUREG-0554, Section 2.5, states, in part, the following about the seismic design of single-failure-proof cranes:

> [O]verhead cranes may be operating at the time that an earthquake occurs. Therefore, the cranes should be designed to retain control of and hold the load, and the bridge and trolley should be designed to remain in place on their respective runways with their wheels prevented from leaving the tracks during a seismic event.

FSAR, Section 9.1.4.2.2, states, in part, that "seismic restraints prevent the FHM bridge from overturning or coming off its rails during a seismic event." The staff finds that the FSAR description provides sufficient information on the design features specified in NOG-1-2020.

The FSAR also indicates that the NFJC and the NFE hoist systems are designed in accordance with the ASME NUM-1 design code, as a Type IA lifting device. The code defines a Type IA equipment as "[t]ype I crane, monorail, or hoist that includes single failure-proof features so that any credible failure of a single component will not result in the loss of capability to stop and hold the critical load."

FSAR, Section 9.1.4.2.3 indicates that the NFE components are able to withstand the seismic loading without coming loose and becoming missiles during a seismic event.

Based on the above evaluation, the staff finds that the FHE system is designed so it can withstand the effects of a seismic event without impacting the stored fuel, therefore the FHE system is consistent with the guidance in SRP Section 9.1.4 and therefore complies with the requirements of GDC 2.

9.1.4.4.2 GDC 5, "Sharing of Structures, Systems, and Components"

FSAR, Section 9.1.4.1, "Design Bases," states, in part, the following:

> The FHE is designed to support the periodic refueling of the reactor as well as movement of control rods and other radioactive components within the reactor core, refueling pool (RFP), and spent fuel pool (SFP).…The design of the FHE allows for the performance of fueling activities on one module without affecting the operation of the other modules including potential shutdown and cooldown.

The staff evaluated the applicant's description of the FHE and concludes that this equipment operates on a single NPM at a time and is located in a separate section of the UHS. The other NPMs continue to operate independently. The FHE does not interact with the other operating NPMs. The staff concludes that the use of the FHE system during the refueling of one module will not affect the capability of plant operators to maintain safe operation of the remaining operating modules, including potential shutdown and cooldown, if needed.

Based on the above evaluation, the staff finds that the FHE system design complies with the requirements of GDC 5.

9.1.4.4.3 GDC 61, "Fuel Storage and Handling and Radioactivity Control"

The staff reviewed the FHE system for compliance with the requirements of GDC 61 with respect to its design (1) for protection against releases of radioactivity to the environment because of fuel damage and (2) avoidance of excessive personnel radiation exposure. Compliance with the requirements of GDC 61 is based, in part, on conforming to the guidelines of ANSI/ANS 57.1.

9.1.4.4.3.1 Protection against Personnel Radiation Exposure

As described in FSAR, Section 9.1.4.2.3, “System Operation,” except for the handling of new fuel assemblies using the NFJC on the refueling floor, all other fuel transfer and storage operations using the NFE or the FHM are conducted underwater to provide adequate radiation shielding during refueling.

Acceptable shielding is maintained by designing and configuring the FHE system to comply with ANSI/ANS 57.1. As indicated in FSAR, Section 9.1.4.2.3, radiation shielding is provided by maintaining a minimum coverage of water over irradiated fuel. In FSAR, Section 14, Table 14.2-44, the applicant indicated that the interlock would maintain at least 3 m (10 feet) of water above the top of the fuel assembly when lifted to its maximum height with the pool level at the lower limit of the normal operating low water level. This minimum depth of water coverage is an exception allowed in RG 1.13 to the dose rate limit established in ANSI/ANS 57.1. Section 6.3.4.1.5 of ANSI/ANS 57.1, states that “fuel handling equipment shall be designed so that the operator will not be exposed to > 2.5 mrem/h [> 0.025 millisievert per hour (mSv/h)] from an irradiated fuel unit, control component, or both, elevated to the up position interlock with the pool at normal operating water level.”

The NFE is monitored and provided with a mechanical stop and radiation monitor interlock to limit vertical movement of spent fuel assemblies above exposure limits while operating in spent fuel mode. In new fuel mode, a load sensing interlock prevents motion in the raise direction when the basket is loaded. These design features effectively prevent movement of a spent fuel assembly by the NFE above the minimum water level.

Based on the above evaluation, the staff finds that the FHE system design complies with the requirements of GDC 61 for personnel radiation exposure.

9.1.4.4.3.2 Protection against Radioactivity Releases

For protection against damage by physical contacts, fuel assemblies are raised into a hollow mast during transport within the FHM operating area in the RFP and the SFP.

FSAR, Section 9.1.4.5, “Instrumentation and Control,” describes all relevant interlocks associated with the FHE system. The interlocks for the FHE system are as defined in ANSI/ANS 57.1, paragraph 6.3.1.1, and in Table 1 for the FHM, the NFJC, and the NFE.

These electrical interlocks (i.e., limit switches for control of FHM bridge, trolley, and hoist motions) are used to prevent damage to a fuel assembly and to monitor the fuel assembly load for imparted inertia loads greater than the allowable limits for which the fuel assemblies are designed.

Interlocks are provided to limit the motion of the FHM hoist, bridge, and trolley so that simultaneous vertical or horizontal motion is prevented while fuel assemblies are being moved or when a grapple or other tool is being moved in the proximity of the core such that fuel assemblies in the vessel could be damaged.

The FHM grapple design includes an interlock based on fuel assembly elevation that precludes the release of the fuel assembly in the reactor core if the elevation is above the maximum limits.

This design feature ensures that the grapple is properly engaged to the fuel assembly, the hoist does not lift until the grapple is fully closed and locked, and the grapple does not open with a suspended load.

In addition, to reduce the likelihood of a load drop event, the FHM is designed as a single failure-proof crane in accordance with the guidelines of ASME NOG-1 for a Type 1 crane.

FSAR, Section 9.1.4, indicates that the NFJC is designed as a single failure-proof crane in accordance with ASME NUM-1-2016 for a Type 1A crane. In RG 1.244, Revision 0 “Control of Heavy Loads at Nuclear Facilities,” issued December 2021, the NRC staff describes an approach that is acceptable to meet regulatory requirements for the control of heavy loads at nuclear facilities. In RG 1.244, Regulatory Position C.1.b.(2), lists the design criteria for determining the enhanced handling system reliability, such that the handling system can be acceptable for protecting safety functions during nuclear safety-critical lifts.

During the technical audit of the FSAR design documents (ML24264A049), the NRC staff reviewed the NFJC design and confirmed that the crane meets the design criteria identified in RG 1.244 Regulatory Position C.1 criteria (a) through (g). The staff finds this design ensures that a load drop event can be prevented over the SFP.

Based on the above evaluation, the staff finds that the FHE system design complies with GDC 61 requirements with respect to a radioactivity release because of fuel damage from mishandling or failure of the FHE system.

9.1.4.4.4 GDC 62, “Prevention of Criticality in Fuel Storage and Handling”

The FSAR does not include criticality safety analysis for fuel handling equipment. Section 9.1.4.1, “Design Bases” of the FSAR states the following:

> “Consistent with GDC 62, the FHE is designed such that it does not cause, or contribute to, a criticality accident. Protection from a criticality event is provided by designing the FHE to meet the requirements of ANSI/ANS standard, "Design Requirements for Light Water Reactor Fuel Handling Systems," ANSI/ANS 57.1.”

The FHE for US460 consists of three components: an FHM, an NFJC, and an NFE. The staff reviewed the FHE system and determined that criticality is not plausible for the FHE because the FHE will handle only one fuel assembly at a time. On this basis, the staff concludes that criticality safety analysis is not warranted due to the handling design noted above.

9.1.4.5 Initial Test Program

The staff’s evaluation of the ITP is discussed in Section 14.2 of this report.

9.1.4.6 Technical Specifications

No GTS requirements are associated with the FHE system.

9.1.4.7 Combined License Information Items

Table 9.1.4-1 lists the COL information item and descriptions related to the FHE system, from FSAR, Table 1.8-1.

Table 9.1.4-1 NuScale COL Information Items for Section 9.1.4

COL Item No.	Description	FSAR Section
9.1-3	An applicant that references the NuScale Power Plant US460 standard design will provide the periodic testing plan for fuel handling equipment.	9.1

The staff review the proposed COL Item 9.1-3 and determined that developing a periodic testing plan for fuel handling equipment is consistent with the design standards of the equipment and consistent with the requirements of 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses. Therefore, the staff finds the proposed COL information item acceptable.

9.1.4.8 Conclusion

The FHE system includes all components and equipment for moving fuel and other related light loads between the receiving area, the fuel storage areas, and the reactor vessel.

Based on the above evaluation, the staff finds that the FHE system design complies with requirements of GDC 2, 5, 61, and 62.

9.1.5 Overhead Heavy Load Handling Systems

9.1.5.1 Introduction

The Overhead Heavy Load Handling Systems (OHLHS) consists of all equipment for moving all heavy loads (i.e., loads weighing more than one fuel assembly and control rod assembly) at the plant site. The review focuses on critical load handling, during which inadvertent operations or equipment malfunctions, separately or in combination, could cause a release of radioactivity, criticality accident, or inability to cool the fuel within the reactor vessel or SFP, or could prevent safe shutdown of the reactor.

9.1.5.2 Summary of Application

The applicant provided a detailed description of the OHLHS in FSAR, Section 9.1.5.

For the NuScale US460 design, a heavy load is defined as any load greater than the combined weight of a single fuel assembly and control rod assembly, which is approximately 410 kilograms (kg) (900 pounds (lbs)). The primary piece of equipment used in the OHLHS is the reactor building crane (RBC). Other OHLHS equipment includes various other hoists, jib cranes, load handling devices used by the RBC, and additional tools that are used to hold, inspect, assemble, and disassemble the NPM for refueling (e.g., containment vessel flange tool (CFT), reactor vessel flange tool (RFT), and the module inspection rack). The OHLHS also includes equipment accessories (e.g., slings and hooks), instrumentation, physical stops, electrical interlocks, and associated programmatic controls. FSAR, Figure 9.1.5-1, "Reactor Building Crane Safe Load Paths" shows the safe load path for handling the NPMs. FSAR, Figure 9.1.5-2, shows the design configuration of the RBC, and Figure 9.1.5-3, shows design details of the reactor building crane lower block assembly connection to the top support structure. Finally, FSAR Table 9.1.5-1, gives the design data for major components of the heavy load handling system.

The applicant classified the RBC as “nonsafety-related, risk-significant,” and the remaining hoists and equipment as “nonsafety-related and not risk-significant.”

The applicant’s Table 9.1.5-1, “Heavy Load Handling Equipment Design Data,” shows the tabulated rated capacity of the OHLHS in “tons,” and the applicable ASME design codes with the types of hoisting systems (Equipment) and the seismic categories. The applicant also classified the RBC as a seismic Category I structure and the remaining OHLHS equipment as seismic Category II structures as identified in Table 9.1.5-1.

The applicant defined critical load handling as the handling of a heavy load where inadvertent operations or equipment malfunctions, separately or in combination, could cause a release of radioactivity, a criticality accident, the inability to cool fuel within the reactor vessel or SFP, or prevent safe shutdown of the reactor.

The applicant provided Figure 9.1.5-2, “Reactor Building Crane,” showing the plan, elevation, and isometric views of the RBC. The applicant also included Figure 9.1.5-3, “Reactor Building Crane Lower Block Assembly Connection to the Top Support Structure,” showing the elevation view of the lower block assembly (LBA) located at the bottom of the main hoist of the RBC to lift and move the NPM from the operating bay to the refueling bay/dry dock area and move it back into the operating bay.

The applicant also included COL Item 9.1-4 requiring, “An applicant that references the NuScale Power Plant US460 standard design will describe the process for handling and receipt of critical loads including NPMs,” and COL Item 9.1-5 requiring, “An applicant that references the NuScale Power Plant US460 standard design will provide a description of the program governing heavy loads handling. The program should address

- operating and maintenance procedures.
- inspection and test plans.
- personnel qualification and operator training.
- detailed description of the safe load paths for movement of heavy loads.”

The OHLHS is protected from the effects of external missiles by its location in the RXB.

The applicant listed the following three exceptions for the RBC main hoist and lower block assembly in Note 1 of Table 9.1.5-1:

- 4332: Use of 1.11 for the design factor in plate buckling for extreme environmental loads.
- 4334: Alternate methodology for spacing of transverse stiffeners.
- 4461: Runway and bridge rails conform to DIN 536-1.

Section 9.1.5.4 of this SER discusses exceptions 4332, 4334 and 4461 further.

During a regulatory audit (ML24264A049), the applicant clarified the use of the ASME NUM-1-2016 standard in the US460 design. The applicant referred to Regulatory Position C.1.b(2) in RG 1.244, listing additional criteria necessary for review of ASME NML-1-2019 cranes that were not previously approved by the NRC. Therefore, to justify the applicability of NUM-1-2016 Type IA crane requirements for the US460 design, the applicant assessed criteria (a) through (g) identified in Regulatory Position C.1.b(2) of RG 1.244 and determined NUM-1-2016 to be

applicable for use in accordance with the assessment of criteria (a) to (g) and, therefore, is acceptable.
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ITAAC: FSAR Part 8, Table 3.10-2, "Overhead Heavy Load Handling System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information," specifies the ITAAC for the RBC. Section 14.3 of this SER evaluates these ITAAC.

9.1.5.3 Regulatory Basis

SRP Section 9.1.5, Revision 1, "Overhead Heavy Load Handling Systems," issued March 2007, gives the relevant regulatory requirements for this area of review and the associated acceptance criteria, as summarized below, as well as the review interfaces with other SRP sections:

- GDC 1, "Quality standards and records," as it relates to the design, fabrication, and testing of SSCs important to safety to maintain quality standards

- GDC 2, as it relates to the ability of structures, equipment, and mechanisms to withstand the effects of earthquakes

- GDC 4, as it relates to the protection of safety-related equipment from the effects of internally generated missiles (i.e., dropped loads)

- GDC 5, as it relates to the sharing of equipment and components important to safety

RG 1.244, Revision 0, describes an approach that is acceptable to the staff to meet regulatory requirements for the control of heavy loads at nuclear facilities. Specifically, these requirements are for the licensees to provide appropriate protection against equipment failure that could result in a heavy load drop. RG 1.244 endorses the following ASME codes and standards, with clarifications:

- ASME Std. NML-1–2019, “Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities,” dated June 28, 2019. The NRC staff endorsement of ASME Std. NML-1–2019 in this RG updates the guidance in NUREG-0612, “Control of Heavy Loads,” issued March 1984.

- ASME Std. NOG-1–2020, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” dated December 4, 2020. The NRC staff endorsement of ASME Std. NOG-1–2020 in this RG updates the guidance in NUREG-0554.

- ASME Std. BTH-1–2017, “Design of Below-the-Hook Lifting Devices,” dated March 15, 2017, Chapters 1–3.

However, as described above, to justify the applicability of NUM-1-2016 Type IA crane requirements for the US460 design, the applicant assessed criteria (a) through (g) in Regulatory Position C.1.b(2) of RG 1.244.

9.1.5.4 Technical Evaluation

The OHLHS consists of the components and equipment necessary for the safe handling of heavy loads such as one NPM during refueling or a fully loaded spent fuel transfer cask during normal plant conditions of operating NPMs. SRP Section 9.1.5 defines heavy loads as loads weighing more than the weight of one fuel assembly plus its handling device. For the NuScale US460 design, FSAR, Section 9.1.5, defines heavy loads as “loads whose weight is greater than the combined weight of a single fuel assembly and control rod assembly.” The applicant elected to use 410 kg (900 lbs) as the threshold value, as stated in FSAR, Section 9.1.5. The staff finds the proposed definition acceptable because it is consistent with the use of the FHM in the NuScale US460 design to handle this combined load during a refueling operation for one NPM.

The OHLHS consists of equipment and components used for critical load handling. FSAR, Section 9.1.5, defines a critical load handling evolution as the handling of a heavy load in which inadvertent operations or equipment malfunctions, separately or in combination, could cause one or more of the following:

- a release of radioactivity

- a criticality accident

- the inability to cool fuel within the reactor vessel or SFP

- prevent safe shutdown of the reactor

FSAR, Section 9.1.5.2, "System Description," describes various OHLHS components, as discussed below. Major components of the OHLHS include the traveling jib crane, the articulating traveling jib crane, the dry dock jib crane, the module access platform jib crane, and the auxiliary wet hoist.

In accordance with guidance in SRP Section 9.1.5, the application should conform to general programmatic guidelines for a highly reliable process for handling critical loads at nuclear power plants. As described in FSAR, Section 9.1.5, COL Item 9.1-4 should be addressed by the COL applicant to describe the process for handling and receipt of critical loads including NPMs. These programmatic elements will be addressed by a COL applicant and reviewed by the staff at the COL application stage.

In accordance with guidance in SRP Section 9.1.5, the application should conform to general programmatic guidelines for operation, testing, maintenance, inspection and safe load paths for controlling of heavy loads at nuclear power plants. As described in FSAR, Section 9.1.5.3, COL Item 9.1-5 was included to identify that the COL applicant should address development the heavy load handling program. These programmatic elements will be addressed by a COL applicant and reviewed by the staff at the COL application stage.

NuScale provided the types of hoisting systems and the applicable ASME design codes in FSAR Table 9.1.5-1. The types of hoists for the RBC main hoist, LBA, and sister hook and the RBC auxiliary hoists are identified as Type I and Type IA, respectively, and the standard for the design is identified as ASME NOG-1-2020.

Sections 9.1.5.2.2 and 9.1.5.2.3 of the FSAR also provide the following design criteria and operational characteristics of the RBC for lifting and moving equipment within the RXB to support normal operations, maintenance, and receipt of new equipment and to assist in refueling operations:

- The runway rails of the RBC are anchored more than 140 millimeters (mm) (5.5 inches) from the edge to the RXB across the length of the reactor pool, refueling pool, and dry dock.

- The RBC trolley lifted load is supported by the bridge structure and travels on the bridge rails across the width of the RXB pool.

- The RBC rope reeving system is designed to transfer the load to the remaining ropes without excessive shock in case of a failed rope.

- To monitor for failure of a rope, a load-weighing assembly monitors the tension on the rope for slack rope when a load is lowered, for high loads due to too heavy of a load or hang up, and for a broken rope. Therefore, the design of the assembly ensures a structural failure does not result in a dropped load.

- The LBA design includes clevises with lifting lugs on the top support structure (TSS) of the NPM. The pins' engagement to the TSS lifting lugs is secured with actuators and is confirmed by travel limit switches and visual indications.

- As shown in FSAR Figure 9.1.5-3, four large-diameter pins connect the LBA and TSS.

- The low-capacity lifting for equipment in the RXB is provided by two auxiliary hoists mounted on the RBC. A load-weighing assembly also monitors for slack rope, high loads, and broken ropes.

- The RBC is designed to withstand the RXB environmental conditions and to operate during all modes of plant operations.

- The crane's operator specifies and schedules the tasks using the RBC to transfer an NPM from its installed operating position in the reactor pool to the refueling pool and back. The operator also determines the safe travel paths and enters the attributes into the RBC control system.

- The site operating procedure and associated drawings shall define the safe load paths to mitigate the probability of a heavy load drop that could result in damage to safe-shutdown equipment or unacceptable radiation exposures.

- An alignment before the engagement between the RBC and NPM is performed with the assistance of a position control system.

- The travel path is chosen to accommodate the load on the RBC hoist. Repeatability, proper load path, and proper locations are ensured by semiautomatic crane operation.

Section 9.1.5.5 of the FSAR describes the instrumentation and control systems for the RBC and articulating traveling jib crane (ATJC) for positioning, weighing capacities, temperature, seismicity by limit switches, and interlocks. These have, in part, the following purposes:

- Ensure that travel occurs within intended travel paths.

- Prevent lifting more than the rated load.

- Monitor for high temperature inside the crane drive power panels.

- Use of a programable logic controller for operation and monitoring.

- Use software interlocks to prevent collisions with other SSCs.

- Ensure direct mechanical control system drives are available in the event of power failure.

The RBC and its equipment design conform to the requirements for ASME NOG-1-2020, Type I, and NUM-1-2016, Type IA, cranes that (1) provide redundancy by remaining in place and supporting critical loads during and after a seismic event (SSE), (2) have single-failure-proof features so that any failure of a single component would not result in loss of capability to hold the critical load, and (3) have instrumentation and control systems to verify specific safe operating requirements. The use of these standards also provides assurance that a postulated load drop analysis to assess radiological consequences is not required. Therefore, the staff finds that the safe movement of critical loads by the RBC and its equipment complies with the design requirements in ASME NOG-1-2020, which was also endorsed by RG 1.244, and NUM-1-2016, which is determined to be applicable for use in accordance with the assessment of criteria (a) to (g) in Regulatory Position C.1.b(2) of RG 1.244, and, therefore, is acceptable.

The remaining elements of the OHLHS listed in FSAR Table 9.1.5-1 are the traveling jib crane (TJC), ATJC, dry dock jib crane, and auxiliary wet hoist (AWH), identified as Type IA, and the module access platform (MAP) jib crane identified as Type IA.

- The TJC is a fixed boom, wall-mounted crane that traverses the wall of the refueling pool along a rail system between the operating bays and the dry dock, providing heavy load material handling capability to the dry dock area.

- The ATJC is a wall-mounted jib crane that traverses the wall of the refueling pool along a rail system between the operating bays and the refueling pool, providing heavy load material handling capability to the reactor flange tool (RFT) and the containment flange tool (CFT) during refueling. The applicant provided a list of the following interlock controls in FSAR Section 9.1.5.5:

  – hoist interlock

  – trolley or boom interlock

  – CFT or RFT keep-out zone interlock

  – FHM interlock

- The drydock jib crane is mounted to the top of the wall located between the dry dock and the refueling pool, providing heavy load material handling capability to the dry dock area.

- The AWH is an intermediate hoist that attaches to either the RBC main hoist via the sister hook or to one of the RBC auxiliary hoists. It is used for operations that require the hook to be lowered into the reactor building pool water.

- The MAP jib crane is a movable personnel support structure employed during disassembly and assembly of the NPM in the operating bay for refueling.

As discussed above, the applicant listed the following exceptions for the RBC in note 1 of Table 9.1.5-1:

- 4332: Use of 1.11 for the design factor in plate buckling for extreme environmental loads.
- 4334: Alternate methodology for spacing of transverse stiffeners.
- 4461: Runway and bridge rails conform to DIN 536-1.

The staff reviewed exceptions 4332 and 4334 and found that exceptions 4332 and 4334 are in process to be incorporated through the ASME Ballots 22-1890 and 20-2379 in the 2025 Edition of ASME NOG: (1) Design Factors in Plate Buckling (DFB) of 1.11 for the load combination of "extreme environment," in paragraph 4332, and (2) provide clear guidance on the applicability of the requirements for transverse stiffeners in paragraph 4334, respectively. Further, the staff concludes that the load condition of extreme environment described in paragraph 1150 of ASME NOG-1-2020 probabilities of occurrence equal or more than 10^{-7} per calendar year at the plant of crane installation. Based on the review discussed above, the staff concludes that both exceptions, 4332 and 4334, are acceptable because they are providing advances and clarification to the ASME NOG-1-2020 standard. In addition, the staff reviewed exception 4461 and finds the use of DIN 536-1, "Crane rails; hot rolled flat bottom crane rails (type A); dimensions, section parameters and steel girders," acceptable for the reactor building crane main hoist and lower block assembly. DIN 536-1 defines standardized rail profiles with a well-established record of reliable performance in heavy-duty industrial applications. The rail geometry specified in the standard provides adequate load-carrying capacity, dimensional precision, and compatibility with crane wheel and track designs. When used in conjunction with supporting structures designed in accordance with ASME NOG-1-2020 standard, the application of DIN 536-1 rails supports safe and reliable crane operation under all design-basis loading conditions, including seismic events. NuScale identified that RG 1.244 did not directly endorse the design standard ASME NUM-1-2016 for the OHLHS. Therefore, to justify the applicability of NUM-1-2016 Type IA crane requirements for the US460 design, the applicant assessed criteria (a) through (g) in Regulatory Position C.1.b(2) of RG 1.244 as concluded above.

The ASME NUM-1-2016 Type IA crane requirements that provide redundancy by remaining in place and supporting critical loads during and after a seismic event (SSE) and with single-failure-proof features ensure that any failure of a single component would not result in loss of capability to hold the critical load. The use of this standard also provides assurance that a postulated load drop analysis to assess radiological consequences is not required. Therefore, for the reasons stated above, the staff concludes that the design requirements of ASME NUM-1-2016 ensure safe movement of critical lifts by the TJC, ATJC, drydock jib crane, MAP jib crane, and AWH and, therefore, are acceptable and also comply indirectly with RG 1.244.

For the reasons stated above, the staff finds that the FSAR contains sufficient design information for these cranes.

9.1.5.4.1 GDC 1, "Quality Standards and Records"

The staff reviewed the OHLHS for compliance with GDC 1, which requires that nuclear power plant systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Compliance with the requirements of GDC 1 is based on conformance to the design requirements in (1) ASME NOG-1-2020 for Type I cranes and (2) ASME NUM-1-2016 based on meeting criteria (a) through (g) of RG 1.244, Regulatory Position C.1.b(2), for Type IA cranes. These standards for Types I and IA cranes include provisions for the design, fabrication, installation, inspection, testing, and maintenance of cranes. In the NuScale US460 SDAA design, the primary OHLHS will handle critical loads in the vicinity of or involving spent fuel or safety-related components with single-failure-proof lifting features.

Based on the above evaluation, the staff finds that the OHLHS design is based on the design requirement of ASME NOG-1-2020 for Type I cranes, and ASME NUM-1-2016 for Type IA cranes, and therefore complies with the requirements of GDC 1.

9.1.5.4.2 GDC 2, "Design Basis for Protection against Natural Phenomena"

The staff reviewed the OHLHS for compliance with the requirements of GDC 2, with respect to its design for protection against the effects of earthquakes. Compliance with the requirements of GDC 2 is based on conforming to Regulatory Position C.2 of RG 1.29 and the guidelines in ASME NOG-1.

Regulatory Position C.2 of RG 1.29 states that SSCs that are not required to continue to function after a seismic event, but whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable safety level, should be designed and constructed so that the SSE would not cause such failure.

Compliance with the requirements of GDC 2 is based on conformance to the design requirement in ASME NOG-1-2020 for Type I cranes and ASME NUM-1-2016 based on meeting criteria (a) through (g) of RG 1.244, Regulatory Position C.1.b(2), for Type IA cranes. Types I and IA cranes that meet these standards provide redundancy by remaining in place and supporting critical loads during and after a seismic event (SSE) and have single-failure-proof features to ensure that any failure of a single component would not result in loss of capability to hold the critical load. The use of these standards also provide reasonable assurance that a postulated load drop analysis to assess radiological consequences is not required. The OHLHS is located in the RXB and, therefore, is protected from the other elements of external natural phenomena (e.g., tornadoes, hurricanes, floods, tsunami, and seiches).

Based on the above evaluation, the staff finds that the OHLHS design complies with the requirements of GDC 2.

9.1.5.4.3 GDC 4, "Environmental and Dynamic Effects Design Bases"

The staff reviewed the OHLHS for compliance with the requirements of GDC 4 with respect to protection of fuel and safety-related equipment from the effects of internally generated missiles (dropped loads). A dropped heavy load in a critical area could cause a release of radioactive materials, criticality accident, or inability to cool fuel within the reactor vessel or SFP or could prevent safe shutdown of the reactor.

Compliance with the requirements of GDC 4 is based on conformance to the design requirement in ASME NOG-1-2020 for Type I cranes and ASME NUM-1-2016 based on meeting criteria (a) through (g) in RG 1.244, Regulatory Position C.1.b(2), for Type IA cranes. These standards for Types I and IA cranes include provisions for the design, installation, inspection, testing, and maintenance of cranes. Types I and IA cranes that meet these standards provide redundancy by remaining in place and supporting critical loads during and after a seismic event (SSE) and have single-failure-proof features to ensure that any failure of a single component would not result in loss of capability to hold the critical load. The use of these standards also provide reasonable assurance that a postulated load drop analysis to assess radiological consequences is not required. As the applicant stated in FSAR Section 9.1.5.1, "the OHLHS is protected from the effects of external missile hazards by being located inside the RXB."

In addition, to reduce the probability and mitigate the consequences of an accidental load drop, SRP Section 9.1.5.III.3 requires a description of a heavy load handling program that is consistent with the guidelines of NUREG-0612, Section 5.1.1. In RG 1.244 the NRC staff endorsed ASME NML-1–2019 which updates the guidance in NUREG-0612. NML-1 includes guidance on general programmatic controls for the design, operation, testing, maintenance, and inspection of heavy load handling systems, and establishing safe load paths for critical load handling.

FSAR, Section 9.1.5.2.3 states, in part, the following:

> The RBC transfers an NPM from its installed operating position in the reactor pool to the refueling pool and back. Travel paths are determined, and attributes are entered into the RBC control system. Each task is specified and scheduled by the crane operator.
>
> Safe load paths are defined in operating procedures and equipment drawings [as defined by COL Item 9.1-5]. This restriction reduces the probability of a heavy load drop

that could result in safe shutdown equipment damage or result in a release of radioactive material that could cause unacceptable radiation exposures.

> The position control system assists in aligning the RBC with the NPM for engagement before performing lifting operations. The RBC control system is capable of load-dependent travel restrictions. The travel path is chosen to accommodate this information. Repeatability, proper load path, and proper locations are ensured by semi-automatic crane operation.

The staff's review of FSAR Figure 9.1.5-1 confirmed the defined safe load path for the RBC to move the NPM within the reactor pool and the RFP. In FSAR, Section 9.1.5.5, "Instrumentation and Control," the applicant indicated that the RBC utilizes a programmable logic controller (PLC) for control, monitor, and operations of the crane. The control system utilizes multiple position feedback devices and software interlocks to prevent collisions with other SSCs. The RBC can be operated with automated motions (which include hold points and way points indicated by the load path) or in manual control (with reduced speeds). The staff evaluation of the PLC is discussed in Section 19.1.4 of this report.

Therefore, the staff concludes the above description of the interlocks and controls to be consistent with the guidelines of NML-1-2019 with respect to defining safe load paths.

The RBC and its main hoist, two auxiliary hoists, traveling jib crane hoist, ATJC, DDJC, MAP jib crane, AWH, and attached handling tools are designed as single-failure-proof components in order to minimize the likelihood of a load drop event. The staff finds that these design features are consistent with the purpose described in of RG 1.13, Regulatory Position C.5, and, therefore, are acceptable.

FSAR, Section 9.1.5.4, states that the RBC is inspected and tested in accordance with ASME NOG-1. In-process inspection and testing of the AWH, ATJC, MAP jib crane, drydock jib crane, and traveling jib crane are performed in accordance with ASME NUM-1.

In COL Item 9.1-5, NuScale stated the following:

> An applicant that references the NuScale Power Plant US460 standard design will provide a description of the program governing heavy loads handling. The program should address:
>
> • operating and maintenance procedures.
>
> • inspection and test plans.
>
> • personnel qualification and operator training.
>
> • detailed description of the safe load paths for movement of heavy loads.

SRP Section 9.1.5.III.3 states that the licensee shall describe a heavy load handling program consistent with the NRC general programmatic guidelines for the design, operation, testing, maintenance, and inspection of heavy handling systems. The staff finds that COL Item 9.1-5 includes key elements that the COL applicant will address in its heavy load handling program, and the staff will review specific conformance to the applicable requirements during the COL review stage.

Based on the above evaluation, the staff finds that the OHLHS design complies with the requirements of GDC 4.

9.1.5.4.4 GDC 5, "Sharing of Structures, Systems, and Components"

The staff noted that the six NPMs share the OHLHS to support refueling one NPM at a time. The OHLHS design allows for refueling activities on one module with minimum impact on the operation of the other modules, including potential shutdown and cooldown. Based on this and the information reviewed in FSAR 9.1.5.2.3, "System Operation," the staff finds the applicant's assessment of this interaction complete and acceptable.

Based on the above evaluation, the staff finds that the OHLHS design complies with the requirements of GDC 5.

9.1.5.5 ITAAC

SDAA, Part 8 Section 3 Table 3.10-1: "Overhead Heavy Load Handling System Inspections, Tests, Analyses, and Acceptance Criteria," Table 3.10-2: "Overhead Heavy Load Handling System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information," and Table 3.10-3: "Overhead Heavy Load Handling System Equipment," described the ITAAC for the OHLHS. These ITAAC are evaluated in Section 14.3 of this report.

9.1.5.6 Initial Test Program

FSAR, Table 14.2-45: "Test # 45 Reactor Building Cranes," describes the crane performance testing that will demonstrate proper operation of all control circuits and associated interlocks and proper transport of an NPM from and to its installed position in a reactor bay. The staff evaluates the ITP in Section 14.2 of this report.

9.1.5.7 Technical Specifications

No GTS requirements are associated with the OHLHS.

9.1.5.8 Combined License Information Items

Table 9.1.5-1 lists COL information item numbers and descriptions related to the OHLHS, from FSAR, Table 1.8-1.

Table 9.1.5-1 NuScale COL Information Items for Section 9.1.5

COL Item No.	Description	FSAR Section
9.1-4	An applicant that references the NuScale Power Plant US460 standard design will describe the process for handling and receipt of critical loads including NPMs	9.1
9.1-5	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the program governing heavy loads handling. The program should address • operating and maintenance procedures. • inspection and test plans. • personnel qualification and operator training. • detailed description of the safe load paths for movement of heavy loads.	9.1

The NuScale SDA proposed COL Item 9.1-4, and 9.1.5 which instructs the COL applicant to provide a description of the process of handling critical loads and the program governing heavy load handling. As discussed above in Section 9.1.5.4.3, ASME NML-1-2019 provide the guidelines for developing a heavy loads handling program.

The staff finds the COL information items acceptable because it is appropriate for COL applicants and holders to provide process and a program to address the handling of critical loads and heavy loads.

9.1.5.9 Conclusion

The OHLHS includes components and equipment for the handling of heavy loads at the plant site. Based on the above evaluation, the staff finds that the OHLHS design complies with the requirements of GDC 1, 2, 4, and 5.

9.2 <u>Water Systems</u>

9.2.1 Station Service Water System

FSAR, Section 9.2.1, "Station Service Water System," states the following:

> This section is relevant to light water reactor (LWR) active designs that incorporate a service water system serving as the final heat transfer loop between various heat sources and the plant ultimate heat sink (UHS). The NuScale Power Plant design does not have a service water system.
>
> A typical LWR service water system provides essential cooling to safety-related equipment and can also cool nonsafety-related auxiliary components used for normal plant operation. The NuScale Power Plant US460 passive design does not rely on active systems such as a service water system to provide cooling to essential equipment. The NuScale Power Modules are partially immersed in the reactor pool portion of the plant UHS. This design configuration ensures passive heat transfer from essential systems and components directly to the UHS, with no intermediate heat transfer loop such as that provided by a typical LWR essential service water system.

The staff reviewed the NuScale system design and confirmed this statement. Therefore, no further review will be needed for the station service water system. However, the staff makes the following observations about the NuScale design:

- The UHS, reviewed in Section 9.2.5 of this SER, performs the safety-related function of decay heat removal, which the service water system usually performs in LWR active designs
- The SCWS, reviewed in Section 9.2.7 of this SER, performs the heat removal function for the systems that are not safety-related, which the service water system usually performs in LWR designs

9.2.2 Reactor Component Cooling Water System

9.2.2.1 Introduction

The reactor component cooling water system (RCCWS) is a closed-loop cooling system that is not safety-related and provides cooling for the following SSCs:

- control rod drive mechanism (CRDM) electromagnetic coils housing
- chemical and volume control system (CVCS) nonregenerative heat exchangers
- containment evacuation system (CES) condensers and vacuum pumps
- process sampling system (PSS) coolers and analyzer cooler temperature control units

The CVCS, CES, and PSS components cooled by the RCCWS are located in the RXB. The CRDM electromagnetic coils, which the RCCWS also cools, are located inside containment and outside of the reactor vessel. The RCCWS transfers the heat from these systems to the SCWS and then to the environment through the SCWS cooling tower.

9.2.2.2 Summary of Application

FSAR Section 9.2.2, "Reactor Component Cooling Water System," contains a general description of the RCCWS, including the RCCWS design bases, system descriptions and safety evaluation. Information regarding RCCWS initial testing is provided in FSAR, Section 14.2 (Test # 04).

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): There are no proposed ITAAC related to the RCCWS.

9.2.2.3 Regulatory Basis

In general, SRP Section 9.2.2, Revision 4, "Reactor Auxiliary Cooling Water System," issued March 2007, gives the relevant regulatory requirements for this area of review and the associated acceptance criteria. Because the RCCWS is not a safety-related system, and the cooling it provides is not required for safety-related or risk-significant components to perform their safety function, the cooling water system requirements of GDC 44, "Cooling Water," GDC 45, "Inspection of Cooling Water System," and GDC 46, "Testing of Cooling Water System," are not applicable, as discussed in the section 9.2.2.4.1 of this SER. Accordingly, only the following regulatory requirements noted in SRP Section 9.2.2 are relevant to the US460 design:

- GDC 2, as it relates to the capability of structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without a loss of safety-related functions.

- GDC 4, as it relates to the capability of the system and the structure housing the system to withstand the effects of missiles inside and outside of containment, the effects of pipe whip and jets, environmental conditions from high- and moderate-energy line breaks, and the dynamic effects of flow instabilities and loads (e.g., water hammer) during normal plant operation and upset or accident conditions.

- GDC 5, insofar as it requires that SSCs important to safety not be shared among power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions.

- The following additional regulatory requirements also apply to the RCCWS:

- GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the RCCWS design for the control of releases of radioactive materials to the environment.

- GDC 64, "Monitoring Radioactivity Releases," as it relates to the RCCWS design for monitoring releases of radioactive materials to the environment during normal operation, including anticipated operational occurrences.

- 10 CFR 20.1406, "Minimization of Contamination," as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive waste.

9.2.2.4 Technical Evaluation

The staff reviewed the RCCWS in accordance with the review procedure in SRP Section 9.2.2. The following sections give the results of the staff's review.

9.2.2.4.1 System Design Considerations

Design Basis

The RCCWS is designed to remove the heat load from the CRDMs, the CVCS nonregenerative heat exchangers, the CES condensers and vacuum pumps, and the PSS coolers and temperature control units during normal plant operation. NuScale designated a boundary for the RCCWS that ends outside of the containment and does not include the containment isolation valves for RCCWS cooling water to and from containment. The containment isolation valves associated with the RCCWS provide a safety-related function but are evaluated in Section 6.2.4 of this report. The RCCWS is not required for the orderly shutdown of an NPM or the ability to maintain the NPM shutdown. It provides no safety-related function, is not credited for the mitigation of DBEs, and has no safe-shutdown functions. The applicant considered GDC 2, 4, 5, 60, and 64, and 10 CFR 20.1406 in designing the RCCWS.

The RCCWS provides cooling water to the CRDM electromagnetic coils and thus interfaces directly with NPMs via system piping routed inside containment used to support CRDM cooling. The RCCWS also provides cooling water for the CVCS nonregenerative heat exchangers, the CES condensers and vacuum pumps, and the Process Sampling System (PSS) analyzer coolers and temperature control units during normal plant operation, all of which support NPM operation and are located inside the RXB but outside of containment.

The staff found that the loss of CRDM cooling does not affect the safety function of the CRDM, which is to insert the control rods upon a reactor trip. However, as identified in SRP Section 4.6, "Functional Design of Control Rod Drive System," in order to be in compliance with GDC 26, "Reactivity control system redundancy and capability," two independent reactivity control systems of different design principles must be provided and be capable of reliably controlling reactivity changes under conditions of normal operation, including anticipated operational occurrences (AOOs), to provide assurance that specified acceptable fuel design limits (SAFDLs) are not exceeded. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions. Therefore, the RCCW must be able to provide sufficient cooling to support the CRDM operation during normal operation and AOOs. In FSAR Section 4.6.1, NuScale indicated that the RCCWS will maintain the CRDM winding temperature below the design maximum of 392 degrees.

To verify that the RCCWS design had sufficient capacity and heat removal capability to provide adequate cooling to the loads it services, the staff audited (ML24264A049) the NuScale analyses to demonstrate that the RCCWS will maintain winding temperature below the design maximum and confirmed the analysis supported the statement made in the FSAR. Based on its review of the information provided about the design and operation of the RCCWS, the staff confirmed that (1) although the RCCWS is credited for providing water to CRDMs, the failure of the cooling function will not prevent the CRDMs from performing their safety-related function; (2) RCCWS operation is not required to support NPM cooling during shutdown or post-accident conditions; and (3) during normal operation, the RCCWS will provide sufficient cooling to maintain the CRDM windings below their design maximum temperature. Therefore, the staff finds that GDC 44, 45, and 46 included in SRP Section 9.2.2.II are not applicable to the RCCWS because the system design and operation is such that the RCCWS is not a safety-related system.

9.2.2.4.2 GDC 2, "Design Bases for Protection against Natural Phenomena"

The staff reviewed the RCCWS for compliance with the requirements of GDC 2 with respect to its design for protection against the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. Compliance with the requirements of GDC 2 is based on the RCCWS being designed to withstand the effects of natural environmental phenomena without losing the ability to perform its safety function and on meeting the guidance of RG 1.29, Regulatory Positions C.1 for the safety-related portions of the system and Regulatory Position C.2 for the portions of the system that are not safety-related.

The RCCWS is located in the seismic Category 1 portion of RXB, which is designed to protect SSCs from extreme winds and missiles that may result from natural phenomena such as earthquakes, tornadoes, and hurricanes. FSAR Section 3.5.2, describes the RXB as being designed in accordance with RG 1.13, Revision 2; RG 1.117, Revision 2, "Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants," issued July 2016; and RG 1.221, Revision 0, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," issued October 2011. The RXB also protects the RCCWS from the effects of external flooding as described in FSAR, Section 3.4.2, "Flood Protection from External Sources."

FSAR Table 9.2.2-1, "Classification of Structure, Systems, and Components," gives the location, safety classification, and seismic category for the RCCWS. The staff noted that some SSCs used by the RCCWS that would typically be identified as part of the RCCWS were assigned to other systems. The RCCWS lines that supply water to the CRDMs are functionally part of the RCCWS but are included as control rod drive system cooling lines in the control rod drive system. The RCCWS supply and return lines and the associated containment isolation valves are seismic Category 1 as discussed in FSAR Section 3.1.5.7 and 3.1.5.8.

The staff reviewed the information on the RCCWS in the FSAR and verified that RCCWS components that perform important-to-safety functions, and RCCWS SSCs whose failure could affect SSCs important to safety, were designed in accordance with the guidance in RG 1.29. According to FSAR, Table 9.2.2-1, the piping and components that make up the RCCWS are all designed to Seismic Classification III, which the staff finds to be an appropriate classification for the SSCs located outside the containment and away from other SSCs important to safety. In the containment system, the RCCWS piping from the containment isolation valve to the disconnect flange outside containment is Seismic Classification I, which the staff finds conforms with RG 1.29, Regulatory Position C.1.

Based on the above discussion, the staff concludes that the RCCWS as described in the FSAR complies with the requirements of GDC 2.

9.2.2.4.3 GDC 4, "Environmental and Dynamic Effects Design Bases"

The staff reviewed the RCCWS for compliance with the requirements of GDC 4 with respect to the capability of the system and the structures housing the system to withstand the effects of pipe breaks, including the effect of pipe whip, jet impingement, and the environmental conditions resulting from high- and moderate-energy line breaks, as well as the effect of flow instabilities and attendant loads (water hammer). Compliance with the requirements of GDC 4 is based on the identification of the essential portions of the system as protected from dynamic effects, including internal and external missiles, pipe whip, and jets, and the ability of the system to continue to perform its safety function in the environmental conditions that may result from high- and moderate-energy line breaks and the resulting discharged fluid.

The RCCWS containment isolation valves perform the safety-related function of containment isolation. The containment isolation valves associated with the RCCWS provide a safety-related function but are part of the containment isolation system and are evaluated in Section 6.2.4 of this report. These valves and their associated sensors may be subject to harsh environmental conditions and therefore must comply with the requirements of 10 CFR 50.49, "Environmental qualification of electric equipment important-to-safety for nuclear power plants." FSAR Table 3.11-1, "List of Environmentally Qualified Equipment Located in Harsh Environments," indicates that these valves will be located in Equipment Qualification Zone RXBP-1 and qualified for a harsh environment. As the RCCWS valves will be qualified for a harsh environment, they will be designed to perform their safety-related isolation function while subject to the harsh environment. Section 3.11 of this report addresses compliance with 10 CFR 50.49 for the qualification of equipment located in a harsh environment. Therefore, the staff finds that the RCCWS complies with the environmental provisions of GDC 4.

9.2.2.4.4 GDC 5, "Sharing of Structures, Systems, and Components"

The staff reviewed the design of the RCCWS for compliance with the requirements of GDC 5 with respect to shared systems among NPMs. Compliance with GDC 5 requires that the nuclear power module designs include provisions to ensure that an event with one NPM does not adversely impact the ability of any other NPM units to perform their safety functions, including the ability to safely achieve and maintain safe shutdown. Meeting these requirements provides a level of assurance that the events will be isolated to one NPM.

Component failures such as failed-open flow control valves or pipe breaks inside containment would not impact the ability of the RCCWS to continue to support the remaining NPMs once the failure was isolated. A loss of RCCWS water into containment does result in a containment flooding event for the associated NPM. The applicant evaluated this event in FSAR Section 15.1.6. Section 15.1.6 of this SER contains the staff's evaluation. The staff finds that

(1) the failure of components in the RCCWS does not significantly impair the ability of other NPMs to perform their safety functions, and (2) the requirement of GDC 5 with regard to sharing of systems between units is satisfied.

9.2.2.4.5 GDC 60, "Control of Releases of Radioactive Materials to the Environment," and GDC 64, "Monitoring Radioactivity Releases"

The staff reviewed the design of the RCCWS for compliance with the requirements of GDC 60 for the control of releases of radioactive materials and GDC 64 for the monitoring of radioactive releases. Compliance with GDC 60 and GDC 64 requires provisions in the nuclear power module design to monitor and suitably control the release of radioactive materials during normal operation, including AOOs.

The RCCWS does not normally contain radioactive process fluid. However, all systems cooled by the RCCWS, with the exception of the CRDMs, contain fluid that has the potential to contaminate the RCCWS with radioactivity, as indicated in FSAR Section 9.2.2.3. The RCCWS is designed as a closed-loop system to act as an intermediate system between radioactive systems and the nonradioactive SCWS which transfers the heat to the environment. The RCCWS is designed to ensure that any contamination is contained within the RXB. Radiation monitors are located downstream of the cooled components, and the design incorporates the ability to isolate and sample potentially contaminated systems, as specified in FSAR Table 11.5-1 "Process and Effluent Radiation Monitoring Instrumentation Characteristics." In addition, a single adjacent-to-line radiation monitor is provided on the normally noncontaminated RCCWS drain tank to ensure the prompt identification of radiological contamination in reactor component cooling water. All coolers and condensers have manual isolation valves to isolate leaks. Based on the above evaluation, the staff finds that the RCCWS design as described in the FSAR complies with the requirements of GDC 60 and GDC 64.

9.2.2.4.6 Compliance with 10 CFR 20.1406, "Minimization of Contamination"

In 10 CFR 20.1406, the NRC requires that each standard design approval applicant shall describe how the facility design will minimize, to the extent practicable, contamination of the facility and the environment, as well as the generation of radioactive waste. The RCCWS reduces the possibility of radioactive leakage to the environment by providing an intermediate barrier between radioactive or potentially radioactive systems and the SCWS.

The staff reviewed the design of the RCCWS for compliance with the requirements of 10 CFR 20.1406. As described in FSAR Section 9.2.2.3, the RCCWS design ensures that any potential contamination is contained within the RXB and that radiation monitors are located downstream of the cooled components to alert the control room if there is a radioactive fluid leak into the RCCWS. All coolers and condensers have manual isolation valves to isolate leaks.

The applicant addressed the compliance of RCCWS design features with RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," in FSAR Table 12.3-31, "Regulatory Guide 4.21 Design Features for Reactor Component Cooling Water System." In the table, the applicant identified the design features for the RCCWS and stated how they address the objectives in RG 4.21. Section 12.3 of this report gives the staff's general review of NuScale compliance with RG 4.21.

Based on the above discussion, the staff finds and concludes that the RCCWS as described in the FSAR complies with 10 CFR 20.1406.

9.2.2.5 Initial Test Program

The staff evaluates the ITP in Section 14.2 of this SER.

9.2.2.6 Technical Specifications

No GTS requirements are associated with the RCCWS.

9.2.2.7 Conclusion

Based on review of the information described above, the staff finds the RCCWS design acceptable because it is consistent with the applicable acceptance criteria of SRP 9.2.2 and meets the applicable regulatory requirements of GDC 2, 4, and 5. In addition, the RCCWS meets GDC 60 and 64 and 10 CFR 20.1406 for minimizing contamination.

9.2.3 Demineralized Water System

9.2.3.1 Introduction

The DWS is not safety-related and is designed to treat the water from the utility water system (UWS) and provide and distribute high-quality demineralized water to the plant. The major components of the DWS include one demineralized water treatment (DWT) skid, one demineralized water storage tank (DWST), and DWS pumps.

The DWS provides normal makeup for pool water evaporation in the reactor pool to ensure sufficient shielding for the spent fuel assemblies (SFAs) and the NPMs for normal operation. During these situations, the operators monitor the DWST water level to ensure its availability for use. The DWS is not required for any DBE.

9.2.3.2 Summary of Application

FSAR Section 9.2.3 gives a general description of the DWS, including information on the system design bases, identification of the system's major components, instrument requirements, and a discussion of system operation, inspection, and testing. There are no proposed ITAAC related to the DWS.

9.2.3.3 Regulatory Basis

The staff determined that no current SRP section is directly applicable to the DWS (SRP Section 9.2.3, "Demineralized Water Makeup System," was withdrawn in December 1996). Consistent with the basis for withdrawing SRP Section 9.2.3, the staff selected applicable portions of SRP Section 9.2.2 and Section 9.2.6, Revision 3, "Condensate Storage Facilities," issued March 2007, as guidance. The following regulatory requirements apply:

- GDC 2, as it relates to the capability of structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without a loss of safety-related functions

- GDC 5, insofar as it requires that SSCs important to safety not be shared among power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions

- 10 CFR 20.1406, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive waste.

9.2.3.4 Technical Evaluation

The staff reviewed the DWS design for compliance with the regulatory basis given in Section 9.2.3.3 of this SER. The following sections give the results of the staff's review.

9.2.3.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

GDC 2 establishes requirements with respect to the DWS design for protection against the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena without a loss of capability to perform their safety functions. The application of GDC 2 to the DWS design ensures that SSCs important to safety will not be adversely affected by DWS failure resulting from the physical interaction of failed portions of the DWS with SSCs important to safety, or because of the effects of discharged fluids from the DWS on SSCs important to safety resulting from the failure of non-seismic portions of the DWS.

Compliance with the requirements of GDC 2 is based on adherence to RG 1.29, Regulatory Position C.1, for the safety-related portion of the system and to Regulatory Position C.2 for the portions of the system that are not safety-related.

FSAR Section 9.2.3.1, "Design Bases," states that the DWS does not perform safety-related functions, is not credited for the mitigation of design basis accidents (DBAs) and has no safe shutdown functions. FSAR Section 9.2.3.3 states that portions of the DWS that are in proximity to seismic Category I SSCs are designed to seismic Category II standards. In general, the DWS is a seismic Category III system because the system is not required to continue operating after a seismic event, and failure of its SSCs is not expected to affect the operability of seismic Category I SSCs or the occupants of the control room. Any portions of the DWS whose structural failure could adversely affect the function of seismic Category I SSCs are seismic Category II, in accordance with FSAR Section 3.2.

The staff reviewed the FSAR information about the DWS and found that for the DWS major components (i.e., DWT skid, DWS pumps, and DWST) and the portions of the DWS located in buildings or plant areas that do not contain or house SSCs important to safety, the system safety and seismic classifications were appropriate because the failure of DWS SSCs will have no impact on plant safety. Therefore, on this basis, the design adheres to RG 1.29, Regulatory Position C.2, for all areas except within the control building (CRB) and RXB.

Within the CRB, the DWS supports the control room ventilation system by providing the normal control room HVAC system (CRVS) water used by the humidifier. While the DWS is generally categorized as seismic Category III, as indicated in FSAR Table 9.2.3-1, when SSCs (or portions thereof) as determined in the as-built plant could, as the result of a seismic event, adversely affect seismic Category I SSCs or result in incapacitating injury to occupants of the control room, they are categorized as seismic Category II, which is consistent with Regulatory Position C.2 of RG 1.29.

The staff finds the DWS design in the CRB adequate because failure of the portion of the DWS in the CRB will have no effect on the control room habitability envelope or any important-to-safety SSCs, as the DWS does not penetrate the control room habitability envelope. Therefore, the staff finds that the portion of the DWS in the CRB is designed in accordance with RG 1.29, Regulatory Position C.2.

Within the RXB, the DWS provides water to a variety of systems that are not safety-related. However, FSAR Table 7.1-4, includes “demineralized water system isolation” as an engineered safety feature (ESF) function. The entry in this table is associated with the isolation of the demineralized water supply to the CVCS makeup pumps and is intended to terminate an inadvertent boron dilution event as described in FSAR Section 15.4.6.1. The demineralized water supply isolation valves are designated as part of the CVCS and are addressed in Section 9.3.4 of this report. The staff concluded that the DWS has no safety-related SSCs within the RXB. Therefore, the staff finds that the DWS isolation valve is designed in accordance with RG 1.29, Regulatory Position C.2. Based on the discussion above, the staff finds the DWS in compliance with GDC 2.

9.2.3.4.2 GDC 5, “Sharing of Structures, Systems, and Components”

Compliance with GDC 5 requires that nuclear power units shall not share SSCs important to safety unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

In the NuScale US460 design, six NPMs share the DWT skid, DWST, and DWS pumps. However, as indicated FSAR, Section 9.2.3.3, “Safety Evaluation,” the DWS has no safety-related or risk-significant functions. Therefore, the DWS has no functions that are impacted if there is an accident in one module coincident with the shutdown and cooldown of the remaining modules. Therefore, the staff finds that the design of the DWS as described in the FSAR complies with the provisions of GDC 5.

9.2.3.4.3 Compliance with 10 CFR 20.1406, “Minimization of Contamination”

In 10 CFR 20.1406, the NRC requires, in part, that each standard design applicant shall describe how the facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and the environment, as well as the generation of radioactive waste.

The DWS does not contain radioactive materials but does interface with some systems that could contain radioactivity. The DWS design incorporates provisions to prevent radioactive material from contaminating and being released to the environment from the DWS. Notably, the DWS distribution lines are equipped with backflow preventers to prevent contamination of the DWS as indicated in FSAR Table 12.3-22.

In addition to design features, the use of appropriate operating procedures and maintenance programs also minimizes contamination. The applicant addressed the programmatic aspects required for compliance with 10 CFR 20.1406 in FSAR Section 12.3. Section 12.3 of this report includes the staff’s review of those features.

Based on the above discussion, the staff concludes that the DWS design as described in the FSAR complies with the requirements in 10 CFR 20.1406 as it (1) provides a means for preventing contamination of the DWS by interfacing systems, (2) incorporates radiation monitors to provide early indication of the leakage of radioactivity into the DWS, and (3) provides the means for system isolation in the event of a DWS line break.

9.2.3.5 Initial Test Program

Initial Test Program: FSAR Section 14.2 (Test # 11) gives information about DWS initial testing. The staff evaluates the ITP in Section 14.2 of this report of this report. The preoperational test related to the DWS is Demineralized Water System Test (11), which ensures the various design aspects related to the DWS are implemented. The test is performed in accordance with FSAR Tables 14.2-11.

9.2.3.6 Conclusion

Based on the review of the information described above, the staff finds the DWS design acceptable because it meets the applicable regulatory requirements, including GDC 2 for protection from natural phenomena, GDC 5 for shared systems, and 10 CFR 20.1406 for the minimization of contamination.

9.2.4 Potable and Sanitary Water Systems

9.2.4.1 Introduction

The potable water system (PWS) and sanitary water system (SWS) are non-safety-related systems that provide potable water for human use and sanitary water collection throughout the plant for treatment and discharge.

9.2.4.2 Summary of Application

FSAR Section 9.2.4, "Potable and Sanitary Water Systems," gives information on the PWS and SWS. NuScale indicated that the PWS and SWS serve no safety-related functions, are not credited for mitigation of DBAs, and have no safe shutdown functions.

FSAR Section 9.2.4.2, "System Description," states that the PWS and SWS provide water to, and accept wastewater from, the control room envelope (CRE). It also states that each PWS and SWS supply and return line that penetrates the CRE includes a passive isolation device (loop seal) located inside the CRE.

Initial Test Program: FSAR Section 14.2 (Test # 09) gives information about PWS initial testing. There are no proposed ITAAC related to the PWS and SWS.

9.2.4.3 Regulatory Basis

The relevant requirements of the Commission's regulations for this area of review, and the associated acceptance criteria, are as follows:

- GDC 2, as it relates to the capability of structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of safety-related functions.

- GDC 5, as it relates to the capability of shared systems and components to perform required safety functions.

- GDC 60, as it relates to design provisions for the control of the release of liquid effluents containing radioactive material to prevent the contamination of the Potable and Sanitary Water Systems.

- 10 CFR 20.1406, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive.

There are no review interfaces identified for this section.

9.2.4.4 Technical Evaluation

The staff reviewed the PWS and SWS design in accordance with the review procedures in SRP Section 9.2.4, Revision 3. The sections that follow give the results of the staff's review.

9.2.4.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

GDC 2 establishes requirements with respect to the PWS design regarding protection against the effects of natural phenomena such as earthquakes, tornados, hurricanes, and floods. GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. The application of GDC 2 to the PWS design ensures that SSCs important to safety will not be adversely affected by PWS failure due to a seismic event.

Compliance with the requirements of GDC 2 is based on adherence to Regulatory Position C.1 of RG 1.29, for the safety-related portion of the system, and Regulatory Position C.2, for the non-safety-related portions of the system.

FSAR Section 9.2.4 states that the PWS has no safety-related functions, is not credited for mitigation of DBAs, and has no safe-shutdown functions. FSAR Table 9.2.4-1 indicates that all components of the PWS are not safety-related, not risk significant, and seismic Classification III, except for PWS and SWS piping (including loop seals) penetrating the CRE, which are classified as seismic Category I. FSAR Table 9.2.4-1 also states that portions of the system that are in proximity to seismic Category I SSCs are designed to seismic Category II.

The PWS piping in the CRB penetrating the CRE and habitability boundary is provided with isolations to control the potential for flooding in the envelope in the event of a line break and loss of system pressure. FSAR Section 9.2.4.2 states that each PWS supply and return line to or from the CRE includes a passive isolation device (loop seal) located inside the CRE. If a line is damaged by a seismic event, it is isolated by the loop seal to protect the control room from in-leakage of atmospheric radioactive contaminants.

The staff finds that specifying that the PWS supply and return lines from the CRE outer wall to the isolation device are to be seismic Category II will ensure that the PWS would not fail in a way that would result in incapacitating injury to occupants of the control room or cause the failure of seismic Category I SSCs that are required for a safe shutdown. In addition, since the credited isolation is accomplished based on the use of passive design features instead of isolation valves, the staff finds the system design to be sufficient to protect against in-leakage of radioactive contaminated air into the control room and ensures CRE integrity under a seismic event.

In FSAR Section 9.4.1., NuScale stated that under certain postulated conditions the CRE is isolated, and air is provided by the control room habitability system (CRHS). NuScale discussed the control room habitability and the CRE in FSAR Section 6.4. In that section, NuScale described the CRHS as a non-safety-related system that is designed to provide breathable air to the control room during the first 72 hours following an accident. In its design evaluation presented in FSAR Section 6.4.4, NuScale stated that the CRE and the supporting habitability systems and components are not safety-related. In addition, FSAR Section 9.4.1.1, states that the CRVS serves no safety-related functions, is not credited for mitigation of DBAs, and has no safe-shutdown functions. Since the CRE is classified as not safety-related and the CRHS can provide breathable air to the control room during the first 72 hours following an accident, the staff finds that the PWS design complies with GDC 2.

9.2.4.4.2 GDC 5, "Sharing of Structures, Systems, and Components"

Since the Potable and Sanitary Water Systems serves no safety-related functions, is not credited for mitigation of DBAs, and has no safe-shutdown functions, there are no safety-related, risk-significant, or safe-shutdown functions in the PWS that are shared between NPMs. The staff finds that the design of the PWS, and SWS, as described in the FSAR does not incorporate sharing of any important to safety SSCs among the nuclear power modules. Therefore, the PWS and SWS comply with the provisions of GDC 5.

9.2.4.4.3 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

The staff reviewed the design of the Potable and Sanitary Water Systems for compliance with the requirements of GDC 60 for the control of releases of radioactive materials. FSAR Section 9.2.4.3 states that the PWS and SWS piping is not interconnected with other system piping that conveys radioactive materials, and that the system employs backflow prevention measures, such as backflow preventers and air gaps, that separate the PWS and SWS from interfacing water systems to prevent cross contamination. Based on the measures used in the design of the PWS, as described above, the staff finds that the PWS satisfies GDC 60 with respect to preventing PWS contamination by interfacing with radioactive or potentially radioactive systems.

9.2.4.4.4 10 CFR 20.1406, "Minimization of Contamination"

As discussed in Section 9.2.4.4.3 of this report, PWS piping is not interconnected with other system piping that conveys radioactive materials. Additionally, PWS SSCs are protected from contamination by being located separate from contaminated systems. By minimizing the probability of the PWS being cross contaminated, radiation levels in plant areas being served by the PWS are also minimized. Therefore, the staff concludes that the PWS design complies with the requirement in 10 CFR 20.1406.

9.2.4.5 Initial Test Program

The preoperational test related to the PWS is Potable Water System Test (09) which ensures the various design aspects related to the PWS are implemented. The test is performed in accordance with FSAR Tables 14.2-911. The staff evaluates the ITP in Section 14.2 of this report.

9.2.4.6 Combined License Information Items

Table 9.2.4-1 lists the PWS related COL information items and descriptions for the PWS from Table 1.8-1 of the FSAR.

Table 9.2.4-1 NuScale COL Information Items for Section 9.2.4

COL Item No.	Description	FSAR Section
14.2-5	An applicant that references the NuScale Power Plant US460 standard design will provide a test abstract for the potable water system pre-operational testing.	14.2

9.2.4.7 Conclusion

Based on the review of the information described above, the staff finds the PWS and SWS designs acceptable because they are consistent with applicable SRP acceptance criteria and meet the applicable regulatory requirements, including GDC 2 and 5. In addition, the PWS meets the requirements of GDC 60 for controlling radioactive effluent releases and 10 CFR 20.1406 on the minimization of contamination.

9.2.5 Ultimate Heat Sink

9.2.5.1 Introduction

The NuScale US460 UHS is a set of safety-related pools of borated water that consists of the combined water volume of the reactor pool, RFP, and SFP. The UHS pools are located below grade in the RXB. Up to six NPMs are located in the reactor pool and share the combined volume of water. The UHS provides several safety functions, including (1) serving as a cooling medium for the decay heat removal system (DHRS), CNVs, and the spent fuel assemblies stored in the storage racks, (2) providing borated water for reactivity control during refueling, and (3) shielding radiation for the spent fuel assemblies and NPMs. During accident scenarios, the NuScale US460 design credits the safety-related water inventory stored in the UHS to passively remove the decay heat. NuScale considered GDC 2, 4, 5, 45, 46, and 61 and Principal Design Criterion (PDC) 44, “Cooling Water,” in the design of the UHS. The staff evaluates the safety-related UHS function in this section of the SER.

FSAR, Section 9.1.2, discusses the design of provisions of the SFP and support components to ensure adequate safe storage of the spent fuel. FSAR, Section 9.1.3, discusses the design and performance of the pool support systems, which include the spent fuel pool cooling subsystem, the pool clean up subsystem, and the pool surge control subsystem. The staff evaluates these pool-related systems in Sections 9.1.2 and 9.1.3 of this report. This section of the SER addresses the safety-related, passive function of maintaining the spent fuel covered and cooled.

9.2.5.2 Summary of Application

FSAR, Section 9.2.5 provides information on the UHS design bases, system description, normal operation, operation during abnormal and accident conditions, refueling operations, safety evaluation, inspection and testing requirements, and instrumentation requirements. In addition, FSAR, Figure 9.2.5-1, provides the basic layout of UHS pools. FSAR, Table 9.2.5-1, lists relevant UHS parameters, and FSAR, Table 9.2.5-2, “Ultimate Heat Sink Heat Loads: Boil-off Analysis Results,” lists the analysis results of a boiloff event.

SDAA Part 8, Section 3.6, and Table 3.6-1, provides ITAAC for UHS piping and connections. These ITAAC are evaluated in Section 14.3 of this report. There are no ITP for the UHS.

9.2.5.3 Regulatory Basis

SRP Section 9.2.5, Revision 3, "Ultimate Heat Sink," issued March 2007, SRP Section 9.1.3, and SRP 19.3 provide the relevant regulatory requirements for this area of review and the associated acceptance criteria, as summarized below:

- GDC 2, as it relates to the capability of structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.

- GDC 4, as it relates to the protection of SSCs important to safety from the dynamic effects of missiles resulting from equipment failures.

- GDC 5, as it relates to the capability of shared systems and components important to safety to perform required safety functions.

- GDC 44, as it relates to the following:

    - the capability to transfer heat loads from safety-related SSCs to the heat sink under both normal operating and accident conditions

    - suitable component redundancy so that safety functions can be performed assuming a single, active component failure coincident with a loss of offsite power

    - the capability to isolate components, systems, or piping if required so safety functions are not compromised

    The applicant has provided rationale to support that an exemption request requested an exemption from certain electrical power provisions of GDC 44 and, as described in FSAR, Section 3.1.4.15, identified PDC 44, which eliminates consideration of onsite and offsite electrical power, would be justified. The staff's evaluation regarding the rationale provided to support such an exemption that supports PDC 44 is documented provided in Section 8.1.5 of this report.

- GDC 45, as it relates to the design provisions to permit inservice inspection of -safety-related components and equipment.

- GDC 46, as it relates to the design provisions to permit pressure and functional testing of safety-related systems or components.

- GDC 61, as it relates to the requirement that the fuel storage system be designed to ensure adequate safety under normal and postulated accident conditions, including the capability to permit appropriate periodic inspection and testing of components important to safety; suitable shielding for radiation protection; appropriate containment, confinement, and filtering capability; residual heat removal capability that reflects the importance to safety of decay heat and other residual heat removal; and the capability to prevent a significant reduction in fuel storage coolant inventory under accident conditions.

9.2.5.4 Technical Evaluation

The UHS typically consists of an assured supply of water that is credited for dissipating reactor decay heat and essential station heat loads after a normal reactor shutdown or a shutdown following an accident or transient, including a LOCA. SRP Section 9.2.5 provides guidance for evaluating the capability of water sources to perform the UHS function in accordance with the requirements of GDC 2, 5, 44, 45, and 46. The SFP is an integral part of the UHS, and the volumes of water between these two pools are in communication (when the pool level is above the separation weir). Therefore, this section of the SER evaluates the safety-related function of maintaining the spent fuel covered and cooled during all scenarios. SRP Section 9.1.3 gives guidance for the evaluation of the design provisions credited to provide adequate coverage and cooling of the stored fuel in accordance with GDC 4 and 61. The staff reviewed the UHS described in the FSAR in accordance with the applicable sections of SRP Sections 9.1.3 and 9.2.5. Section 9.1.3 of this report contains the staff's evaluation of the pool support systems, as described in FSAR, Section 9.1.3.

9.2.5.4.1 Principal Design Criterion 44, "Cooling Water"

General Design Criterion 44 versus Principal Design Criterion 44

SRP Section 9.2.5 states that GDC 44 applies to the UHS as it relates to the capability to transfer heat loads from safety-related SSCs to the heat sink under both normal operating and accident conditions. This section of this report contains the staff's review of the heat transfer to the UHS under accident conditions, and Section 9.1.3 of this report includes the review under normal operating conditions.

As part of Chapter 8 of this SER, NuScale provided rationale to support that an exemption request from the requirements of electrical power requirements in GDC 17, "Electric power systems," and GDC 18, "Inspection and testing of electric power systems," and the electric power provisions of GDC 33, 34, 35, 38, 41, and 44 would be justified. As indicated in Chapter 8, the staff evaluated the rationale provided and concluded that the rationale supports that such an exemption request would be justified. In FSAR, Section 3.1.4.15, the applicant proposed PDC 44 instead of GDC 44 to eliminate electrical power provisions associated with onsite and offsite power. The applicant adopted the following definition of PDC 44:

> A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that the system safety function can be accomplished, assuming a single failure.

The staff evaluated PDC 44 and found that it proposed to remove the requirements related to electric power systems, without changing the requirements of the cooling capability identified in GDC 44. Therefore, the staff concluded that PDC 44 would adequately address the necessary capabilities for the UHS.

Ultimate Heat Sink Cooling Capability

When the SFP water level is above the weir separating the SFP from the RFP, the waters of these two sections of the UHS are in communication, and any accident scenario will impact both sections. GDC 61, related to the system design for fuel storage, requires the design to provide for a decay heat removal capability that reflects its importance to safety and include the capability to prevent a reduction in the fuel storage coolant inventory under accident conditions. The staff evaluated the UHS cooling capability during accident conditions against PDC 44 and the applicable portions of GDC 61.

FSAR Section 9.2.5.2.1, states that the UHS will remove the decay heat from each NPM, the stored spent fuel assemblies in the SFP, refueling activities, during normal and accident conditions assuming a single failure for at least 72 hours without operator actions or electrical power, either alternating current (AC) or direct current (DC). The applicant also stated that the UHS water level continues to perform these functions for more than 7 days following an accident.

The staff evaluated the system description and confirmed that the NuScale US460 design does not credit any non-safety-related system with performing functions during the initial 7 days of coping with an event. Therefore, the staff determines that NuScale does not credit SSCs to perform a regulatory treatment of non-safety systems (RTNSS) B function per SRP 19.3.

The staff reviewed FSAR, Section 9.2.5, on the UHS design capacity for normal, abnormal, transient, and accident conditions, including the size and heat loads of the UHS, to verify the adequacy of the long-term UHS capacity. The UHS thermal analysis presented the boiloff calculation, heat loads, assumptions, initial conditions, water level, and initial pool temperature to demonstrate the adequacy of the long-term cooling capacity of the UHS. The applicant stated that the UHS is a passive system and does not require electric power (AC or DC) to remove heat. The applicant stated that the heatup analysis assumes that an accident resulting in the shutdown of one NPM happens concurrently with a loss of AC power that results in the shutdown of the remaining NPMs. The volume of water already in the pool provides the inventory for the necessary heat removal from the power modules and spent fuel for greater than 30 days without the need for operator action, makeup water, or electric power. In addition, personnel can add makeup water through the seismic category I UHS makeup line from outside of the RXB using nonsafety-related equipment to stabilize pool water inventory.

The initial conditions of the analysis are consistent with GTS LCO 3.5.3. GTS LCO 3.5.3 identifies that the UHS water level shall be maintained between 15.8 m (52 ft) and 16.5 m (54 ft) from the bottom of the pool. The analysis demonstrates that the minimum water level needed for maintaining 72 hours of coverage of the DHRS is 14.7 m (48.2 ft) of water.

The staff noted that in FSAR, Table 9.2.5-1, “Relevant Ultimate Heat Sink Parameters,” the applicant indicated that the PCWS penetrates the UHS at 15.1 m (49.5 ft). Based on the relative elevation of the pipe penetration, the staff determined that the UHS minimum safety water level of 14.7 m (48.2 ft) is adequately protected from siphoning.

The scope of the SFP design in the FSAR does not include spent fuel racks. COL Item 9.1-2 has been included for the COL applicant to provide the design of the spent fuel racks. However, as described in FSAR, Section 9.2.5.3, the pool boiloff thermal calculation conservatively assumes that the SFP contains 10 years of accumulated spent fuel assemblies and five additional failed fuel assemblies, and that the analyzed conditions occur at a point in the refueling schedule that maximizes the heat load contributed by the spent fuel assemblies. The assumed refueling schedule consists of three modules refueled in succession every 9 months. The thermal evaluation also assumes no heat transfer between the UHS and the pool walls, which means that all the decay heat is dissipated by the steaming pool water and not by heating the wall components. The thermal analysis assumes that the evaporated pool water is released to the environment. Therefore, the applicant did not credit condensing water returning to the UHS. The staff evaluated these assumptions and determined them to be conservative and acceptable.

In determining the maximum heat load from the NPMs, the applicant indicated that the analysis assumes that an accident will occur in one NPM with a coincident loss of AC power and the shutdown of five NPMs. For a plant with one NPM in refueling operations at the time of an accident, the total heat load to the UHS would be less than the limiting case with six NPMs in operation. The staff discusses its evaluation of the different DBAs in Chapter 15 of this report. The staff finds that using the highest heat load from the DBAs discussed in FSAR, Chapter 15, is a conservative assumption and, therefore, acceptable.

SRP Section 9.2.5 indicates that the UHS shall have the capacity to dissipate the maximum possible total heat load, under the worst combination of adverse environmental conditions, to cool the module (or units) for a minimum of 30 days without makeup unless acceptable makeup capabilities can be demonstrated.

The FSAR indicated that the UHS has sufficient water inventory to remove the decay heat from the NPM and the stored fuel from the pool through boiling and evaporation, removing enough heat to maintain the spent fuel and fuel in the NPMs sufficiently cool to prevent fuel damage. The boiling and evaporation of the UHS water increases the pressure inside the RXB. Section 9.1.3 of this report addresses how the NuScale US460 design vents the steam, created by the passive boiling and steaming of the UHS water, from the RXB.

9.2.5.4.2 GDC 2, "Design Bases for Protection against Natural Phenomena"

SRP Section 9.2.5 states that GDC 2 applies to the UHS as it relates to the capability of structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.

FSAR, Section 9.2.5.2.1, stated that the structural components forming the reactor pool, RFP, and SFP, including the base, structural walls, and weir wall, are part of the RXB structure. FSAR, Section 3.8.4.1.1, describes the RXB and it states that the RXB is designed to withstand the effects of natural phenomena (earthquake, rain, snow, wind, tornado, hurricane) without affecting operability of the safety-related SSCs in the building.

In addition, FSAR, Section 9.2.5.2.1, stated that "the UHS has a makeup line that meets Regulatory Guide (RG) 1.26, Quality Group D standards, RG 1.29 Seismic Category I standards, and American Society of Mechanical Engineers B31.1 requirements, and is protected from external natural phenomena. The UHS makeup line includes a fire protection connector that facilitates hookup of emergency sources of water for the water supply."

The UHS includes the dry dock area. The applicant did not credit this volume of water to be available to provide UHS makeup. The applicant also stated that a failure of this gate is not postulated (when the dry dock is drained) because the dry dock gate is designed to seismic

Category II requirements and the dry dock gate supports are design as seismic Category I, which is consistent with RG 1.29.

Additional information on GDC 2 compliance for the SFP function is in FSAR, Sections 9.1.2 and 9.1.3, and SER Sections 9.1.2 and 9.1.3 contains the staff's evaluation.

Based on the seismic design of the RXB and the UHS as discussed above, the staff finds that the UHS is adequately designed and protected against the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods, to remain functional following a natural phenomenon.

9.2.5.4.3 GDC 4, "Environmental and Dynamic Effects Design Bases"

Compliance with GDC 4 requires that SSCs important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs and dynamic effects from pipe whip, missiles, and discharging fluids.

FSAR Section 9.2.5.3 indicates that the UHS is located within RXB structures that provide protection from the effects of turbine missiles, without loss of the UHS safety functions. FSAR, Section 3.5, "Missile Protection," provides additional detail on protection from turbine missiles. The staff evaluates missile protection in Section 3.5 of this report. The applicant also indicated that the RXB structures are designed to withstand environmental and dynamic effects, including the effects of postulated missiles, pipe whip, and discharging fluids that may result from equipment failures and from events and conditions that may occur within the RXB but outside the UHS boundary.

The applicant stated that the UHS is protected from freezing temperatures because (1) there are several resident heat sources in the UHS, (2) the RXB HVAC system controls the environment inside the RXB, and (3) the UHS is below grade.

The UHS is a large pool of water, and a seismic event is capable of generating wave motion. The applicant indicated that an analysis of sloshing has been completed and the maximum expected wave in the UHS is less than 3.5 ft tall. In Section 9.2.5.3 the applicant acknowledged that the pool water could overtop the west end and the south-west corner of the refueling pool and spill into an adjacent room. The applicant indicated that the adjacent room does not contain safety-related or risk-significant SSCs. After this, the UHS will still retain sufficient water to perform its safety function.

The staff evaluated the information discussed above and finds that the applicant's design complies with the requirements of GDC 4, in that SSCs important to safety are protected against the effects of missiles from events and conditions outside the nuclear power module and the effects of environmental pool sloshing dynamics.

Section 3.6.1 of this SER discusses the staff evaluation of the protection of essential SSCs against pipe failures. Additional information regarding GDC 4 compliance for the SFP function is in FSAR, Sections 9.1.2 and 9.1.3; SER Sections 9.1.2 and 9.1.3 contain the staff's evaluation.

9.2.5.4.4 GDC 5, "Sharing of Structures, Systems, and Components"

SRP Section 9.2.5 states that GDC 5 applies to the UHS as it relates to the capability of shared systems and components important to safety to perform required safety functions.

FSAR, Section 9.2.5.3, states that the UHS is a shared system that is capable of providing sufficient cooling to dissipate the heat from an accident in one module and permitting the simultaneous and safe shutdown of the remaining units and maintaining them in a safe shutdown condition, without requiring makeup water.

The staff verified that the NuScale UHS design capacity for abnormal and accident conditions, as described in FSAR, Section 9.2.5, includes the combined heat loads from all NPMs and the SFP. Therefore, the staff concludes that GDC 5 is satisfied, as it relates to the capability of the shared UHS to perform required safety functions.

9.2.5.4.5 GDC 45, "Inspection of Cooling Water System," and GDC 46, "Testing of Cooling Water System"

SRP Section 9.2.5 states that GDC 45 and 46 apply to the UHS as they relate to the design provisions to permit inservice inspection and testing of safety-related components and equipment.

FSAR, Section 9.2.5.3, states that the NuScale US460 UHS design conforms to GDC 45 and 46. The UHS design permits inspections and tests that verify its continued performance, integrity, and safety. The pools that comprise the UHS are accessible for periodic inspections. The UHS structural leak tight integrity is demonstrated by maintaining the pool water level and monitoring for leaks through the pool leak detection system. These inspections and tests verify system integrity and operability as a whole. The UHS does not rely on any active components to perform the required safety functions.

Based on the above, the staff verified that the NuScale US460 UHS design conforms to GDC 45 and 46 because the proposed provisions for inspection and testing are consistent with the guidance in SRP Section 9.2.5 on inspection and testing of the UHS.

9.2.5.4.6 GDC 61, "Fuel Storage and Handling and Radioactivity Control"

The NuScale US460 UHS contains the SFP. Compliance with GDC 61 requires that the SFP do the following:

- Demonstrate the capability for the periodic testing of components important to safety.
- Provide for containment.
- Include provisions for decay heat removal that reflect its importance to safety.
- Prevent reduction in fuel storage coolant inventory under accident conditions.
- Demonstrate the capability and capacity to remove corrosion products, radioactive materials, and impurities from the pool water and reduce occupational exposures to radiation.

The staff evaluates the design for compliance with GDC 61, as it relates to decay heat removal capability during an accident scenario, in Section 9.2.5.4.1 of this report. The staff evaluates the SFP for compliance with GDC 61 in Section 9.1.3.4.4 of this report.

9.2.5.4.7 Leakage and Makeup

FSAR, Section 9.1.3, discusses potential UHS pool leakage and water makeup. Section 9.1.3 of this report documents the staff's review of SFP/UHS leakage and makeup.

9.2.5.4.8 Instrumentation

SRP Section 9.1.3.I.2.G indicates that the review should consider the instrumentation provided for initiating appropriate safety actions. SRP Section 9.2.5 indicates that the main safety function of the UHS is to dissipate the decay heat of all NPMs and stored spent fuel assemblies for abnormal and accident conditions.

FSAR, Section 9.2.5.4, "Instrumentation Requirements," provides the NuScale US460 instrumentation design information. The SFP cooling system temperature instrumentation is used to monitor the UHS and is discussed in FSAR, Section 9.1.3. Section 9.1.3 of this SER discusses the staff's evaluation of the temperature instrumentation.

The level instrumentation is discussed in FSAR, Section 9.2.5.4.2, which states that the UHS has two level instruments located in the SFP capable of monitoring water level from the normal UHS level to the top of the stored spent fuel in the SFP. In addition, the reactor pool and the RFP each have another level instrument. The level instruments are designed as seismic Category I. The instruments are powered from the plant lighting system, and each instrument has a dedicated battery backup power supply capable of powering the instrument for 14 days.

The instruments are qualified to operate at saturation conditions for an extended period of time. The instrumentation is protected from natural phenomena, physically separated and mounted at opposite ends of the pools. The instruments display information in the control room. The instrument alarm alerts the operators of these parameters during both normal and post-accident conditions.

The staff confirmed that the safety function of the UHS, to adequately dissipate the long-term decay heat, would require UHS pool temperature and level instrumentation, which the NuScale US460 design includes for initiating appropriate safety actions. Based on the above, the staff finds that the application has adequately addressed the UHS instrumentation because it satisfies the functional requirement for the UHS and is consistent with the applicable guidance in SRP Sections 9.1.3 and 9.2.5.

9.2.5.5 Initial Test Program

The applicant has not identified any preoperational tests for the UHS.

9.2.5.6 Technical Specifications

SDAA Part 4 provides GTS in LCO 3.5.3 to specify the level, temperature, and boron concentration in the UHS for all times. The specified level and temperature are related to the thermal analysis discussed in SER Section 9.2.5.4.1.

The SER for FSAR Chapter 16 provides additional review of the TS.

9.2.5.7 Combined License Information Items

The applicant has not identified any COL for the UHS system, however, the SFP is part of the UHS and SFP COL Item 9.1-2 is discussed in Section 9.1.2 of this report.

9.2.5.8 Conclusion

Based on the above, the staff has determined that the proposed UHS design and guidance as described in the application for the UHS are consistent with the acceptance criteria in SRP Sections 9.2.5 and 9.1.3 and GDC 2, 4, 5, 45, 46, and 61 and PDC 44.

9.2.6 Condensate Storage Facilities

9.2.6.1 Introduction

The condensate storage facilities (CSFs) use a condensate storage tank (CST) to support each NPM's condensate and feedwater system. The CST provides a volume for makeup and rejection of condensate to and from the condensate collection tank (CCT) based on condensate collection tank level. The CST is not safety-related, does not serve an important to safety function, and does not interface with other systems that could adversely affect safety-related or augmented equipment.

9.2.6.2 Summary of Application

The CST supplies makeup water to the steam cycle. FSAR Section 9.2.6, "Condensate Storage Facilities," provides very limited information on the system. It states that the CSF includes the tank, piping, valves, tank level instrumentation, vents, drains, and connections to the CCT. It also states that the CST does not provide makeup water to systems that remove heat from the reactor if normal heat removal methods are unavailable.

There are no proposed ITAAC related to the CST.

9.2.6.3 Regulatory Basis

In general, SRP Section 9.2.6 gives the relevant regulatory requirements for this area of review and the associated acceptance criteria, as well as review interfaces with other SRP sections. Because the CSF is not safety-related, is not credited for providing water to safety-related cooling systems, and has no safety-related functions, only the following requirements are relevant to this design:

- GDC 2, as it relates to the system's capability to withstand the effects of natural phenomena, including earthquakes and tornadoes.
- GDC 5, as it relates to the capability of shared systems and components to perform required safety functions.
- GDC 60, as it relates to tanks and systems handling radioactive materials in liquids.
- 10 CFR 20.1406, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive waste.

9.2.6.4 Technical Evaluation

The staff reviewed the CSF design in accordance with the review procedures in SRP Section 9.2.6. The sections that follow give the results of the staff's review.

9.2.6.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

Compliance with the requirements of GDC 2 is based on adherence to Regulatory Position C.1 of RG 1.29, for the safety-related portion of the system, and Regulatory Position C.2, for the non-safety-related portions of the system.

In FSAR Section 9.2.6, the applicant stated that the CST does not serve a safety function, and it does not interface with other systems that could adversely affect safety-related or augmented systems. Section 9.2.6 also states that the CST is located outside the turbine generator building (TGB).

The staff reviewed the FSAR information on the CSF and found that the FSAR did not contain a description of the CSF system or a piping and instrumentation drawing. The FSAR addressed only the CST, which it identified as being located outside the TGB. However, FSAR Section 10.4.1.2.2, "System Operation," states that the CCT level is regulated automatically using makeup from the CST; thus, the portions of the CSF that connects the CST to the CCT are also located outside in the yard. Because the CSF system has no safety-related portions, only Regulatory Position C.2 of RG 1.29 is applicable. FSAR Table 10.4-4 indicates that the CSTs, which are located outside the TGB, are specified as seismic Classification III. The failure of these tanks may result in the discharge of large volumes of fluids. Based on the location of the CST and other CSF system components, the staff found that the failure of the CSF would not have an adverse impact on SSCs important to safety.

As indicated in SRP Section 9.2.6.I, for the CSF, the staff reviewed the provisions for mitigating the environmental effects of system leakage or storage tank failure. FSAR, Section 3.4.1.4, "Flooding Outside the Reactor Building and the Control Building," states that for the RXB and CRB, water from tanks and piping that are nonseismic and not protected from tornadoes or hurricanes is a potential flooding source outside the buildings.

The staff reviewed the system description information in FSAR Section 9.2.6, the plant layout description in FSAR Chapter 1, "Introduction and General Description of the Plant," and the information on flooding outside the RXB and CRB in FSAR Section 3.4.1.4. Because the CSTs are nonseismic, a seismic event could cause the failure of as many as six CSTs, which would result in the release of the water stored in the CSTs. The staff was unable to verify the proximity of the CSTs to the CRB and RXB based on information in the FSAR; however, the staff did find that COL Item 3.4-1 directs COL applicants that reference the NuScale US460 design to confirm that site-specific tanks or water sources are placed in locations where they cannot cause flooding of the RXB or CRB. In addition, the staff found that the water released from CST failures would not adversely affect SSCs important to safety because, as stated in FSAR, Section 3.4.1.4, the site is graded to transport water away from the RXB and CRB. Therefore, the failure of equipment outside the RXB and CRB will not cause internal flooding inside those buildings. Based on the above discussion, the staff finds that the NuScale US460 CSF complies with GDC 2.

9.2.6.4.2 GDC 5, "Sharing of Structures, Systems, and Components"

The staff's evaluation found that the design of the CSF, as described in FSAR Section 9.2.6, does not share SSCs important to safety with any of the nuclear power modules, and the CSF does not affect the plant's ability to achieve safe and orderly shutdown and cooldown of the NPMs because the CSF is not relied on for safety-related cooling, and its failure will not adversely affect plant SSCs important to safety. Therefore, the staff finds that the CSF complies with the provisions of GDC 5.

9.2.6.4.3 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

The staff reviewed the design of the CSF for compliance with the requirements of GDC 60 for the control of releases of radioactive materials. According to SRP Section 9.2.6.II, the guidance in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" is an acceptable way of meeting the GDC 60 requirement.

FSAR, Section 9.2.6, states that, in accordance with RG 1.143, CST instrumentation includes a high level alarm. The alarm gives an early indication of a potential tank overfill.

The staff reviewed FSAR Figure 9.2.6-1, "Condensate Storage Facility." The drawing shows two input lines to the CST, one line from the DWS, which provides demineralized water to the CST, and a second line from the condensate and feedwater system, which returns condensate from the condensate and feedwater system to the CST when determined to be necessary by the CCT level controller. The CST level controller controls the demineralized water, which is the largest source of water to the CST. The control room has high level alarms.

The CST level instrument will minimize the likelihood of tank overflow because it will alert operators to high CST levels, which would allow the operators to stop the flow of water to the CST before an overflow condition occurs. Because the concentration of radionuclides in the CST is normally expected not to be significant, and the CST includes a feature that prevents or minimizes the potential for overflow, the staff finds the NuScale US460 design to be in compliance with GDC 60.

9.2.6.4.4 10 CFR 20.1406, "Minimization of Contamination"

As discussed in SRP Section 9.2.6.II, the CSF potentially contains radioactive material through its connections with the secondary coolant system.

The design of the CSF is consistent with the risk-informed approach in RG 4.21, Regulatory Position C.1.2. The applicant also stated that, consistent with 10 CFR 20.1406, the instrumentation for each CST includes high- and low-level alarms giving early indication of the potential for a tank overfill or a substantial leak from the tank. Furthermore, COL Item 12.3-6 directs the COL applicant to develop, in part, processes and programs to demonstrate compliance with 10 CFR 20.1406 and the guidance in RG 4.21.

Based on the staff's review of the design of the CSF, the staff finds that the CSF, as designed, will contain leak detection capability and will be able to accommodate inspections, if necessary, to locate, identify, and repair leaks that may occur during the life of the plant. Therefore, the staff finds the CSF design as described in the SDAA to be in compliance with 10 CFR 20.1406.

9.2.6.5 Initial Test Program

The preoperational test related to the CSF is Air Cooled Condenser System Test (07), which is performed to ensure the condensate storage tank ability to support the air cooled condenser system (ACCS) by providing makeup water to maintain water level in the condensate collection tank. The staff evaluates the ITP in Section 14.2.

9.2.6.6 Technical Specifications

No GTS requirements are associated with the CSF.

9.2.6.7 Combined License Information Items

In accordance with FSAR Table 1.8-2 and Section 9.2.6, the applicant has not identified any COL information items that are directly applicable to the CSF. The staff did not identify any additional COL information items that should be in FSAR, Table 1.8-1.

9.2.6.8 Conclusion

Based on the review of the information described above, the staff finds the CSF design acceptable because it meets the applicable regulatory requirements, including GDC 2, 5, and 60 and the provisions of 10 CFR 20.1406 for the minimization of contamination as it applies to CSF SSCs that may have the potential to release radioactive materials to the facility, site, or environment.

9.2.7 Site Cooling Water System

9.2.7.1 Introduction

The purpose of the SCWS is to transfer heat from plant auxiliary systems to the SCWS cooling towers. The SCWS is a nonsafety-related system and is not required to operate during and after a DBE. Serviced loads for the SCWS include equipment in the RXB, central utility building, and TGB.

9.2.7.2 Summary of Application

FSAR Section 9.2.7, “Site Cooling Water System,” describes the SCWS. The SCWS supports the following systems by providing cooling water to turbine generator system (TGS), RCCWS, ACCS, PSS, Chilled Water System (CHWS), instrument and control air system (IAS), PCWS, and FWS.

ITAAC: FSAR, Part 8, Table 3.9-1 (Item 11), contains ITAAC to test the SCWS automatically responds to a SCWS high-radiation signal by closing the SCWS blowdown isolation valve. The ITAAC are evaluated in Section 14.3 of this report.

Initial Test Program: Table 14.2-8 (Test # 08) of the ITP indicates the SCWS supports the following systems by providing cooling water to the turbine generator system, RCCWS, ACCS, PSS, chilled water system (CHWS), instrument and control air system (IAS), PCWS, and FWS. The ITP is evaluated in Section 14.2 of this report.

9.2.7.3 Regulatory Basis

Although FSAR Section 9.2.1 states that the NuScale Power Plant does not have a service water system, the SCWS serves the same function as a typical LWR service water system in terms of the ability to cool nonsafety-related auxiliary components used for normal plant operation. Therefore, the staff relied on SRP Section 9.2.1, Revision 5, "Station Service Water System," issued March 2007, for its review of the NuScale SCWS and SRP Section 9.2.5, Revision 3, "Ultimate Heat Sink," issued March 2007, for the evaluation of the site cooling water tower. These two SRP sections identify the following relevant regulatory requirements for this area of review, as well as the review interfaces with other SRP sections:

- GDC 2, as it relates to the capabilities of structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without a loss of safety-related functions.

- GDC 4, as it relates to the effects of missiles inside and outside containment, the effects of pipe whip and jets, environmental conditions from high- and moderate-energy line breaks, and dynamic effects of flow instabilities and attendant loads (e.g., water hammer) during normal plant operation and upset or accident conditions.

- GDC 5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions.

- GDC 60, as it relates to the nuclear power module design including provisions to suitably control the releases of radioactive materials in gaseous and liquid effluents during normal operation, including anticipated operational occurrences.

- 10 CFR 20.1406, as it relates to the standard plant designs and how the design and procedures for operation will minimize contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

Because the NuScale SCWS is not safety-related and its failure does not adversely impact safety systems, the requirements of GDC 44, 45, and 46 and the guidance of SRP Sections 9.2.1 and 9.2.5 on safety-related systems do not apply.

9.2.7.4 Technical Evaluation

The SCWS is not a safety-related system. It supplies cooling water to plant auxiliary systems in the RXB, central utility building, and TGB. Specifically, the SCWS provides cooling water to the CHWS, RCCWS, PCWS, PSS chillers, turbine generator heat coolers, and instrument air system coolers.

The SCWS is a two-loop system comprising a closed-loop subsystem that interfaces with plant loads, and an open-loop subsystem that rejects heat to the environment. The major components of the closed-loop subsystem include the closed-loop pumps, heat exchangers, and a pressurized tank. The major components of the open-loop subsystem include the cooling tower pumps, the cooling towers and associated basin, stationary screens, and the water treatment skid. Both subsystems include a standby pump that automatically starts when a low-pressure condition is detected in its associated system.

As stated in Section 9.2.7.2, the UWS raw water pumps provide makeup to the SCWS tower basin and the DWS provides makeup to the closed-loop portion of the SCWS.

9.2.7.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

FSAR Section 9.2.7.1, states that the SCWS is not a safety-related or risk-significant system and has no system functions that support ESFs. The SCWS is not required to operate during or after a DBE. No systems cooled by the SCWS are safety-related.

The staff based its review of SCWS compliance with GDC 2 requirements, in part, on adherence to RG 1.29, Regulatory Position C.2. Based on its review of the FSAR, the staff understands that the SCWS is a nonsafety-related, seismic Category III designed system.

FSAR Table 1.9-3 indicates NuScale conformed with SRP Section 9.2.1, acceptance criterion II.1, which states that GDC 2 is applicable to the SCWS design. SDAA Table 1.9-3 states the following:

> The site cooling water system (SCWS) does not provide essential cooling to safety-related SSC and is not safety-related or augmented quality. The applicability of GDC 2 to the SCWS reviewed under this acceptance criterion is limited to aspects ensuring a failure of the nonsafety-related SCWS does not result in an adverse effect on a Seismic Category I SSC. For the design, this is provided by the design and construction of the nonsafety-related SCWS to meet the provisions of RG 1.29, Staff Regulatory Guidance C.2.

RG 1.29, Regulatory Position C.2, states the following:

> Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room, should be designed and constructed so that the SSE would not cause such failure. Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.

FSAR, Table 9.2.7-1, Note 4, states the following:

> [W]here SSC (or portions thereof) as determined in the as-built plant that are identified as Seismic Category III in this table could, as the result of a seismic event, adversely affect Seismic Category I SSC or result in incapacitating injury to occupants of the control room, they are categorized as Seismic Category II consistent with Section 3.2.1.2 and analyzed as described in Section 3.7.3.8.

The staff understands that Note 4 applies to all seismic Category III piping installed within a seismic Category I structure or with potential to impact Seismic Category I SSC's. Therefore, staff understands that failure of the SCWS will not challenge the supported systems ability to support ESF or emergency shutdown of any of the NPM units.

Based on the above FSAR statement and Note 4 to clarify seismic Category II SSCs to be consistent with RG 1.29, Regulatory Position C.2, the staff finds that the design of the SCWS contains adequate protection against failure and complies with the requirements of GDC 2.

9.2.7.4.2 GDC 4, "Environmental and Dynamic Effects Design Bases"

The staff based its review of SCWS compliance with the requirements of GDC 4 on the determination that failure of the SCWS, from pipe break or malfunction of the system, does not adversely affect any of the plant's essential systems or components (i.e. those necessary for safe shutdown or accident mitigation). FSAR Section 9.2.7, states that GDC 4 was considered in the design of the SCWS with regard to the ability to identify and isolate leaks that could impact safety-related equipment. Additional information regarding individual components tested and available in the SCWS can be found in Table 14.2-8. Based on Table 14.2-8 criteria, the staff finds the SCWS contains the ability for monitoring within MCR and the presence of isolation valves at various points throughout the system, as noted above. Therefore, the system contains design features that allow any sudden large leak to be identified and isolated promptly. In addition, failure due to pipe break will not impact support ESF or emergency shutdown of any of the NPM units.

Based on the monitoring and isolation features discussed above, the staff finds that the design of the SCWS complies with the requirements of GDC 4.

9.2.7.4.3 GDC 5, "Sharing of Structures, Systems, and Components"

The staff reviewed the design of the SCWS for compliance with the requirements of GDC 5 with respect to shared systems among NPMs. Compliance with GDC 5 requires that the nuclear power module designs include provisions to ensure that an event with one NPM does not adversely impact the ability of any other NPM units to perform their safety functions, including the ability to safely achieve and maintain safe shutdown. Meeting these requirements provides a level of assurance that the events will be isolated to one NPM. As indicated in FSAR Section 9.2.7.3, the components of the SCWS that are shared among units do not impair system's ability to perform their safety function, including ability to shutdown NPMs.

The staff finds that the design of the SCWS as described in the FSAR does not share components among modules and does not impair the ability of other systems to perform their safety functions. Therefore, the SCWS complies with the requirements of GDC 5.

9.2.7.4.4 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

GDC 60 requires the SCWS to be designed to control the release of radioactive material in liquid effluent, including operational occurrences, by preventing the inadvertent transfer of contaminated fluids to a noncontaminated drainage system for disposal.

The SCWS is normally a clean (non-radioactive) system that supplies cooling water to heat loads in the RXB, Central Utility Building, and the TGB. As indicated in Table 12.3-37, the possibility of radioactive contamination occurs when a heat exchanger leaks. Radioactive effluent release, through the SCWS, is minimized by maintaining the process fluid at a higher pressure than potentially contaminated interfacing systems. The SCWS outlet of the PCWS heat exchangers, and the RCCWS heat exchangers have radiation detectors to detect the presence of radiation in the SCWS and ensure that the operators are alerted to abnormal conditions so that action can be taken to isolate the affected section. Radiation monitoring, with sampling capability, is also available on the SCWS cooling tower blowdown line. As described in SDAA, Part 8, Section 3.9.1, the SCWS automatically responds to the SCWS high-radiation signal by closing the SCWS blowdown isolation valve to mitigate a release of radioactivity. In addition, SCWS drains within the isolated boundary are directed to the liquid radwaste system.

FSAR Table 11.5-4 "Effluent and Process Monitoring Off Normal Radiation Conditions" indicates SCWS response to radiation detection as follows: "Upon alarm, operators are alerted to abnormal condition, prompting them to investigate and isolate leaks or terminate other conditions that contribute to the off-normal conditions, through valve closures."

Based on its review, the staff finds the SCWS contains sufficient protection from an inadvertent transfer of radioactive fluid to the environment by use of monitoring design features and capabilities for the SCWS. Therefore, the staff finds that the SCWS complies with the requirements of GDC 60.

9.2.7.4.5 10 CFR 20.1406, "Minimization of Contamination"

The regulations in 10 CFR 20.1406 require, in part, that each standard design applicant describe how the design will minimize, to the extent practicable, contamination of the facility and the environment; facilitate eventual decommissioning; and minimize, to the extent practicable, the generation of radioactive waste. The SCWS provides cooling water to the tube side of heat exchangers in systems that contain, or could contain, radioactive material. As discussed above, the SCWS is designed to have a higher operating pressure than that in the interfacing systems and has monitoring features to detect the leakage of radionuclides into the system. In addition, the design includes provisions to safely drain isolated sections of the piping that could possibly become contaminated to the radioactive waste drain. Section 12.3 of this report contains additional information on the evaluation of the US460 design with regard to the minimization of contamination.

Based on the SCWS design features described in FSAR Section 9.2.7, the staff finds that the SCWS design and operation comply with the requirements of 10 CFR 20.1406.

9.2.7.5 Initial Test Program

The staff evaluates the ITP in Section 14.2 of this SER.

9.2.7.6 Technical Specifications

No GTS requirements are associated with the SCWS.

9.2.7.7 Conclusion

The staff evaluated the SCWS for the NuScale US460 design using the guidance of SRP Sections 9.2.1 and 9.2.5. Based on the above evaluation, the staff finds that the SCWS design meets GDC 2, 4, 5, and 60; 10 CFR 20.1406; and is consistent with RG 1.29, Regulatory Position C.2.

9.2.8 Chilled Water Systems

9.2.8.1 Introduction

The function of the CHWS is to provide, during plant normal operation, a heat sink for various air handling units (AHUs) and cooling loads in the radioactive waste processing systems. These AHUs include those in the normal CRVS, the radioactive waste building (RWB) HVAC system (RWBVS), and the reactor building HVAC system (RBVS). Other cooling loads include condensers in the LRWS and gas coolers in the gaseous radioactive waste system (GRWS).

9.2.8.2 Summary of Application

The applicant provided the design bases for and the description of the CHWS in FSAR Section 9.2.8. The major components of the CHWS include the main CHWS pumps, a standby CRVS pump, main CHWS chillers, a standby CRVS chiller, expansion tanks, and air separators. The CHWS provides cooling for the CRVS, RWBVS, and RBVS. Figure 9.2.8-1, “Chilled Water System Diagram,” shows a basic flow diagram for the CHWS.

9.2.8.3 Regulatory Basis

SRP Section 9.2.7, “Chilled Water System,” gives the relevant regulatory requirements for this area of review and the associated acceptance criteria, as summarized below, as well as the review interfaces with other SRP sections:

- GDC 1, as it relates to SSCs important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2, as it relates to the capabilities of the structures housing the system and the system itself to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of safety functions

- GDC 4, as it relates to the effects of missiles inside and outside containment, the effects of pipe whip and jets, environmental conditions from high and moderate energy line breaks, and the dynamic effects of flow instabilities and attendant loads (e.g., water hammer) during normal plant operation and upset or accident conditions

- GDC 5, as it relates to the sharing of equipment and components important to safety among multiple operating units at one single site, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units

- GDC 44, as it relates to the capability to transfer heat from SSCs important to safety to a heat sink during both normal and accident conditions, with suitable redundancy, assuming a single active component failure coincident with either the loss of offsite power or the loss of onsite power

- GDC 45, as it relates to design provisions for appropriate periodic inspection of important components to ensure the integrity and capability of the system

- GDC 46, as it relates to design provisions for pressure and operational functional testing of cooling water systems and components

- 10 CFR 20.1406(b), as it relates to the standard plant design and how facility design will minimize contamination of the facility and the environment, facilitate eventual decommissioning, and minimize the generation of radioactive waste

9.2.8.4 Technical Evaluation

The CHWS is not safety-related and is a closed loop cooling system that provides chilled water to the HVAC equipment chilled-water coils and other cooling loads in the radioactive waste processing systems. The CHWS consists of two subsystems: a primary system and a standby system. Specifically, the primary CHWS provides cooling for the normal CRVS, RWBVS, RBVS, and other equipment in the radioactive waste processing systems, while the standby CHWS is dedicated only to the CRVS in the event of a loss of normal AC power. FSAR Figure 9.2.8-1 shows the system configuration.

The primary CHWS consists of three chillers and three pumps, all piped in parallel and coupled together. Any of the three chillers can receive flow from any of the three variable-speed pumps. Chilled water flow varies throughout the evaporators of the operating chillers as well as through the HVAC cooling coils. The primary CHWS rejects heat via water-cooled chiller refrigeration units. Each chiller contains a condenser, evaporator, and associated piping and controls. The chiller condensers are supplied with cooling water from the SCWS. The staff evaluates the SCWS in Section 9.2.7 of this report.

The CRVS standby CHWS consists of an air-cooled chiller and one standby pump. These standby components operate when the backup power supply system (BPSS) is activated and the primary CHWS is unavailable to support the CRVS.

The major CHWS components are in the central utility building and the CRB. The CRVS, RWBVS, and RBVS chilled water-cooling coils are in the CRB, RWB, and RXB, respectively.

9.2.8.4.1 GDC 1, "Quality Standards and Records"

SRP Section 9.2.7 provides guidance for addressing GDC 1 requirements that are applicable to important to safety SSCs in the CHWS of an active PWR plant. The NuScale US460 passive design classifies the entire CHWS as not safety-related. The staff determined that the only important to safety functions performed by the CHWS are the prevention of adverse seismic system interactions and 10 CFR 20.1406(b) requirements related to minimizing contamination, as evaluated in the sections on GDC 2 and 10 CFR 20.1406 below. The staff concluded that the applicant has met the requirements of GDC 1 in a manner commensurate with the safety functions performed by the CHWS.

9.2.8.4.2 GDC 2, "Design Basis for Protection against Natural Phenomena"

The staff reviewed the CHWS for compliance with the requirements of GDC 2 with respect to its design for protection against the effect of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods. GDC 2 establishes requirements with respect to the CHWS design against protection from the effects of natural phenomena without loss of capability to perform any safety function. Compliance with the requirements of GDC 2 is based, in part, on conforming to RG 1.29, Regulatory Positions C.1.a through C.1.h, for seismic classification of safety-related SSCs and Regulatory Position C.2, for any nonsafety-related component whose failure during a seismic event could potentially affect the performance of safety-related SSCs.

FSAR Table 9.2.8-1, provides the component safety classifications, seismic category, applicable codes and standards, and locations of the SSCs. All CHWS components are designated as non-safety and seismic Category III (nonseismic) with a footnote stating that if these components in the as-built plant, as the result of a seismic event, are determined to adversely affect seismic Category I SSCs or result in incapacitating injury to occupants of the control room, they are categorized as seismic Category II consistent with FSAR Section 3.2.1.2 and analyzed as described in FSAR Section 3.7.3.8. Because of this footnote, the staff finds that the applicant's position on this subject is acceptable and that the CHWS meets the requirements of GDC 2 because the design has acceptable seismic classifications that are in accordance with RG 1.29.

9.2.8.4.3 GDC 4, "Environmental and Dynamic Effects Design Bases"

SRP Section 9.2.7 provides guidance for addressing GDC 4 requirements that are applicable to safety-related SSCs in the CHWS of an active PWR plant. The NuScale US460 passive design classifies the entire CHWS as not safety-related. Therefore, the applicant did not specifically address GDC 4 requirements for the CHWS. Because the CHWS does not perform any important to safety function other than preventing adverse seismic interactions with seismic Category I SSCs (as discussed in GDC 2 relating to RG 1.29, Regulatory Position C.2), the staff finds the applicant's position on this subject acceptable.

9.2.8.4.4 GDC 5, "Sharing of Structures, Systems, and Components"

The staff reviewed the CHWS for compliance with the requirements of GDC 5, which specifies the following:

> Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

As stated in FSAR Section 9.2.8.3, operation of the CHWS does not interfere with the ability to operate or shut down a module.

The staff verified from Figure 9.2.8-1 of FSAR that CHWS provides cooling to HVAC systems and to radioactive waste systems and does not provide cooling to individual nuclear power modules. Based on the above FSAR description, the staff finds that GDC 5 is satisfied.

9.2.8.4.5 GDC 44, "Cooling Water"

SRP Section 9.2.7 provides guidance for addressing GDC 44 requirements that are applicable to safety-related SSCs in an active PWR plant. In the NuScale US460 passive design, the CHWS does not support any safety-related SSCs under normal and accident conditions, and the entire system is classified as not safety-related. Additionally, there is no important to safety function associated with providing cooling water. The staff determined that since the CHWS performs no important to safety cooling water functions, the provisions of GDC 44 do not apply to this system.

9.2.8.4.6 *GDC 45, "Inspection of Cooling Water System," and GDC 46, "Testing of Cooling Water System"*

SRP Section 9.2.7 provides guidance for addressing GDC 45 and 46 requirements that are applicable to important SSCs in a chilled water system of an active PWR plant. The NuScale US460 passive design classifies the entire CHWS as not safety-related. In the NuScale US460 passive design, the CHWS serves no safety-related functions, is not risk-significant, is not credited for mitigation of DBAs and has no safe shutdown functions. The CHWS does not provide cooling to safety-related or risk-significant components.

Because the CHWS is not safety-related and has no cooling function important to safety, the staff finds that GDC 45 and 46 are not applicable to the CHWS.

9.2.8.4.7 *10 CFR 20.1406, "Minimization of Contamination"*

The regulations in 10 CFR 20.1406 require that applicants for standard plant designs describe how the facility design will minimize contamination of the facility and the environment and the generation of radioactive waste. The CHWS is designed to be a closed loop, non-radioactive system. The design of the CHWS provides protection against the spread of contamination in accordance with 10 CFR 20.1406, as discussed in FSAR Section 12.3.

As described in FSAR Table 12.3-17, contamination is minimized by having CHWS pressure higher than the LRWS and GRWS pressures at the heat exchanger. To avoid the possibility of the CHWS being at a lower pressure than LRWS, the CHWS is isolated from the LRWS skid if the pressure difference is less than the setpoint. The LRWS and GRWS are the only radioactive systems that interface with the CHWS. Section 12.3 of this report contains additional information on the evaluation of the US460 design with regard to the minimization of contamination

FSAR Table 1.9-3, states, "The CHWS is at a higher pressure than the liquid radioactive waste system and gaseous radioactive waste system where the systems interface, precluding introduction of radioactive contaminants into the CHWS."

Because the CHWS has design features described in FSAR to protect against contamination entering the system, the staff finds that the CHWS complies with the requirements of 10 CFR 20.1406.

9.2.8.5 *Initial Test Program*

FSAR Table 14.2-5, "Test #05 Chilled Water System Test," describes the system performance testing of the CHWS. The ITP is discussed further in Section 14.2 of this report.

9.2.8.6 *Technical Specifications*

No GTS requirements are associated with the CHWS.

9.2.8.7 *Conclusion*

The staff evaluated the CHWS for the NuScale US460 design in accordance with the guidance of SRP Section 9.2.7. The staff finds that the CHWS design meets the requirements of GDC 1, 2, 4, and 5 and 10 CFR 20.1406 and conforms to RG 1.29, Position C.2.

9.2.9 Utility Water Systems

9.2.9.1 Introduction

This section describes the staff's review of the UWS. The UWS provides raw water and clarified water to the fire water tank, DWS, PWS, RXB, CRB, annex building, RWB, turbine building, central utility building, and other plant users. The source of water for the UWS and the required chemical treatment is site-specific. The water supplied by the UWS does not provide cooling functions.

9.2.9.2 Summary of Application

FSAR Section 9.2.9, "Utility Water System," provides very limited information on the system. The UWS comprises raw water pumps, a utility water treatment skid, a utility water storage tank, and utility water supply pumps. The applicant indicated that UWS serves no safety-related or risk-significant functions, is not credited for mitigation of DBAs and has no safe shutdown functions.

9.2.9.3 Regulatory Basis

SRP Section 9.2.4 gives the relevant regulatory requirements for this area of review and the associated acceptance criteria, as well as review interfaces with other SRP sections.

- GDC 5, as it relates to the capability of shared systems and components to perform required safety functions.
- GDC 60, as it relates to the UWS design, includes provisions to suitably control the release of radioactive materials in gaseous and liquid effluents during normal operation, including anticipated operational occurrences.
- GDC 64, as it relates to the UWS design for monitoring releases of radioactive materials to the environment during normal operation, including anticipated operational occurrences.
- 10 CFR 20.1406, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive waste.

9.2.9.4 Technical Evaluation

The utility water system (UWS) provides distribution of raw water and clarified water for various plant uses and provides the single point liquid effluent release to the environment. The UWS serves no safety-related or risk-significant functions, is not credited for mitigation of design-basis accidents, and has no safe shutdown functions.

The UWS comprises raw water pumps, a utility water treatment skid, a utility water storage tank, and utility water supply pumps. Water from the UWS supplies maintenance activities such as general wash downs in areas including the Reactor Building, the Radioactive Waste Building, and the Turbine Generator Building.

The staff reviewed the UWS design for compliance with the regulatory basis given in Section 9.2.9.3 of this SER. The following sections give the results of the staff's review

9.2.9.4.1 GDC 5, "Sharing of Structures, Systems, and Components"

The applicant stated that the UWS serves no safety-related functions, is not credited for the mitigation of DBAs, and has no safe-shutdown functions. The UWS is not required to function during or after a natural phenomenon event or other events that result in the generation of missiles, pipe whip, or fluid discharge. Portions of the system that are in proximity to seismic Category I SSCs are designed to seismic Category II standards. FSAR, Table 9.2.9-1, classifies all UWS components as seismic Category III. The applicant's note at the end of Table 9.2.9-1, states the following:

> Where SSC (or portions thereof) as determined in the as-built plant that are identified as Seismic Category III in this table could, as the result of a seismic event, adversely affect Seismic Category I SSC or result in incapacitating injury to occupants of the control room, they are categorized as Seismic Category II, consistent with Section 3.2.1.2, and analyzed as described in Section 3.7.3.8.

FSAR, Table 9.2.9-1, also provides the quality group classification of UWS components and equipment. The applicant stated that the portion of the UWS that receives radioactive water and discharges it to the environment is classified Quality Group D. The staff finds that the UWS is appropriately classified because the classification follows the guidance in RG 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants," in that Quality Group D should be applied to water- and steam-containing components that are not part of the reactor coolant pressure boundary (RCPB), but are part of systems or portions of systems that contain or may contain radioactive material.

The applicant stated that GDC 5 was considered in the design of the UWS. The UWS pump and storage tanks are shared by the NPMs. The UWS has no safe-shutdown functions that are shared between NPMs. The applicant indicated the UWS has no functions that are impacted if there is an accident in one module coincident with the shutdown and cooldown of the remaining modules. Because the failure of the UWS does not affect the functional performance of safety-related systems, the staff finds that the design complies with GDC 5 and is therefore acceptable.

9.2.9.4.2 GDC 60, "Control of Releases of Radioactive Materials to the Environment" and GDC 64, "Monitoring Radioactivity Releases"

The applicant stated that the UWS provides a single-point, liquid effluent release to the environment. FSAR, Section 11.2 describes the treatment of the "Liquid Waste Management System". Treated effluent is either recycled for use within the plant or discharged to the environment through the UWS. The UWS includes an off-line radiation monitor in the discharge line to the environment with the capability to take samples that are representative of the liquid effluent stream. As indicated in FSAR Table 11.5-4, UWS provides an alarm in the MCR and locally. The alarm alerts the operators to abnormal conditions and the need to isolate the source. Because the UWS monitors the UWS discharge path for radiation, provides for an alarm in the MCR and waste management control room, and allows for isolation if required, the staff finds that the UWS complies with the requirements of GDC 60 and 64.

9.2.9.4.3 10 CFR 20.1406, "Minimization of Contamination"

The regulations in 10 CFR 20.1406 require, in part, that each standard design applicant describe how the facility design will minimize, to the extent practicable, contamination of the facility and environment and the generation of radioactive waste. FSAR, Table 12.3-40, lists the design features specific to the UWS for the minimization of contamination, such as the use of corrosion-resistant materials that are compatible with operating conditions and radiation monitors. In addition, the applicant stated that UWS components are selected with 60-year design life.

The UWS is the single-point, liquid effluent release path to the environment, and it is sampled and monitored for radiation. An off-line radiation monitor provides continuous indication of effluent parameters. The raw water pumps and supply pumps are provided with redundant pumps to allow for one pump to be taken out of service for maintenance. Each pump has an upstream and downstream isolation valve to enable separation from the system for maintenance and/or replacement.

The staff reviewed FSAR Sections 9.2.9 and 12.3 as related to the prevention and minimization of contamination. Because the NuScale US460 FSAR describes adequate measures for radioactive leak detection and controls in the UWS design to minimize contamination, as summarized above, the staff concludes that the system as described in the FSAR complies with 10 CFR 20.1406.

9.2.9.5 Initial Test Program

The staff evaluates the ITP in Section 14.2 of this SER.

9.2.9.6 Conclusion

Based on the review above, the staff concludes that the UWS for the NuScale US460 design satisfies the relevant requirements for the UWS as described in Section 9.2.9.3 of this SER.

9.3 <u>Process Auxiliaries</u>

9.3.1 Compressed Air Systems

9.3.1.1 Introduction

The compressed air system (CAS) consists of the instrument and control air system (IAS), service air system (SAS), and the nitrogen distribution system (NDS). The CAS is not a safety-related system and is designed such that a failure of any component or the loss of a compressed air source will not prevent any system, subsystem, or device from performing its safety functions.

9.3.1.2 Summary of Application

FSAR, Section 9.3.1, "Compressed Air System," provides the CAS description and operation, including design bases, instrumentation, and the inspection and testing program.

9.3.1.3 Regulatory Basis

SRP Section 9.3.1, Revision 2, “Compressed Air System,” issued March 2007, gives the relevant regulatory requirements for this area of review and the associated acceptance criteria, summarized below, as well as the review interfaces with other SRP sections:

- GDC 1, as it relates to important-to-safety SSCs designed, fabricated, and tested to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2, as it relates to important-to-safety SSCs being designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions

- GDC 5, as it relates to the sharing of important-to-safety SSCs among nuclear power units

- 10 CFR 50.63, “Loss of all alternating current power,” as it relates to the ability of a plant to withstand for a specified duration and recover from a station blackout (SBO)

9.3.1.4 Technical Evaluation

The staff reviewed FSAR, Section 9.3.1, in accordance with SRP Section 9.3.1 to ensure compliance with the regulatory requirements listed in Section 9.3.1.3 of this report. The applicant stated that the CAS is composed of the IAS, SAS, and the NDS, which are not safety-related or risk-significant systems and are not required to provide compressed air or nitrogen to actuate or control equipment that requires supplied compressed air or nitrogen to perform safety-related functions during normal operations, transients, or accidents. The applicant also stated that compressed air is not required during a loss of offsite power or SBO to achieve safe shutdown, including the closing of containment isolation valves. The CAS also does not support safety-related functions for maintaining the ability to actuate or control equipment necessary for core cooling and decay heat removal or for maintaining containment integrity following an SBO. Nevertheless, the IAS is designed in compliance with the criteria specified in ANSI/Instrument Society of America S7.3-R1981, “Quality Standard for Instrument Air,” for minimum instrument air quality standards.

The applicant also considered the requirements of GDC 2 in the design such that portions of the CAS in which failure caused by an SSE could reduce the functioning of a seismic Category I SSC to an unacceptable safety level or could result in incapacitating injury to occupants of the control room are designed and constructed to preclude such failure. These SSCs are classified as seismic Category II, as stated in FSAR, Section 3.2.1.2, and conform to the design guidance of RG 1.29 to ensure that there are no deleterious interactions with a seismic Category I SSC.

The applicant also considered the requirement of GDC 5 in the design of the CAS because there is no compromise in the ability of systems and components to perform their safety-related functions for each NPM regardless of CAS equipment failures or other events that may occur in other NPMs. Furthermore, unacceptable effects of equipment failures or other events occurring in a particular NPM will not propagate to unaffected NPMs.

Because the CAS is not credited for coping with or recovering from an SBO condition, the staff has determined that the requirements of 10 CFR 50.63 and the guidance of RG 1.155, “Station

Blackout,” issued August 1988, on the plant’s ability to withstand for a specified duration and recover from an SBO are not applicable.

Based on the above, the staff determines that the licensee met the design requirements of GDC 1, 2, and 5 and provided the appropriate design features and preoperational tests to find that the design commitments are met, and the as-built plant conditions will operate in accordance with the standard design.

9.3.1.5 Initial Test Program

FSAR, Section 14 Tables 14.2-12 “Test # 12 Nitrogen Distribution System,” Table 14.2-13 “Test # 13 Service Air System,” and Table 14.2-14 “Test # 14 Instrument Air System” describes the proposed tests for the subsystems that belong to the CAS. The staff evaluates the ITP in Section 14.2 of this report.

9.3.1.6 Technical Specifications

No GTS requirements are associated with the CAS.

9.3.1.7 Conclusion

The CAS, which includes the IAS, SAS, and NDS, is not credited for maintaining the ability to actuate or control equipment necessary for core cooling and decay heat removal or maintaining containment integrity following an SBO. Furthermore, the design of the CAS considers the requirements of GDC 1, 2, and 5, and the applicant provided appropriate preoperational tests. Therefore, the staff concludes that the CAS will operate as designed and its post-accident failures will not impact the performance of any safety-related equipment in the plant.

9.3.2 Process Sampling Systems

9.3.2.1 Introduction

The process sampling system (PSS) allows the plant staff to obtain liquid and gaseous samples and determine their chemical and radiochemical conditions by measurement and analysis. Centralized and local facilities permit samples of primary and secondary process streams and components to be taken. The PSS collects representative liquid and gaseous samples from various plant systems using the following systems:

- the primary sampling system
- the containment sampling system (CSS)
- the secondary sampling system (SSS)

9.3.2.2 Summary of Application

FSAR, Section 9.3.2, “Process Sampling System,” gives the PSS description and operation. The PSS serves no safety-related functions, is not credited for mitigation of DBAs, has no safe-shutdown functions, and is not credited to maintain the integrity of the RCPB. The PSS is operable during normal operations, including at power, shutdown, and startup. The main components in the PSS are sample coolers, sample panels, a return pump, and remotely operated sample line isolation valves.

The system has the ability to obtain samples at the normal system operating temperatures and pressures from various locations. The PSS obtains samples that are representative of the process under evaluation. For sampling of liquid process streams, sample points are located in a turbulent flow zone, which minimizes particulate dropout and re-entrainment in sample piping. For sampling of tanks, the sample points are located in the tank recirculation loop to ensure sediments or solid particulates are distributed uniformly in the fluid mixture. The PSS design criteria ensure representative samples from gaseous process streams and tanks are in accordance with ANSI/Health Physics Society N13.1-2011, "Sampling and Monitoring Releases of Airborne Radioactive Substances From the Stacks and Ducts of Nuclear Facilities."

FSAR Tables 9.3.2-1 through 9.3.2-4 give detailed descriptions of all the sample points and the type of sampling done at each point.

For normal sampling at power, the primary sampling system performs continuous and semi-continuous sampling and analysis of reactor coolant discharge from the reactor coolant system (RCS). Personnel collect grab samples from various sample locations in the CVCS process loop.

During normal operation, the containment sampling system monitors gas discharged from the containment evacuation system for hydrogen and oxygen gas concentration. Normal operation of the secondary sampling system includes continuous monitoring of the condensate pump discharge, condensate polisher effluents, feedwater, and main steam. The frequency for sample collection and required analyses for local process sample points are addressed in the primary, secondary, and ancillary chemistry program and procedures.

Initial Test Program: FSAR Section 14.2 (Test # 46) gives information regarding PSS initial testing. The staff evaluates the ITP in Section 14.2 of this SER.

9.3.2.3 Regulatory Basis

SRP Section 9.3.2, "Process and Post-Accident Sampling Systems" provides staff review guidance for the PSS. The following are the relevant regulations for this area of review:

- GDC 1, as it relates to the design of the PSS and components in accordance with standards commensurate with the importance of their safety functions.
- GDC 2, as it relates to the ability of the PSS to withstand the effects of natural phenomena.
- GDC 13, "Instrumentation and Control," as it relates to monitoring variables that can affect the fission process, the integrity of the reactor core, and the RCPB.
- GDC 14, "Reactor Coolant Pressure Boundary," as it relates to testing of the RCPB by sampling for chemical species that affect the RCPB.
- GDC 26, "Reactivity Control System Redundancy and Capability," as it relates to the capability to reliably control the rate of reactivity changes by sampling boron concentration.
- GDC 60, as it relates to the capability of the PSS to control the release of radioactive materials to the environment.

- GDC 63, as it relates to detecting conditions that may result in excessive radiation in the fuel storage and radioactive waste systems.

- GDC 64, as it relates to monitoring the containment atmosphere and plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences.

- Three Mile Island (TMI) Action Plan Item III.D.1.1 in NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980 (as amended by SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," dated April 2, 1993), which relates to leakage of radioactive material out of containment through sampling points.

- 10 CFR 20.1101(b), as it relates to using engineering controls to keep doses to workers and the public ALARA

As noted in SRP Section 9.3.2, the applicant should demonstrate sampling of the sites mentioned under item 1 of the "SRP Acceptance Criteria." Also, sampling procedures should be consistent with the guidelines of RG 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," specifically, Regulatory Positions C.2, C.6, and C.7; EPRI 3002000505, "Pressurized Water Reactor Primary Water Chemistry Guidelines"; and EPRI 1016555, "Pressurized Water Reactor Secondary Water Chemistry Guidelines." SER Section 9.3.4 and Chapter 10 discuss in more detail the use and acceptability of these EPRI guidelines in the NuScale US460 design.

9.3.2.4 Technical Evaluation

The staff reviewed the PSS in accordance with the review procedure in SRP 9.3.2. The sections that follow give the results of the staff's review.

9.3.2.4.1 GDC 1, "Quality Group Classifications and Standards"

The staff based its review of PSS compliance with GDC 1 requirements on adherence to RG 1.26, Regulatory Position C.1, C.2, and C.3. Based on its review of the NuScale FSAR, the staff understands that the PSS is a non-safety-related, seismic Category III designed system. FSAR Table 1.9-3 indicates the NuScale PSS design conforms with SRP Section 9.3.2 (acceptance criterion II.4), which refers to GDC 1 requirements applicable to the PSS design.

According to the applicant, the PSS SSCs would be designed, fabricated, erected, and tested to appropriate quality standards such that their failure does not impact the function of safety-related or risk-significant systems. The SSCs in the PSS are designed to Quality Group D standards, in accordance with Regulatory Guide 1.26 and the PSS piping conforms to ASME B31.1, "Power Piping."

The staff determines that the proposed PSS meets the quality standards requirements of GDC 1 because the applicant has designed the sampling lines and components of the PSS to conform to the classification of the system to which each sampling line and component is connected, in accordance with RG 1.26, Regulatory Positions C.1, C.2, and C.3.

9.3.2.4.2 GDC 2, "Design Bases for Protection against Natural Phenomena"

The staff based its review of PSS compliance with GDC 2 requirements on adherence to RG 1.29, Regulatory Position C.2. Based on its review of the NuScale FSAR, the staff understands that the PSS is a non-safety-related, seismic Category III designed system. FSAR Table 1.9-3 indicates the NuScale US460 PSS design conforms with SRP 9.3.2 (acceptance criteria II.4), which refers to GDC 2 requirements.

RG 1.29, Regulatory Position C.2, indicates the following:

> Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level or could result in incapacitating injury to occupants of the control room, should be designed and constructed so that the SSE would not cause such failure. Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.

According to the applicant, the primary sampling system and containment sampling system components are located in the RXB and are protected from earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches to the extent that the RXB is protected from such events. The PSS does not connect to seismic Category I piping.

The PSS does not employ sample lines that penetrate the CNV and the RPV; therefore, there is no containment isolation function associated with the system. There is no physical interaction of PSS SSCs with safety-related SSCs. PSS failure does not adversely affect the integrity of safety-related systems. designed and constructed so that the SSE would not cause such failure

The staff finds that the guidance of RG 1.29, Regulatory Position C.2, for non-safety-related portions has been appropriately followed by the applicant and therefore concludes that the PSS complies with the requirements of GDC 2.

9.3.2.4.3 GDC 13, "Instrumentation and Controls"

The staff has determined that the proposed PSS meets the requirements of GDC 13 to monitor variables that can affect the fission process, the integrity of the reactor core, and the RCPB during normal operation by providing the capability to sample the reactor coolant system and associated auxiliary systems.

9.3.2.4.4 GDC 14, "Reactor Coolant Pressure Boundary"

The staff determined that the proposed PSS meets the requirements of GDC 14 to monitor variables that can affect the RCPB and to ensure a low probability of abnormal leakage, rapidly propagating failure, and gross rupture, respectively, by providing the capability to sample and analyze reactor system coolant samples to verify that key parameters, such as chloride, hydrogen, and oxygen concentrations, are within prescribed limits and that impurities are properly controlled, ensuring many mechanisms for corrosive attack are mitigated and do not adversely affect the RCPB.

9.3.2.4.5 GDC 26, "Reactivity Control System Redundancy and Capability"

The staff determined that the proposed PSS meets the requirements of GDC 26 to control the rate of reactivity changes by providing the capability to sample the reactor coolant and the contents of the storage tanks in the boron addition system (BAS), which allows the verification of the boron concentration necessary for the control of core reactivity changes.

9.3.2.4.6 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

GDC 60 requires that the nuclear power module design include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including AOOs.

According to FSAR Section 9.3.2.3, systems that have effluent release paths to the environment have local grab sample points, permitting effluent sample analysis before release. The PSS routes samples back to the system of origin or to the applicable radioactive waste system as appropriate to control the release of radioactive material.

The PSS design limits the potential reactor coolant loss from the rupture of a sample line. A failure of a sample line results in a loss of flow to either a continuous analyzer or a grab sample panel that can be detected via instrument indication. In addition, a break in a sample line results in activity release that might actuate the fixed area radiation monitors located in the containment sampling system equipment area and the primary sampling system equipment area, as described in FSAR Section 12.3, "Radiation Protection Design Features." The CVCS sample lines include isolation valves that fail closed to control the potential release of radioactive materials to the environment. These isolation valves are downstream of the CVCS discharge line containment isolation valves and the CVCS module isolation valves. The small PSS line sizes restrict the break flow of a sample line outside containment.

Based on the design of the PSS, the staff determined that the PSS meets the requirements of GDC 60 to control the release of radioactive materials to the environment by (1) purging and draining sample streams back to the system of origin or to an appropriate liquid radioactive waste system (LRWS) and (2) providing either redundant isolation valves that fail in the closed position or passive flow restrictions in the sampling lines.

9.3.2.4.7 GDC 63, "Monitoring Fuel and Waste Storage"

GDC 63 requires that appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

According to FSAR Section 9.2.3.3, the PCWS includes the grab sampling capability of the SFP and reactor pool water. The LRWS and the GRWS include local sample points that enable analyses to detect conditions in the fuel storage and radioactive waste systems that could result in excessive radiation levels and excessive personnel exposure.

Based on the design of the PSS, the staff determined that the PSS meets the requirements of GDC 63 to detect conditions that may result in excessive radiation levels in fuel storage and the LRWS by providing the capability to sample the SFP water, reactor pool water, LRWS, and GRWS for radioactivity.

9.3.2.4.8 GDC 64, "Monitoring Radioactivity Releases"

The staff determined that the proposed PSS meets the requirements of GDC 64 to monitor for radioactivity that may be released during normal operations, including anticipated operational occurrences, by providing those monitors and sampling capabilities to determine the radiological conditions of plants systems.

9.3.2.4.9 10 CFR 20.1101, "Radiation Protection Programs"

The staff determined that the proposed PSS meets the requirements of 10 CFR 20.1101(b) to keep radiation exposures ALARA because design features are included to ensure that doses associated with sampling are ALARA during normal operation (as discussed below in the radiation protection evaluation subsection).

9.3.2.4.10 TMI ACTION PLAN ITEM III.D.1.1 IN NUREG-0737, "CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS"

The staff determined that the proposed PSS meets the requirements of 10 CFR 50.34(f)(2)(xxvi) (Item III.D.1.1 in NUREG-0737) to include provisions for leakage control and detection to levels as low as practical to prevent exposures to workers and the public (as discussed below in the radiation protection evaluation subsection). The remaining programmatic requirements of 10 CFR 50.34(f)(2)(xxvi) will be fulfilled at the COL stage by including applicable portions of the systems in a leakage control program that provides for periodic leak testing and measures to minimize the leakage from the systems (COL Item 9.3-1).

Post-accident Sampling 10 CFR 50.34(f)(2)(viii) Exemption Request

In SDAA Part 7, Section 16, NuScale provided rationale to support that an exemption request from 10 CFR 50.34(f)(2)(viii) would be justified. NuScale stated that approval of the exemption request would eliminate the requirement to provide a design capability to obtain samples as described in FSAR Section 9.3.2. The justification for the exemption request is that circumstances necessitating such a sample are unexpected and of low probability, and there are other indications available for the necessary parameters.

Regulatory Requirements

- 10 CFR 50.34(f)(2)(viii) states the following:

> Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

- 10 CFR 52.7, "Specific Exemptions," 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 two conditions that must be met for granting an exemption are the following:

  > (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, "Licenses, Certifications, and Approvals for Nuclear Power Plants." 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.

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 has determined that the justification for the proposed exemption would be 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 DBE, and does not create new accident precursors. Therefore, as required by 10 CFR 50.12(a)(1), the staff has determined that the justification for exemption would pose no undue risk to public health and safety.

Consistent with Common Defense and Security

The requested exemption would 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 has determined that the common defense and security would not be impacted by this proposed 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 requirements in 10 CFR 50.34(f)(2)(viii), state that an applicant must provide a capability to promptly obtain and analyze samples from the RCS and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 50 mSv (5 rem) to the whole body or 500 mSv (50 rem) to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. The underlying purpose of this rule is to ensure that operators are safely able to evaluate indicators of core damage during accident conditions. The following includes the staff's determination on the different aspects of 10 CFR 50.34(f)(2)(viii) for which NuScale has provided rational to support a proposed exemption.

<u>Radionuclides</u>: To determine the potential need to sample for radionuclides, the staff evaluated the applicant's design, which includes radiation monitors under the bioshield and core exit thermocouples that can be used to assess core damage. The staff notes that in accordance with 10 CFR 50.47(b)(9), the COL applicant must address that they have adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition in their site-specific application. However, this regulation does not specifically require a post-accident sampling capability, and the guidance in Section II.I of NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," references only a post-accident sampling capability as a means of meeting the requirements of 10 CFR 50.47(b)(9). The staff has determined that a post-accident sampling capability would not be necessary for sampling radionuclides for the NuScale US460 design based on the use of radiation monitors to assess core damage.

<u>Hydrogen in the Containment Atmosphere</u>: The purpose of post-accident sampling hydrogen in the atmosphere is to ensure that hydrogen and oxygen concentrations do not support combustion that could challenge the containment. For the NuScale US460 design, sampling of containment hydrogen and oxygen is unnecessary to ensure containment integrity and precludes a combustible atmosphere (less than 4 percent) following a beyond design basis event by using passive autocatalytic recombiners (PAR) to limit the oxygen concentration.

<u>Dissolved Gases</u>: Sampling for dissolved gases has generally been required to ensure that natural recirculation is not inhibited. This is not necessary for NuScale because the accumulation of non-condensable gases interfering with post-accident natural circulation is insusceptible due design considerations of the NuScale US460 emergency core cooling system (ECCS), as documented in the staff's evaluation of the SDAA Part 7, Section 1 for exemption for high point vents in Section 5.4.4 of this report.

Chlorides: The purpose of sampling reactor coolant for chlorides is to monitor chlorides concentration, which can induce stress corrosion cracking, pitting, and crevice corrosion of stainless-steel components exposed to reactor coolant. These corrosion mechanisms are dependent on the material, pH, temperature, and chloride concentration. The NuScale US460 design limits chloride sources by design and operation, such as containment cleanliness requirements and minimal use of chlorinated cable insulation. The NuScale US460 design also limits chloride by monitoring and control of reactor water chemistry based on industry guidelines in the EPRI Guidelines. The staff finds it would be acceptable for NuScale to not perform post-accident chloride sampling because of the minimal use of chlorinated cable insulation and the monitoring of chlorine concentration using the EPRI Guidelines during normal operation.

Boron Concentrations: The purpose of sampling the boron concentration of the RCS is to ensure that there is adequate shutdown margin to achieve and maintain safe shutdown. The only Type B variables identified in the NuScale US460 design that provide direct indication and are used to assess the process of accomplishing or maintaining reactivity control are neutron flux and core inlet and exit temperatures. The transient and accident analyses described in FSAR Chapter 15, does not rely on the measurement of RCS boron concentration and are not expected to be necessary to implement the plant operating procedures and maintain the plant critical safety functions for transients within the scope of the Chapter 15 safety analyses. Therefore, the staff has determined that a post-accident boron sample would not be necessary for the NuScale US460 design.

The staff notes that the justification for the proposed exemption stated that the capability to ascertain the RCS boron concentration is an important long-term issue when water, other than the original reactor coolant inventory, will be used to refill the reactor vessel or to flood the containment. While this statement is correct, RCS boron concentration is also important in post-event recovery actions when exiting passive ECCS and DHRS cooling modes, as described in FSAR Sections 4.3.1.5 and 4.3.2.1, and needs to be accounted for to ensure shutdown margin limits are preserved. The staff notes that these post-event recovery actions are outside the scope of the FSAR review but are important to capture in the development of operating procedures. NuScale included COL Item 13.5-3 in FSAR Section 13.5.2, “Operating and Maintenance Procedures,” for development of operating procedures at a future licensing stage.

Based on the above, the staff has determined that application of the regulation in these particular circumstances is not necessary to achieve the underlying purpose of the rule because the NuScale design includes alternative instrumentation that can provide the necessary information to inform operators for accident management. Therefore, the special circumstances in 10 CFR 50.12(a)(2)(ii) would be met.

In SDAA Part 7, NuScale stated that special circumstances described in the 10 CFR 50.12(a)(2)(iv) associated benefits 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) would not be necessary for the exemption to be granted. Because the staff has determined that special circumstances would not be present in accordance with 10 CFR 50.12(a)(2)(ii), the staff would make no finding regarding the presence of special circumstances described in 10 CFR 50.12(a)(2)(iv).

Exemption request evaluation conclusion

Based on the evaluation above, the staff finds that the applicant's design would meet the requirements for an exemption under 10 CFR 50.12(a). The staff considered NuScale's exemption request and determined that this exemption, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

The above language does not exempt a COL applicant proposing to use the NuScale design from complying with the emergency preparedness planning standard in 10 CFR 50.47(b)(9). A COL applicant proposing to use the NuScale design would still need to identify what adequate methods, system, and equipment are available for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition.

Further, since the NuScale design would be exempt, under the conditions stated above, from 10 CFR 50.34(f)(2)(viii) and the applicant has removed all information from the FSAR indicating that post-accident samples will be taken, the staff did not assess the radiological dose consequences to a worker obtaining and analyzing RCS and containment atmosphere samples following any accident.

9.3.2.4.11 Radiation Protection Evaluation

The staff reviewed FSAR Section 9.3.2, and supporting FSAR sections to ensure that the radiological aspects of the PSS design and the associated COL information item are in accordance with the applicable regulatory requirements. These include 10 CFR 20.1101(b); 10 CFR 20.1406; 10 CFR Part 50, Appendix A, GDC 60, 63, and 64; 10 CFR 50.34(f)(2)(viii); and 10 CFR 50.34(f)(2)(xxvi).

As described in NuScale FSAR Section 9.3.2 and Table 9.3.2-1, "Primary Sampling System Normal Sample Points," primary reactor coolant samples are normally collected in the CVCS gallery in the RXB. Continuous samples are collected from sample lines coming off the CVCS letdown line. Grab samples can also be taken at this point, as well as downstream of the CVCS purification equipment, and at the CVCS injection line to the RCS. Grab sample stations for primary fluid are provided with a vent hood to minimize personnel exposure to radioactive fluids. The vent hood exhaust is connected to the ventilation duct of the RXB HVAC system. The staff has determined that since the vent hood is designed to control airborne radioactivity to reduce worker intake and the spread of radioactive material to the plant environment, it is an appropriate design feature to ensure that doses to workers are ALARA, in accordance with 10 CFR 20.1101(b). Grab samples are analyzed in the counting room.

FSAR Section 9.3.2.2.1, states that the secondary sampling system provides a means for monitoring and collecting fluid samples in the steam cycle systems. This includes grab samples, as specified in FSAR Table 9.3.2-3, "Secondary Sampling System Normal Sample Points." While the secondary sampling panel does not include a vent hood, the concentration of radioactive materials and chemicals in the secondary system is expected to be maintained low enough such that the use of personal protective equipment (e.g., gloves, safety glasses) is sufficient for worker safety.

In addition, FSAR Section 9.3.2.2.2, specifies that while primary to secondary leakage is a potential concern for contamination of the secondary system, the process radiation monitors located on the main steam lines and the radiation monitors on the air-cooled condenser system provide the capabilities of detecting primary to secondary leakage and alerting the operators to abnormal conditions and the need to take appropriate manual action. The staff notes that respiratory protection could be worn to limit the dose to a worker taking secondary samples, if necessary. The staff finds the radiation monitors and the use of procedural controls sufficient to provide reasonable assurance that the doses to workers taking secondary samples will be within the requirements of 10 CFR Part 20, “Standards for Protection against Radiation,” including the ALARA requirements, and are acceptable.

Various other local sample points are provided throughout the plant, as stated in the FSAR Section 9.3.2 tables and FSAR Table 11.5-2, “Provisions for Sampling Gaseous Process and Effluent Streams” and Table 11.5-3, “Provisions for Sampling Liquid Process and Effluent Streams.” These include local sample points for the LRWS, GRWS, solid radioactive waste system, reactor and SFP, and others. These sampling points are consistent with SRP Section 9.3.2 and are, therefore, acceptable.

FSAR Section 9.3.2 and Chapters 11 and 12, provide several examples of features that are included in the NuScale US460 design to limit radiation exposure to workers and members of the public and minimize contamination in accordance with 10 CFR 20.1101(b) and 10 CFR 20.1406. These include the following:

- vent hoods, where appropriate, to limit personnel exposure to radioactive gases
- locating sample coolers, isolation valves, and associated piping in shielded compartments or away from sample panels to the extent practical
- providing sample panels located in the RXB with sloped floors to direct leakage or spills to the drain hubs leading to a radioactive waste drain system (RWDS) sump
- locating grab sample lines directly over sample sinks. Sample sinks servicing lines that would be anticipated to potentially contain radioactive material are routed to the RWDS.

The staff evaluated these design features and finds them to be consistent with the guidance of RG 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable,” and RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” and the associated requirements of 10 CFR 20.1101(b) and 10 CFR 20.1406. These design features were also found to provide assurance that plant sampling during normal operation will be conducted in accordance with 10 CFR Part 50, Appendix A, GDC 60, 63, and 64 because the applicant included CES monitors and sampling capabilities to determine the radiological conditions within the CNV. The applicant also satisfies GDC 64, in part, by providing other sampling points to allow plant staff to verify radiological conditions, as found in FSAR Tables 11.5-2 and 11.5-3. Staff notes that while this section discusses how the sampling systems are designed to meet GDCs 60, 63, and 64, other aspects of compliance with GDCs 60, 63, and 64 are addressed in other FSAR and SER sections, especially Chapters 11 and 12, which include information on the radiation monitoring systems.

Post-accident Conditions

As discussed above, NuScale provided rationale to support that an exemption from the post-accident sampling requirements of 10 CFR 50.34(f)(2)(viii) would be justified. Since post-accident sampling will not occur in the NuScale US460 design, the staff did not review the radiation dose consequences of collecting or analyzing post-accident samples.

However, the staff evaluated the sampling system to determine if it is an area necessary to permit access under 10 CFR 50.34(f)(2)(vii). The staff notes that while the sampling system includes hydrogen and oxygen monitors, in FSAR Part 7, NuScale has provided rationale to support an exemption from 10 CFR 50.34(c)(4) and 10 CFR 50.34(f)(2)(xvii)(C), which requires the capability for monitoring combustible gases during an accident. Based on this, there would be no requirements to access the sampling system for sampling or monitoring following an accident. Since post-accident sampling will not occur and since access to the sampling system would not be necessary, the NRC staff did not evaluate the sampling system as an area requiring post-accident access under 10 CFR 50.34(f)(2)(vii).

9.3.2.5 Initial Test Program

The staff evaluates the ITP in Section 14.2 of this SER.

9.3.2.6 Combined License Information Items

Table 9.3.2-1 lists relevant COL information item numbers and descriptions from FSAR Table 1.8-1.

Table 9.3.2-1 NuScale COL Information Items for Section 9.3.2

COL Item No.	Description	FSAR Section
9.3-1	An applicant that references the NuScale Power Plant US460 standard design will submit a leakage control program for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. The leakage control program will include an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems to as low as practical.	9.3

As described above, COL Item 9.3-1 identifies operational procedures and programs needed to address 10 CFR 50.34(f)(2)(xxvi) at the COL stage. The staff finds the COL information item acceptable because it is appropriate for COL applicants and holders to provide procedure and program details and demonstrate compliance with 10 CFR 50.34(f)(2)(xxvi).

9.3.2.7 Conclusion

The applicant has described extensive sampling of liquid and gaseous systems, including concentrations of impurities and added chemicals in the RCS and secondary water and fission products in water and gas space. The staff evaluated the PSS for the NuScale US460 design using the guidance of SRP Section 9.3.2, RG 1.21, and the EPRI Guidelines. Based on the above evaluation, the staff finds that the PSS design meets GDC 1, 2, 13, 14, 26, 60, 63, and 64 and 10 CFR 20.1101(b). As a result, the staff concludes that the design of the PSS during normal operation is acceptable. As discussed above, the staff considered NuScale’s exemption request from post-accident sampling requirements specified in 10 CFR 50.34(f)(2)(viii) and determined that that exemption, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

9.3.3 Equipment and Floor Drain Systems

9.3.3.1 Introduction

The equipment and floor drain system ensures that waste liquids, valve and pump leakoffs, and plant system drainage are collected and directed to the correct drain system components for processing or disposal and that excessive water accumulation and flooding is limited. The equipment and floor drain system comprises two separate, unconnected systems, the RWDS and the BPDS. The RWDS receives both radiologically contaminated and noncontaminated liquids and transfers the liquids to the LRWS for processing. The BPDS collects and segregates normally nonradioactive liquid waste from areas associated with power related or process-related functions outside the radiologically controlled area. The BPDS does not serve the RXB or RWB.

9.3.3.2 Summary of Application

FSAR, Section 9.3.3, "Equipment and Floor Drain Systems," provides a complete description of the equipment and floor drain system. Information provided includes the design bases, system and component descriptions, monitoring instrumentation, and details about the equipment and floor drain system operation.

ITAAC: FSAR, Part 8, Table 3.9-1 (BPDS Item 06 and RWDS Item 10). The ITAAC are evaluated in Section 14.3 of this report.

Initial Test Program: FSAR Table 14.2-21, "Test #21 Balance of Plant Drain System," describes the preoperational tests related to the BPDS that are being evaluated as part of the review. FSAR Table 14.2-30, "Test #30 Liquid Radioactive Waste System," describes the preoperational tests related to the RWDS that are being evaluated as part of the review. The ITP are evaluated in Section 14.2 of this SER.

9.3.3.3 Regulatory Basis

SRP Section 9.3.3, "Equipment and Floor Drainage System," contains the relevant regulatory requirements for this area of review and the associated acceptance criteria, summarized below, as well as review interfaces with other SRP sections:

- GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

- GDC 4 requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

- GDC 60 requires the nuclear power module design to include a means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

In addition, the following regulatory requirements also apply to the Equipment and Floor Drainage System:

- GDC 5 requires that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- GDC 64 requires that a means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

- 10 CFR 20.1406 as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive waste.

9.3.3.4 Technical Evaluation

The staff reviewed the equipment and floor drainage system design for compliance with the regulatory basis in Section 9.3.3.3 of this SER. The sections that follow give the results of the staff's review.

9.3.3.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

FSAR, Section 9.3.3.1, "Design Bases," states that the RWDS and BPDS serve no safety-related or risk-significant functions, are not credited for mitigation of a DBA, and have no safe-shutdown functions. FSAR, Section 9.3.3.3, "Safety Evaluation," states that the RWDS and BPDS do not require protection against external flooding, as the plant site selection criteria place the maximum external flood level at 1 ft below grade. Therefore, the staff's evaluation of GDC 2 in this case is based on the guidance in Regulatory Position C.2 of RG 1.29, which specifies that failure of systems that are not safety-related should not adversely affect safety-related systems. FSAR, Section 9.3.3.3, states that portions of the RWDS and BPDS system that could interact adversely with seismic Category I SSCs are designed to seismic Category II standards. FSAR, Table 9.3.3-1, classifies all RWDS and BPDS components as seismic Category III. The applicant noted BPDS piping that penetrates the CRE includes seismic Category I loop seal isolation devices at each penetration.

The applicant provided Note 4 at the end of Table 9.3.3-1-1, which states the following:

> Where SSC (or portions thereof) as determined in the as-built plant that are identified as Seismic Category III in this table could, as the result of a seismic event, adversely affect Seismic Category I SSC or result in incapacitating injury to occupants of the control room, they are categorized as Seismic Category II consistent with Section 3.2.1.2 and analyzed as described in Section 3.7.3.8.

Because portions of the as-built RWDS and BPDS that in a seismic event would adversely affect seismic Category I SSCs or would result in incapacitating injury to occupants of the control room will be categorized as seismic Category II, the staff finds that the design of the RWDS and BPDS is consistent with guidance in RG 1.29 and therefore meets GDC 2.

9.3.3.4.2 GDC 4, "Environmental and Dynamic Effects Design Bases"

The internal flood analysis in FSAR Section 3.4.1 evaluates the potential flooding impact on SSCs from pipe breaks, equipment failures, and fire suppression water. The internal flood analysis takes no credit for water removal by the RWDS or BPDS. Because failure of the RWDS and BPDS does not impact safety-related equipment functions or SSCs important-to-safety, the staff finds that the design of the RWDS and BPDS meets GDC 4.

Section 3.4 of this report provides the staff's evaluation of internal flooding.

9.3.3.4.3 GDC 5, "Sharing of Structures, Systems, and Components"

The applicant stated in FSAR Section 9.3.3.3 that the NPMs share the use of components in the RWDS and BPDS; however, failure of the shared RWDS and BPDS does not impair the ability of other NPMs to perform their safety functions. In the event of an accident in one NPM, the failure of this system to perform its functions that are not safety-related does not prevent an orderly shutdown and cooldown of the remaining NPMs. Because the failure of the RWDS or BPDS does not affect the functional performance of safety-related systems or SSCs important to safety, the staff finds that the design of the RWDS and BPDS meets GDC 5.

9.3.3.4.4 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

In FSAR Table 11.5-4, the applicant stated that if a high radiation condition is detected in the BPDS, the associated waste water sump pumps automatically shut down and transfer to manual control, and the discharge flowpath to the BPDS collection tanks automatically isolate. FSAR, Section 9.3.3.2.1, indicates that the RWDS and BPDS are designed to include additional tank capacity to support other activities, such as runoff from firefighting activities. Both systems are designed with radiation monitoring, which includes features that control the release of radioactive materials. Because the BPDS and RWDS have the capability to isolate and terminate tank discharge, the staff finds the design of the BPDS and RWDS meets GDC 60.

9.3.3.4.5 GDC 64, "Monitoring Radioactivity Releases"

The applicant stated that there are radiation monitors for source streams into the BPDS that have the potential to contain radioactive material. The BPDS radiation monitors provide continuous indication to the MCR. The RWDS is designed to receive radiologically contaminated liquids and normally noncontaminated liquids, including from the RCCWS drains. A radiation monitor located on the RCCWS drain tank monitors the normally noncontaminated liquid from the RCCWS to alert operators to an abnormal condition. Because the RWDS and BPDS monitor the discharge paths for radiation, the staff finds that the design of the RWDS and BPDS meets GDC 64.

9.3.3.4.6 Compliance with 10 CFR 20.1406, "Minimization of Contamination"

The requirements of 10 CFR 20.1406(b) specify, in part, that each standard design approval applicant describe how the facility design and procedures for operation will minimize, to the extent practicable, contamination of the facility and environment and the generation of radioactive waste. FSAR Tables 12.3-14 and 12.3-35 list the design features specific to the BPDS and RWDS, respectively, for the minimization of contamination. Some examples of these design features include the use of corrosion resistant stainless steel with smooth surfaces, use of components designed for 60-year design life of the plant, and the minimal use of embedded piping by the RWDS and BPDS. Because the NuScale US460 design provides adequate measures for leak detection and controls in the BPDS and RWDS design to minimize contamination, as described above, the staff finds that the system conforms to 10 CFR 20.1406.

9.3.3.5 Initial Test Program

Section 14.2 of this report provides the staff's evaluation of the ITP.

9.3.3.6 Conclusion

Based on the review of the information described above, the staff finds the equipment and floor drain system design acceptable because it meets the applicable regulatory requirements, including GDC 2, 4, 5, 60, and 64 and the provisions of 10 CFR 20.1406 for the minimization of contamination as it applies to equipment and floor drain SSCs that may have the potential to release radioactive materials to the facility, site, or environment.

9.3.4 Chemical and Volume Control System

9.3.4.1 Introduction

The NuScale US460 chemical and volume control system (CVCS) purifies reactor coolant, manages chemistry of the coolant (including boron concentration), provides reactor coolant inventory makeup and letdown, and supplies spray flow to the pressurizer to reduce the reactor coolant system pressure. The staff notes that the applicant defines the CVCS as only portions of the system outboard of the containment isolation system flanges. Regardless of how the US460 design defines the CVCS boundaries, the staff performed the review documented in this section consistent with the system scope defined in NuScale Design-Specific Review Standard (DSRS) 9.3.4, "Chemical and Volume Control System," which include CVCS SSCs inside containment such as the charging and letdown lines.

9.3.4.2 Summary of Application

The applicant provided a description of the CVCS in FSAR, Section 9.3.4, "Chemical and Volume Control System," summarized in Section 9.3.4.1 of this report. In addition, the NuScale US460 CVCS includes a reactor pressure vessel (RPV) high point degasification line, separate from the primary CVCS circulation flow path, to remove noncondensable gases that collect in the pressurizer vapor space, and a nitrogen distribution system connection supplies nitrogen to the RPV to support module startup activities.

ITAAC: ITAAC for the CVCS are provided in SDAA Part 8, Section 2.2.2, "Inspections, Tests, Analyses, and Acceptance Criteria." These ITAAC are evaluated in Section 14.3 of this report.

Technical Specifications: NuScale FSAR Part 4, Volume 1, "US460 Generic Technical Specifications," includes the TS associated with FSAR, Section 9.3.4. The GTS include LCO 3.1.9, "Boron Dilution Control," and LCO 3.4.6, "Chemical and Volume Control System Isolation Valves."

9.3.4.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 1, as it relates to system components being assigned quality group classifications and application of quality standards in accordance with the importance of the safety function to be performed

- GDC 2, as it relates to structures housing the facility and the system itself being capable of withstanding the effects of earthquakes

- GDC 4, as it relates to the capability of the system and the structure housing the system to withstand dynamic effects

- GDC 5, as it relates to shared systems and components important to safety being capable of performing required safety functions

- GDC 14, requires that the RCPB be designed, fabricated, erected, and tested to ensure an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture

- GDC 29, "Protection against anticipated operational occurrences," as it relates to the reliability of the CVCS to provide negative reactivity to the reactor by supplying borated water to the RCS in the event of anticipated operational occurrences, if the plant design relies on the CVCS to perform the safety function of boration for mitigation of DBEs

- GDC 33, "Reactor coolant makeup," as it relates to the CVCS capability to supply reactor coolant makeup in the event of small breaks or leaks in the RCPB, to function as part of the ECCS assuming a single active failure coincident with the loss of offsite power, and to meet ECCS TS, if the plant design relies on the CVCS to perform the safety function of safety injection as part of the ECCS

- GDC 60 and GDC 61, as they relate to CVCS components having provisions for venting and draining through closed systems to confine radioactivity associated with the effluents

- 10 CFR 20.1406, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive waste

- 10 CFR 50.34(f)(2)(xxvi), as it relates to providing for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials following an accident

The guidance in DSRS Section 9.3.4 lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

9.3.4.4 Technical Evaluation

The staff used a graded review approach to evaluate the CVCS. FSAR Table 9.3.4-3, details the SSC classification for the CVCS. Sections 3.2.2 and 17.4 of this report provide the basis for acceptability of the SSC classification for safety- and risk-significance, respectively. The staff reviewed the description and analysis of the CVCS provided in FSAR 9.3.4 including system and component descriptions, and normal and off-normal operation. The staff evaluation of applicable regulatory requirements is provided below.

9.3.4.4.1 GDC 1, "Quality Standards and Records"

The staff reviewed the applicant's CVCS to determine whether the appropriate portions of the CVCS SSCs were properly considered for quality classification. The staff noted that the applicant followed the guidance in RG 1.26.

The staff reviewed the classification of the CVCS lines and components inside containment (i.e., injection line, discharge line, pressurizer spray line, and high-point degasification line). FSAR Table 5.2-6, "Classification of Structures, Systems, and Components," identifies that these CVCS lines inside containment (designated as components of the reactor coolant system for the US460), but outside the reactor pressure vessel, are classified as safety-related and Quality Group A. The staff confirmed that, according to FSAR Section 3.2.2, these Quality Group A SSCs are appropriately designed to the ASME Code, Section III, Class 1. The staff finds this CVCS quality classification acceptable because 10 CFR 50.55a(c)(1) requires that components that are part of the reactor coolant pressure boundary (RCPB) meet the requirements for Class 1 components in the ASME Code, Section III.

The staff noted that the CVCS has a safety-related isolation function carried out by the demineralized water isolation valves. The containment isolation function is performed by the safety-related containment isolation valves (CIVs) that are a part of the containment system and is reviewed in Section 6.2 of this SER. Through the review of diagrams and quality classification tables, the staff confirmed that the demineralized water isolation valves are categorized as Quality Group C, as defined in RG 1.26. The staff also confirmed in FSAR Table 6.2-8, "Classification of Structures, Systems, and Components," and Figure 6.6-1, "ASME Class Boundaries for the NuScale Power Module Piping Systems," that the CVCS injection check valve, the CVCS discharge air operated valve, the CVCS pressurizer spray check valve, and the RPV high-point degasification solenoid operated valve, along with the associated piping from the CIVs, are categorized as ASME Code, Section III Class 3 and Quality Group C. The staff finds these CVCS quality classifications acceptable because the applicant followed the guidance in RG 1.26.

The staff notes that all other CVCS components and piping outside containment are categorized as Quality Group D, which is also defined in RG 1.26. The staff also finds this CVCS quality classification acceptable because the applicant followed the guidance in RG 1.26.

The staff finds that the NuScale US460 CVCS meets the requirements of GDC 1 because the design has acceptable quality group classifications that are in accordance with RG 1.26.

9.3.4.4.2 GDC 2, "Design Bases for Protection against Natural Phenomena"

The staff reviewed the applicant's CVCS to determine whether it was designed according to the appropriate seismic category. The staff noted that the applicant followed the guidance in RG 1.29.

The staff reviewed applicable design information in FSAR Sections 5.1, 5.2, and 9.3.4. The staff noted that the CVCS lines inside containment (i.e., injection line, discharge line, pressurizer spray line, and high-point degasification line) and the components within these lines are all designed to seismic Category I standards, which the staff finds acceptable because the applicant followed the guidance in RG 1.29.

The staff confirmed that the DWS isolation valves, the CVCS injection check valve, the CVCS discharge air operated valve, the CVCS pressurizer spray check valve, and the RPV high-point degasification solenoid operated valve, along with the associated piping from the CIVs, are all designed to seismic Category I standards. The staff finds these CVCS seismic classifications acceptable because the applicant followed the guidance in RG 1.29.

The staff noted that the hydrogen bottle, spool piece vent valves, degasification line flexible hose, and instrumentation and mechanical ball joints in the module bay are designed to seismic Category II standards. The staff also confirmed that the piping from the Class 3 CVCS valves to the disconnect flanges is designed to seismic Category II standards. The staff finds this acceptable because having such a seismic Category II classification will prevent the equipment and piping from adversely impacting a seismic Category I component following a safe-shutdown event.

The staff noted that all other CVCS components and piping outside containment are designed to seismic Category III standards, which the staff finds acceptable because continued reliance on these SSCs after an SSE is not required and failure of these SSCs after an SSE would not adversely affect any other seismic Category I SSC.

The staff also confirmed that the portion of the RXB where the CVCS is located is designed to seismic Category I, which protects the CVCS from external phenomena.

The staff finds that the NuScale US460 CVCS meets the requirements of GDC 2 because the design has acceptable seismic classifications that are in accordance with RG 1.29.

9.3.4.4.3 GDC 4, “Environmental and dynamic effects design bases”

The staff reviewed the CVCS for compliance with the requirements of GDC 4 with respect to the capability of the system and the structures housing the system to withstand the effects of pipe breaks, including the effect of pipe whip, jet impingement, and the environmental conditions resulting from high- and moderate-energy line breaks, as well as the effect of flow instabilities and attendant loads (water hammer). Compliance with the requirements of GDC 4 is based on the identification of the essential portions of the system as protected from dynamic effects, including internal and external missiles, pipe whip, and jets, and the ability of the system to continue to perform its safety function in the environmental conditions that may result from high- and moderate-energy line breaks and the resulting discharged fluid.

Sections 3.6.1 and 3.6.2 of this SER document the staff’s review of protection against dynamic effects associated with pipe ruptures for the CVCS.

The CVCS containment isolation valves perform the safety-related function of containment isolation, but are part of the containment isolation system and are evaluated in Section 6.2.4 of this SER. The demineralized water isolation valves also perform a safety-related function to isolate the DWS in the event of a dilution transient. These valves and their associated sensors may be subject to harsh environmental conditions and, therefore, must comply with the requirements of 10 CFR 50.49. FSAR Table 3.11-1, “List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments,” indicates that these valves will be located in Equipment Qualification Zone RXBP-1 for the containment isolation valves and Zone RXBG-2 for the demineralized water isolation valves and qualified for a harsh environment. As the CVCS valves will be qualified for a harsh environment, they will be designed to perform their safety-related isolation function while subject to the harsh environment. Section 3.11 of this SER addresses compliance with 10 CFR 50.49 for the qualification of equipment located in a harsh environment. Therefore, based upon the aforementioned information, the staff finds that the CVCS complies with the environmental provisions of GDC 4.

9.3.4.4.4 GDC 5, "Sharing of Structures, Systems, and Components"

The staff reviewed the NuScale US460 design to determine whether the nuclear power units share the CVCS. The staff confirmed that the nuclear power units do not share the CVCS; that is, each NPM has its own, dedicated CVCS.

The staff notes that the NPMs share the module heatup system and the BAS. However, the staff also confirmed that the module heatup system and the BAS are not safety-related systems and are not relied on to shut down any NPM.

The staff finds that the NuScale US460 CVCS meets the requirements of GDC 5 because the nuclear power units do not share any CVCS components important to safety.

9.3.4.4.5 GDC 14, "Reactor Coolant Pressure Boundary"

The staff reviewed the NuScale US460 CVCS design to ensure that the RCPB will have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

The staff reviewed system and component descriptions to determine whether those systems and components were adequately designed for the appropriate pressures. The staff compared system and component design pressures with normal RCS pressure and confirmed that the CVCS piping, and components were all rated for the appropriate pressure. The staff finds this acceptable because it will preclude failure of the RCPB that is maintained by the CVCS during normal operation.

The staff noted that the NuScale US460 CVCS neither has a volume control tank such as a typical large PWR would have, nor does it have associated holdup volume tanks. The CVCS simply continually recirculates RCS water through the purification components via the recirculation pumps and lets down water to the LRWS as necessary. Thus, the staff notes that because no tank is used in the CVCS during normal operation, there are no wall-inward buckling requirements for the CVCS. The staff does note, however, that an expansion tank connects to the CVCS, which is isolated via two isolation valves. This expansion tank is in service only when the CVCS is in recirculation mode (i.e., isolated from the NPM). This expansion tank provides the appropriate net-positive suction head to the recirculation pumps when they are running in recirculation mode. Because this expansion tank is normally isolated from the CVCS via two isolation valves and operates only when the CVCS is isolated from the RCS, the staff did not review it for adequate protection of wall-inward buckling as part of this section of the SER. The staff determined that wall-inward buckling of tanks containing radioactive effluents is not a concern for the NuScale US460 CVCS because the CVCS has no tank that could fail via wall-inward buckling during normal operation.

The staff reviewed FSAR Section 9.3.4 to determine whether the CVCS's purification components have adequate overtemperature and overpressure protection. The staff noted that the NuScale US460 CVCS consists of a bypass line that diverts flow around the ion exchange vessel on a high-temperature signal to protect the ion exchange resins from damage. The staff confirmed that the overtemperature protection for the CVCS's purification components was adequately designed. In addition, FSAR Section 9.3.4.2 and Section 9.3.4.4 state that the CVCS design includes instrumentation to measure the differential pressure across ion exchange vessels and reactor coolant filters, as well as alarms to notify control room personnel of high differential pressure that could indicate reduced performance of these components. This instrumentation is also shown in FSAR Figure 9.3.4-1.

The staff reviewed the system descriptions and schematics to determine whether the CVCS components and piping, which contain boric acid, are adequately protected against boric acid precipitation. FSAR Section 9.3.4.2.1, states that all BAS components are located in the RXB, which is maintained above 10 °C (50 °F), and Table 9.3.4-2, "Boron Additional System Major Equipment with Design Data and Parameters," identifies the maximum boron concentration for the BAS as 6000 ppm**.** The staff confirmed the specified maximum boron concentration at the minimum RXB temperature is less than the solubility limit of boric acid in water, and therefore, the staff finds that the CVCS and BAS are adequately protected against boron precipitation.

The staff also reviewed the drawings and schematics associated with the CVCS to determine whether the CVCS and its interfacing components were designed to appropriately maintain chemistry in all portions of the CVCS to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The staff noted in FSAR Figure 6.3-1, that the ECCS line between the reset valve, trip valve, and associated ECCS valve, which contains CVCS water, is isolated from the CVCS during normal operation. The staff confirmed that the water chemistry in the stagnated water lines, such as in the ECCS, will be appropriately maintained in accordance with a site-specific water chemistry program. The primary mechanisms for controlling chemistry in the stagnated water lines are through startup procedures and periodic line flushing and recirculation, both of which are the responsibility of the COL holder as stated in COL Items 13.5-1 and 13.5-4. The staff notes that NuScale's FSAR also contains COL Item 5.2-2, which directs the COL applicant to develop and implement a strategic water chemistry plan that is in accordance with the EPRI Guidelines (Appendix H to these guidelines focuses on establishing low oxygen and low anion chemistry in stagnant lines). Furthermore, the staff notes that the design of the NuScale US460 CVCS filtration and purification components ensures that particulates do not accumulate in the stagnated water lines. The staff finds that the COL information items referenced above, in addition to the design of the CVCS, are adequate for ensuring that the water chemistry of the stagnated water lines in the ECCS is appropriately maintained.

The staff reviewed FSAR Section 9.3.4, on the materials and chemistry aspect of the CVCS. The FSAR references the EPRI Guidelines as the standard for evaluating water chemistry. Although the staff does not formally review or issue a safety evaluation of the various EPRI water chemistry guidelines (including the EPRI Guidelines), the guidelines are recognized as representing industry best practices in water chemistry control. Extensive experience in operating reactors has demonstrated that following the EPRI Guidelines minimizes the occurrence of corrosion-related failures. Further, the EPRI Guidelines are periodically revised to reflect evolving knowledge of best practices in chemistry control.

FSAR Section 9.3.4.3, states that "Action Level 2 and Action Level 1 conditions require correction within 24 hours and 7 days, respectively." This section of the FSAR also states that Action Level 3 conditions would require immediate corrective action. The immediate corrective action would be in accordance with the EPRI Guidelines. The NRC staff has reviewed the proposed corrective actions for out of specification primary water chemistry parameters and time requirements for corrective action proposed by NuScale. The staff finds the proposed corrective actions and associated timeframes acceptable because they will be in accordance with the EPRI Guidelines.

The staff finds that the NuScale US460 CVCS meets the requirements of GDC 14 because the design has acceptable overpressure and overtemperature control and acceptable boric acid precipitation protection, and it provides reasonable assurance that the probability of corrosion-induced failure of the RCPB will be minimized, thereby maintaining the integrity of the RCPB.

9.3.4.4.6 GDC 29, "Protection against Anticipated Operational Occurrences"

The staff reviewed the NuScale US460 CVCS to ensure that it was designed to accomplish its safety functions in the event of an anticipated operational occurrence. The staff noted that the only safety function of the CVCS is to provide containment isolation and isolation for dilution sources. In the CVCS, the only dilution source comes from the DWS, which is part of the makeup subsystem of the CVCS. In Chapter 15, an inadvertent dilution of the RCS by the CVCS is considered an anticipated operational occurrence. The staff notes that the module protection system (MPS), which is safety-related, along with the two redundant, safety-related demineralized water isolation valves, provides an extremely high probability that the CVCS will accomplish its safety function once called upon in the event of this anticipated operational occurrence. The Chapter 15 analysis also assumes at certain conditions, automatic letdown will be restricted. This will allow an inadvertent boron dilution event to be mitigated by a reactor trip on an increase in pressurizer level. The staff further notes that the NuScale US460 reactor does not rely on the CVCS to provide borated water to the RCS for any anticipated operational occurrence or accident. Therefore, the staff finds that the CVCS meets the requirements of GDC 29 because NuScale designed the CVCS with redundant demineralized water isolation valves that will isolate the dilution source once it receives a signal from the MPS, and because the CVCS is not relied on for mitigating the effects of any anticipated operational occurrence or accident.

9.3.4.4.7 Exemption from GDC 33, "Reactor Coolant Makeup"

The applicant provided rationale to support that an exemption request from GDC 33, which requires a system to supply reactor coolant makeup for protection against leaks and small breaks in the RCPB, would be justified. The applicant stated that the US460 design does not require makeup to protect against leaks and small breaks in the RCPB. The applicant stated that the NPM designs, in conjunction with the passive design and operation of the ECCS and DHRS, ensure that fuel integrity is not challenged (i.e., does not exceed the SAFDLs) in the event of a leak or small break in the RCPB.

- 10 CFR 52.137(a)(3) requires a standard design application to contain a final safety analysis report 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 including the following:

  …

  (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 Part 50, Appendix A, GDC 33, states the following:

Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that 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) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.10 CFR 52.7 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) states, in part, that the two conditions that must be met for granting an exemption are the following:'

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

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 has determined that the justification for the proposed exemption would be 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 DBE, and does not create new accident precursors. The NuScale Power Plant US460 design incorporates specific design provisions ensuring adequate reactor coolant inventory so that RCPB leaks and small breaks do not result in loss of core cooling and specific acceptable fuel design limits are not exceeded. Therefore, as required by 10 CFR 50.12(a)(1), the staff has determined that the justification for the proposed exemption would pose no undue risk to public health and safety.

Consistent with Common Defense and Security

The proposed exemption would 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 determined that the common defense and security would not be impacted by this proposed exemption.

Special Circumstances

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 purpose of GDC 33 is to provide "protection against small breaks in the reactor coolant pressure boundary." The staff reviewed the NuScale US460 CVCS design to determine the technical adequacy of the applicant's request for an exemption from the requirements of GDC 33. FSAR Section 9.3.4 states that one function of the CVCS during normal operation is maintain reactor coolant inventory by providing and controlling RCS makeup and letdown. The staff also notes that the applicant's CVCS design and function are consistent with the CVCS of typical PWRs for maintaining RCS inventory during normal operations via a pressurizer level control program. Therefore, the staff has determined that the NuScale US460 design would meet the underlying purpose of GDC 33, in part, by providing a system (i.e., the CVCS, which is important for the day-to-day, safe operation of the plant) that protects against specified acceptable fuel design limit (SAFDL) violations that could occur during normal operation as a result of RCPB leakage.

Further, FSAR Section 5.4.3 and Section 6.3 states the main function of the DHRS and ECCS, respectively, is to provide decay heat removal during and after anticipated operational occurrences and postulated accidents. The staff also notes that the applicant's DHRS does not communicate directly with the RCS and its ECCS design does not inject an external supply of coolant into the reactor as an ECCS does in a typical PWR. Instead, the ECCS redirects the reactor coolant, which is already in the RPV, into a flowpath where heat passively exchanges with the UHS, maintaining core coolability. The applicant cannot meet, verbatim, the portion of GDC 33 that requires a reactor coolant makeup system to ensure that the SAFDLs are not exceeded during off-normal operation (e.g., anticipated operational occurrences, accidents) as a result of small RCPB piping or component ruptures. However, the staff notes that the application would meet the underlying purpose for off-normal operation in two ways. The first is the applicant relies on the safety-related containment isolation and DHRS and ECCS to provide SAFDL protection by maintaining core inventory and coolability. Within this context the staff considers core coolability to be sufficient heat removal such that the fuel cladding barrier remains intact and is separate from the less restrictive core cooling acceptance criteria prescribed in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." Second, for off-normal transients that do not actuate ECCS, the applicant has demonstrated that the results do not exceed SAFDL acceptance criteria.

The staff finds that, while the NuScale US460 design does not have a safety system capable of providing coolant injection (e.g., safety injection pump injecting water from a borated refueling water storage tank into the core as seen in typical large LWRs), as required by GDC 33, the US460 design has an alternative means, as described above, of maintaining reactor coolant inventory and coolability during off normal-transients. Therefore, based on the evaluation above, the staff has determined that the NuScale CVCS would meet the underlying purpose of GDC 33 because the NuScale CVCS protects against the detrimental effects of normal operational coolant loss and the safety-related NuScale DHRS and ECCS protects against the detrimental effects of off-normal operational coolant loss.

Section 6.3 and Chapter 15 of this report provide the staff's review of the analyses that demonstrate SAFDL protection.

Exemption request conclusion

Based on the evaluation above, the staff finds that the applicant's design would meet the requirements for an exemption under 10 CFR 50.12(a). The staff considered NuScale's exemption request and determined that this exemption, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

9.3.4.4.8 GDC 60, "Control of Releases of Radioactive Materials to the Environment," and Leakage Detection

The staff reviewed the NuScale US460 CVCS design to determine whether the system contained appropriate provisions for venting and draining to ensure that the release of radioactive material from the CVCS would be carried out in a controlled manner. The staff also reviewed the CVCS to determine whether such vent and drain systems were designed appropriately to ensure adequate confinement of radioactivity associated with the vented and drained effluents.

The staff confirmed through the review of information in the FSAR, including drawings and diagrams, that the CVCS is equipped with appropriate vent lines and drain lines that discharge radioactive effluents from the CVCS to the plant's radioactive waste management system (which includes the liquid, gaseous, and solid waste management systems). Section 9.3.3 of this report provides the staff's review of the equipment and floor drainage system, and Chapter 11 of this report contains the staff's review of the radioactive waste management system.

The staff reviewed the NuScale US460 CVCS to determine whether it includes appropriate provisions for leakage control and detection. The staff confirmed through the review of information in the FSAR that the CVCS design includes leak detection instrumentation that is capable of providing an alarm in the MCR. If letdown flow is higher than a predetermined setpoint then letdown automatically isolates. The staff finds this leakage detection and control scheme appropriate because it reasonably minimizes leakage from the CVCS. Furthermore, the staff confirmed that NuScale has identified COL Item 9.3-1, which directs a COL applicant that references the US460 standard design to submit a leakage control program, including an ITP, a schedule for retesting these systems, and the actions to take for minimizing leakage from such systems.

The staff finds that the NuScale US460 CVCS meets the requirements of GDC 60 because the CVCS is equipped with vent lines and drain lines that discharge radioactive effluent from the CVCS to the plant's radioactive waste management system thereby ensuring that the release of radioactive material from the CVCS would be carried out in a controlled manner. The staff further finds that the NuScale US460 CVCS meets the design requirements of 10 CFR 50.34(f)(2)(xxvi) because the CVCS is equipped with a leakage detection and control system. The programmatic aspects of 10 CFR 50.34(f)(2)(xxvi) are identified by COL Item 9.3-1. The staff did not review or approve these programmatic aspects of 10 CFR 50.34(f)(2)(xxvi) in connection with the NuScale US460 SDAA.

9.3.4.4.9 GDC 61, "Fuel Storage and Handling and Radioactivity Control"

The staff reviewed the NuScale US460 CVCS design to determine whether it could ensure adequate safety under normal and postulated accident conditions. The staff noted that the CVCS has provisions for being shielded, where necessary, to minimize radiation levels. For example, the FSAR indicates that concrete cubicles provide shielding of the CVCS highly radioactive ion exchange vessels and reactor coolant filters. Furthermore, the staff noted that primary coolant piping in CVCS equipment rooms is shielded to minimize dose rates. The staff also noted that the CVCS ion exchange resins are expected to be adequately retained in the ion exchange vessel and are transferred to the solid waste management system under controlled procedures. The staff confirmed that the CVCS design factored in ALARA goals. Chapter 12 of this report contains a more detailed review of ALARA considerations.

The staff also confirmed that the appropriate portions of the CVCS have associated inservice inspection and testing requirements, as delineated by the ASME Code.

Based on the review above, the staff finds that the NuScale US460 CVCS meets the requirements of GDC 61 because the CVCS design confines radioactive material and reduces the potential exposure to radioactive materials to the lowest practical levels.

9.3.4.4.10 10 CFR 20.1406, "Minimization of Contamination"

In Sections 12.3 and 12.4 of this report, the staff reviewed the NuScale US460 CVCS design to determine whether it could facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive waste.

9.3.4.4.11 Safety and Risk-Insights

The staff notes that the capability of the CVCS to provide makeup inventory to the RCS is relied on in several sequences in the probabilistic risk assessment (PRA). FSAR Table 19.1-3 identifies the CVCS check valve locations and CVCS flow area restrictions as key design features resulting from the PRA analyses for minimizing the likelihood of a break outside containment and reducing large release frequency and the conditional containment failure probability, respectively. FSAR Section 6.2.1 states that flow-restricting venturis are provided in the containment vessel penetration for the CVCS injection and CVCS discharge lines. The venturi flow restriction diameter is 2.14 cm (0.844 inches) or less and provides mitigation of design-basis CVCS breaks outside containment. In letter dated July 23, 2024 (ML24205A211 Non-proprietary, ML24205A212 Proprietary), the applicant states the venturi inserts are nonstructural attachments with a non-pressure-retaining function as defined by ASME Code, Section III, Article NE-1132 and includes a figure detailing the ASME Class boundaries for the component. FSAR 6.2.1 identifies the venturis as Seismic Category I and Table 6.2-7 indicates their safety classification is safety-related. Given their role in the safety analysis, that staff finds this safety and seismic classification of the flow-restricting venturis acceptable. The performance of the venturis to mitigate design-basis CVCS breaks and beyond-design-basis unisolated CVCS breaks outside containment are included in the staff's review provided in Chapter 15 and Chapter 19, respectively, of this report.

9.3.4.5 Initial Test Program

The ITP tests related to the CVCS, such as "Chemical and Volume Control System Test #33," and "Primary and Secondary System Chemistry Test #72," described in FSAR Section 14.2, are evaluated as part of the staff's review and are documented in Section 14.2 of this report.

9.3.4.6 Combined License Information Items

Table 9.3.4-1 lists the COL information item numbers and descriptions related to the CVCS from FSAR, Table 1.8-1.

Table 9.3.4-1 NuScale COL Information Items for Section 9.3.4

COL Item No.	Description	FSAR Section
9.3-1	An applicant that references the NuScale Power Plant US460 standard design will submit a leakage control program for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. The leakage control program will include an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems to as low as practical.	9.3

The NRC staff finds NuScale COL Item 9.3-1 acceptable because it is appropriate to develop this information as part of a site-specific application. The staff also finds NuScale COL Item 9.3-

1 appropriate to indicate the remaining requirements of 10 CFR 50.34(f)(2)(xxvi) that the COL applicant must address, as discussed above.

9.3.4.7 Conclusion

Based on the review above, the staff concludes that the NuScale US460 CVCS meets all applicable regulations defined in SRP Section 9.3.4.3 with the exception of the programmatic portions of 10 CFR 50.34(f)(2)(xxvi). The staff further concludes that the applicant has included the appropriate COL information item to ensure that site specific features of the CVCS will be addressed and appropriately implemented.

9.3.5 Standby Liquid Control System (BWR)—Not Applicable

This SRP section is for boiling-water reactors (BWRs) and therefore not applicable to the NuScale US460 SDAA.

9.3.6 Containment Evacuation System and Containment Flooding and Drain System

9.3.6.1 Introduction

The containment evacuation system (CES) removes and analyzes noncondensable gases and water vapor from the containment vessel (CNV) free volume. The main functions of the CES are to (1) establish and maintain vacuum in the CNV, (2) measure CNV pressure, (3) monitor radiation levels, and (4) provide Leak Detection monitoring for CNV. The main components in the CES are vacuum pumps, condenser, sample vessel, and supporting instrumentation.

The CES sample vessel level instrumentation detects a level increase in the CES sample vessel, which correlates to a detection of unidentified RCS leakage rate. The CES inlet pressure instrumentation detects a pressure increase in the CES inlet pressure, which correlates to a detection of unidentified RCS leakage. The RCS leakage detection requirements are evaluated in Section 5.2.5 of this report.

9.3.6.2 Summary of Application

FSAR, Section 9.3.6, “Containment Evacuation System,” provides the design bases, description, and safety evaluation of the CES. The CES performs the function of monitoring RCS leak detection. The CES system removes water vapor from the CNV during NPM startup and operation and provides a method to condense, collect, and sample the water removed to measure leakage within the CNV. FSAR, Table 9.3.6-1, lists design information for the system components. FSAR, Section 9.3.6.2.2, “Component Descriptions” describes each major component and FSAR, Section 9.3.6.2.3, “System Operation” describe CES function.

FSAR Part 8, Section 2.3, “Containment Evacuation System” and Table 2.3-1 describe the ITAAC for CES instrumentation. ITAAC is evaluated in Section 14.3 of this report.

9.3.6.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 2, as it relates to the capabilities of the structures housing the system and the system itself important to safety SSCs being designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of safety functions.

- GDC 5, insofar as it requires that SSCs important to safety not be shared among power units unless the applicant can show that such sharing will not significantly impair the ability of the shared SSCs to perform their safety functions.

- GDC 60, as it relates to the capability to suitably control release of radioactive materials to the environment.

The guidance in NuScale DSRS, Section 9.3.6, "Containment Evacuation and Flooding Systems," lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

NuScale DSRS, Section 9.3.6, identifies regulations in addition to those listed above but states that the specific DSRS acceptance criteria are those acceptable to meet the relevant requirements of GDC 2 and 60. Therefore, the staff did not evaluate CES against the other requirements listed in DSRS, Section 9.3.6. The staff evaluates GDC 5, which is not described in DSRS, Section 9.3.6, to address areas of the NuScale US460 design that were not anticipated by the staff when the DSRS was developed.

9.3.6.4 Technical Evaluation

During normal operation, the CES supports three separate methods that can detect leakage into the CNV: CNV pressure, CES sample vessel level detection, and sample vessel radiation monitoring. Two of these methods, CNV pressure and CES sample vessel level detection, can quantify leakage into the CNV. The CES evacuates the CNV to remove the water that remains after the draining process and to establish the CNV normal operating condition. The RCS leakage detection methodology is evaluated in Section 5.2.5 of this report.

The CES operates from the MCR using the module control system (MCS). The MCS monitors the CES for abnormal conditions, such as high pump suction pressure, high pump discharge temperature, high condenser pressure, and high sample vessel level, and trips the running CES vacuum pump and closes its suction and discharge valves if an abnormal condition is detected. The CES is not a safety-related system and is not assumed to operate during or after any DBA.

The CES off-normal operations include the following:

- high-radiation level in gases discharged from the CES condenser

- equipment failure affecting one or both CES vacuum pumps

A low voltage ac electrical distribution system (ELVS) provides electrical power for the CES vacuum pumps and valves. A normal dc power system (EDNS) provides electrical power for the CES I&C equipment.

9.3.6.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

Consistent with GDC 2, SSCs shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

The applicant stated that it considered GDC 2 in the design of the CES; associated support SSCs that could affect safety-related components to prevent damage from an SSE; and the RXB which provides protection from external natural phenomena. The applicant identified the quality group and seismic category of CES components in FSAR Table 3.2-1. The staff finds that the quality group identification is consistent with RG 1.26. All SSCs of the CES are designed to Quality Group D. The CES inlet pressure instrumentation and its connecting piping, up to and including isolation valves, are designed to Seismic Category I standards.

The CES operates from the MCR using the module control system (MCS). The MCS monitors the CES for abnormal conditions, such as high pump suction pressure, high pump discharge temperature, high condenser pressure, and high sample vessel level, and trips the running CES vacuum pump and closes its suction and discharge valves if an abnormal condition is detected. The CES is not a safety-related system and is not assumed to operate during or after any DBA.

FSAR Table 9.3.6-1 defines the CES system as Seismic Category II with its system interface valves defined as Seismic Category I. All CES instrumentation is Seismic Category II, with the exception of sample vessel radiation transmitter which is Seismic Category III.

Based on the above, the staff concludes that the requirements of GDC 2 are satisfied.

9.3.6.4.2 GDC 5, "Sharing of Structures, Systems, and Components"

All NPMs have independent CESs, and thus, the CES is not shared among the NPMs. Each NPM is supported by its own dedicated CES. To monitor CNV leakage, the CES establishes and maintains a vacuum in the CNV by removing water vapor and noncondensable gases from the CNV using a vacuum pump that draws gases from the top of the CNV and discharges to the CES condenser to sample and measure leakage.

Therefore, the staff concludes that the requirements of GDC 5 are satisfied.

9.3.6.4.3 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

FSAR, Section 9.3.6.3, states the following:

> Consistent with General Design Criterion 60, the CES has the capability to control the release of radioactive materials to the environment during normal operation. Section 11.5, Radiation Monitoring, provides information on the CES radiation monitors.

FSAR Figure 9.3.6-1 shows radiation monitoring locations on sample vessel and piping prior to exhausting to the RBVS or GRWS. The staff finds that locating the radiation monitor between the condenser and the filter bank is acceptable because radiation is monitored before the radioactive gases are filtered by the filter bank and released to the environment.

FSAR Section 9.3.6 defines the CES function for monitoring radioactivity levels in the noncondensable gas removed from the CNV and, depending on the radioactivity level in the gas, either filtering and discharging the gas through the RBVS plant exhaust stack or transferring the gas to the GRWS if radiation levels of the CES gaseous process exceed specified limits.

The CES for each NPM includes a containment evacuation sample vessel which includes pressure, temperature, and radiation monitoring instrumentation. FSAR Table 11.5-1 indicates radiation instrumentation to monitor “sample tank liquid radiation” and “gaseous discharge, noble gas and iodine radiation” within CES piping. The staff noted that the applicant provided rationale to support that an exemption request from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the CES would be justified, which is discussed in Section 6.2.4 of this report.

The staff finds that the information provided in the application is consistent with DSRS Section 9.3.6 and, therefore, concludes that the requirements of GDC 60 are satisfied.

9.3.6.5 Initial Test Program

The staff evaluates the ITP in Section 14.2 of this SER.

9.3.6.6 Technical Specifications

SDAA, Part 4, LCO 3.4.5, “RCS Operational Leakage,” and LCO 3.4.7, “RCS Leakage Detection Instrumentation,” are related to CES instrumentation and leakage limits to verify within required limits.

9.3.6.7 Conclusion

The staff review found that the CES meet GDC 2, 5, and 60, consistent with the guidance provided in DSRS Section. Based on the review above, the staff concludes that the CES for the NuScale US460 design satisfies the relevant requirements for the CES as described in Section 9.3.6.3 of this SER.

9.3.7 Containment Flooding and Drain System

9.3.7.1 Introduction

The CFDS transfers liquids and gases between the CNV free volume and other plant systems.

9.3.7.2 Summary of Application

FSAR Section 9.3.7 provides information on the system. The CFDS, which is not safety-related, is used to flood a CNV with borated reactor pool water and drain water back to the reactor pool. The CFDS can also be used to add water to a CNV during a beyond-design-basis event. FSAR Figure 9.3.7-1 shows the CFDS system diagram.

The CFDS functions include the following:

- flooding the CNV with reactor pool water during NPM cooldown in preparation for refueling operations
- draining the CNV during NPM startup operations and routing water removed from the CNV to the reactor pool through the PCWS

- routing noncondensable gases removed from the CNV during NPM startup operations through a high-efficiency particulate air filter to the RBVS plant exhaust stack for release to the environment, if radioactivity levels are below specified limits

- providing the capability to add borated water from the reactor pool to the CNV to remove decay heat during a beyond-design-basis accident

Multiple NPMs share the CFDS because the system is used only as needed to prepare and recover an NPM from conditions needed for refueling. The CFDS includes two pumps that can be aligned to either flood or drain any of the six supported NPMs and includes a drain separator tank used to separate entrained gases from the water drained from a CNV before the water returns to the reactor pool.

ITAAC: FSAR, Part 8, Table 3.9-1 (Item 5), "Radiation Monitoring – Shared Systems" describes the ITAAC to test that the CFDS automatically responds to a CFDS high-radiation signal by closing the CFDS containment drain separator gaseous discharge isolation valve. These ITAAC are evaluated in Section 14.3 of this report.

9.3.7.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 2, as it relates to the system's capability to withstand the effects of earthquakes

- GDC 5, as it relates to sharing SSCs important to safety among nuclear power units

- GDC 60, as it relates to the capability to suitably control release of radioactive materials to the environment

The guidance in NuScale DSRS, Section 9.3.6, "Containment Evacuation and Flooding Systems," lists the acceptance criteria adequate to meet some of the above requirements, as well as review interfaces with other SRP sections.

NuScale DSRS, Section 9.3.6, identifies regulations in addition to those listed above but states that the specific DSRS acceptance criteria are those acceptable to meet the relevant requirements of GDC 2 and 60. Therefore, the staff did not evaluate CFDS against the other requirements in DSRS, Section 9.3.6. The staff evaluated GDC 5, which is not described in DSRS, Section 9.3.6, to address areas of the NuScale design that were not anticipated by the staff when the DSRS was developed.

9.3.7.4 Technical Evaluation

The staff reviewed NuScale FSAR Section 9.3.7, "Containment Flooding and Drain System," using guidance provided in NuScale DSRS Section 9.3.6.

9.3.7.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

Consistent with GDC 2, SSCs shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

The applicant stated that it considered GDC 2 in the design of the CFDS; associated support SSCs that could affect safety-related components to prevent damage from an SSE; and the

RXB, which provides protection from external natural phenomena. FSAR Section 9.3.7 indicates the SSCs that could adversely affect seismic Category I components are designed to seismic Category II standards in accordance with RG 1.29. FSAR Table 9.3.6-1 defines the CFDS system SSCs as seismic Category III, with system valves defined as seismic Category II. The applicant identified the quality group and seismic category of CFDS components in FSAR Table 9.3.7-1. The staff concludes that the quality group identification is consistent with RG 1.26. The staff finds that the requirements of GDC 2 are satisfied.

9.3.7.4.2 GDC 5, "Sharing of Structures, Systems, and Components"

There are two independent CFDS pumps supporting the isolated six NPMs. Flooding and draining an individual CNV for each NPM is isolated from the CFDS by a CFDS interface valve. The CFDS contains controls to prevent inadvertent makeup to operating NPM. Controls are provided to avoid draining or filling multiple CNV at same time. If more than one CFDS interface valve is open, an interlock either stops the CFDS pump, if operating, or prevents the CFDS pump from starting. The CFDS system functions are not used during normal operation but needed for draining the CNV during startup and flooding the CNV in preparation for startup of NPM. The CFDS also provides the capability of adding borated water to remove decay heat during a beyond-design-basis event.

The CFDS is not a safety-related system. Sharing of the CFDS SSCs among the NPMs does not impair their ability to perform safety functions, such as an accident in one NPM that requires an orderly shutdown and cooldown of the remaining NPMs. Therefore, the staff concludes that the requirements of GDC 5 are satisfied.

9.3.7.4.3 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

FSAR Section 9.3.7 states that water removed from the CNV during draining is pumped to the containment drain separator tank. The separator tank removes entrained gases, which then are vented past radiation monitors. FSAR Table 12.3-16 indicates, "high radioactivity measured by the CFDS process radiation monitor results in automatic isolation of the line on a high-radioactivity indication which minimizes transfer of radioactive effluent to the RXB exhaust stack."

FSAR Figure 9.3.7-1 shows the radiation monitor on the drain separator tank discharge line. FSAR Table 11.5-4 defines response to the event of high radiation indication or loss of signal which results in shutdown of CFDS operating pumps and the isolation of CFDS Drain Separator Gas Discharge. The staff finds this design acceptable because it adequately addresses the control of radioactive gases during draining of the CNV. The staff finds that locating the radiation monitor between the drain separator tank and the filter bank is acceptable because radiation is monitored before the radioactive gases are filtered by the filter bank.

Each system monitors process variables, including pressure, temperature, tank level, and radioactivity for leak detection, as noted in FSAR Table 12.3-16.

The staff finds that the information in the application is consistent with DSRS Section 9.3.6 and, therefore, concludes that the requirements of GDC 60 are satisfied.

9.3.7.5 Initial Test Program

The staff evaluates the ITP in Section 14.2 of this report.

9.3.7.6 Conclusion

Based on the review above, the staff concludes that the CFDS for the NuScale US460 design satisfies the relevant requirements for the CFDS as described in Section 9.3.7.3 of this SER.

9.4 Air Conditioning, Heating, and Ventilation Systems

The HVAC evaluation for each major building or area is provided in the following subsections.

9.4.1 Control Room Area Ventilation System

9.4.1.1 Introduction

The CRVS serves the entire CRB. The CRVS boundary begins at the air intake of the outside of the CRB and extends to the points of discharge from the CRB. The plant protection system (PPS) isolates the control room envelope (CRE) and breathing air is provided by the CRHS under conditions of loss of all ac power, high radiation levels, smoke detection, or toxic gas detection.

The technical support center (TSC) is served by the CRVS, but not the CRHS. If the CRVS is not able to provide air of acceptable quality for pressurization of the TSC, the TSC is determined to be uninhabitable and is evacuated.

9.4.1.2 Summary of Application

FSAR Section 9.4.1, "Control Room Area Ventilation System," provides information associated with this section. The CRVS serves no safety-related functions, is not credited for mitigation of design basis accidents, and has no safe-shutdown functions. In conjunction with the CRHS, the CRVS maintains the CRE within the temperature and humidity limits needed to support personnel and to maintain equipment during normal conditions. The CRHS maintains the environment in the CRE habitable for personnel during abnormal and station blackout conditions when the CRVS is unavailable.

During normal operation, the CRVS maintains temperature and humidity control using two redundant 100 percent capacity air handling units (AHU). The standby AHU starts automatically if the operating AHU fails.

The air filtration unit (AFU) is used to filter outside air when radioactivity is detected. The AFU includes a charcoal adsorber that would be designed, constructed, and tested in accordance with RG 1.140.

The supply, return, and general exhaust ductwork serving the CRE are the only heating, ventilation, and air conditioning penetrations through the CRE. These penetrations include redundant isolation dampers that are located within the CRE to protect its occupants from hazardous conditions. These dampers can be closed to isolate the CRE, allowing the CRHS to pressurize and provide breathable air to the CRE. The CRE isolation dampers are qualified to shut tight against CRE pressure in support of the CRHS for maintaining main control room (MCR) habitability. There are no single active failures that would prevent isolation of the CRE.

The CRVS is normally powered by the low voltage alternating current (AC) electrical distribution system. During a loss of normal AC power, the backup power supply system provides power so that the CRVS can continue to operate.

On a loss of power to both CRVS air handler units or loss of power to the common augmented DC power system (EDAS) battery chargers, after a 10-minute delay, the CRVS isolates the CRE, and the PPS actuates the CRHS. System operation following loss of normal AC power does not affect the safety of MCR personnel or the performance of equipment needed to safely operate the plant.

ITAAC: The applicant gave the ITAAC associated with FSAR Section 9.4, in SDAA Part 8, Section, Section 3.2, "Normal Control Room Heating Ventilation and Air Conditioning System." These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: There are no GTS associated with the CRVS.

9.4.1.3 Regulatory Basis

SRP Section 9.4.1, "Control Room Area Ventilation System," and SRP Sections 12.3–12.4, "Radiation Protection Design Features," provide staff review guidance and acceptance criteria for the CRVS. The following are the relevant requirements of NRC regulations for this area of review:

- GDC 2, as it relates to system capability to withstand the effects of earthquakes.
- GDC 4, as it relates to the CRVS being appropriately protected against dynamic effects and being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal operation, maintenance, testing, and postulated accidents.
- GDC 5, "Control Room," as it relates to shared SSCs among nuclear power units.
- GDC 19, as it relates to maintaining the nuclear power unit in a safe condition under normal conditions and providing adequate radiation protection to permit access to and occupancy of the control room under accident conditions.

  The NRC staff notes that the applicant has provided rationale to support that an exemption request from GDC 19 to implement a design specific Principal Design Criterion (PDC) 19 that ensures the 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, would be justified. The staff's evaluation of the rationale that supports PDC 19 is provided in Section 6.4 of this report.

- GDC 60, as it relates to system capability to suitably control release of gaseous radioactive effluents to the environment.
- 10 CFR 20.1406, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the radioactive waste.
- 10 CFR 50.63, as it relates to necessary support systems providing sufficient capacity and capability to cope with an SBO event.

9.4.1.4 Technical Evaluation

The CRVS serves the entire CRB, which includes the CRE, the TSC, and other areas. Except in recirculation mode, the CRVS maintains the areas served at a positive pressure (at least 31.1 pascals (gauge) (1/8-inch water, gauge) with respect to the outside environment to limit infiltration of dust and radioactive materials.

The staff reviewed the CRVS in accordance with the review procedure in SRP 9.4.1. The results of the staff's review are provided below.

9.4.1.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

FSAR Section 9.4.1.1 states that the CRVS serves no safety-related functions, is not risk-significant, is not credited for mitigation of design-basis accidents and has no safe-shutdown functions. The CRVS is not required to operate during or after a DBE.

The staff based its review of CRVS compliance with GDC 2 requirements, in part, on adherence to RG 1.29, Regulatory Position C.2.

In reviewing the NuScale FSAR, the staff understands that the CRVS is a non-safety-related, seismic Category III-designed system.

FSAR Table 1.93 indicates NuScale conforms to SRP Section 9.4.1 (acceptance criterion II.1), which is GDC 2-applicable to the CRVS design.

The staff finds that the guidance of RG 1.29, Regulatory Position C.2, for non-safety-related portions has been appropriately followed by the applicant and therefore concludes that the CRVS complies with the requirements of GDC 2.

9.4.1.4.2 GDC 4, "Environmental and Dynamic Effects Design Bases"

The CRVS is designed to maintain a suitable ambient temperature and humidity for personnel and equipment in the MCR and other areas of the CRB during normal operation and when the non-safety-related backup diesel generators (BDGs) are available. The CRVS has radiation monitors, and smoke detectors located in the outside air intake and downstream ductwork, which allow the plant protection system or plant control system to isolate the CRE and the outside air intake as needed in the event of fires, failures, malfunctions, or high radiation.

The CRB itself is a mild environment with no credible high-energy sources as the result of equipment failure. FSAR Section 9.4.1 states that there is no credible source of a high-energy pipe failure within the CRB that could cause loss of function of the CRE isolation dampers.

The CRVS is not designed to serve safety-related functions. Based on the description above, the staff finds that the CRVS SSCs, including CRE isolation dampers, are compatible with the environmental conditions during normal operation, including the effects of missiles that may result from equipment failures or tornados, and therefore concludes that the design of the CRVS complies with the requirements of GDC 4.

9.4.1.4.3 GDC 5, "Sharing of Structures, Systems, and Components"

Even though the CRVS is shared between multiple reactor modules, FSAR Section 9.4.1.3, states, "the CRVS does not have a function relative to shutting down a module or maintaining it in a safe shutdown condition. Operation of the CRVS does not interfere with the ability to operate or shut down a module."

The applicant stated that GDC 5 is satisfied because control room operators can safely shut down all reactors should they have to leave the control room, and the reactor modules will remain in a safe condition when control room habitability is lost during a DBA.

According to GDC 5, SSCs important to safety shall not be shared among nuclear power plants unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

To satisfy GDC 5, the staff's review of the CRVS is to ensure that sharing of CRVS SSCs in multiple-NPM plants does not impact shutting down an NPM or maintaining it in a safe shutdown condition, including, in the event of an accident in one NPM, the ability of NuScale control room operators to shut down the remaining NPMs.

Control room habitability is not credited to be operational during DBAs because neither the CRVS nor the CRHS is safety-related. The analyses summarized in Chapter 15 of the FSAR demonstrate that no DBAs require the evacuation of the MCR. In the event of a beyond design basis accident that requires the evacuation of the MCR, very little time (on the order of minutes) is required to trip the unaffected reactors from the control room.

The NuScale US460 design does not credit any operator actions to mitigate DBEs. Specifically, in FSAR Section 15.0.0.6.4, NuScale stated the following:

> There are no operator actions credited in the evaluation of NuScale Power Plant US460 DBEs. After a DBE, automated actions place the NPM in a safe state and it remains in the safe-state condition for at least 72 hours without operator action, even with assumed failures.

According to the applicant, the NuScale US460 design does not need to credit any operator actions to mitigate DBEs. The staff determined that sharing of CRVS SSCs in multiple-NPM plants does not significantly impair their ability to perform their safety functions, including, in the event of an accident in one NPM, an orderly shutdown and cooldown of the remaining NPM(s), and therefore concludes that the design of the CRVS complies with the requirements of GDC 5.

9.4.1.4.4 PDC 19, "Control Room"

In SDAA Part 7, Exemption #17, NuScale provided rationale to support an exemption from GDC 19, "Control Room," would be justified, and proposed to implement a design-specific PDC 19 that meets the underlying purpose of the GDC 19 requirement for means to maintain the NPMs in a safe condition in the event of a control room evacuation (ML20204A986). NuScale states that the NPM design, as reflected in the FSAR (SDAA Part 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.

Upon detection of smoke in the outside air duct, the outside air isolation dampers are closed by the plant control system to isolate the CRB from the environment. The CRVS is then operated in recirculation mode to provide conditioned air to the occupied areas of the CRB, with no outside air being introduced into the building.

When gaseous or particulate radioactivity in the outside air duct exceeds the high setpoint, the normal outside air flowpath is isolated, and 100 percent of the outside air is bypassed through the AFU. If high levels of radiation are detected downstream of the AFU in the CRE supply duct, or if normal ac power is lost for 10 minutes, or if power is lost to all common augmented dc power system (EDAS-C) battery chargers, the CRE is isolated and breathable air is supplied by the CRHS. The staff's evaluation of the performance of the CRHS under DBA conditions is in Section 6.4 of this report.

The staff finds that the system design can protect control room personnel during normal operation and that the guidance of RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release," has been followed. Therefore, the staff concludes that the CRVS complies with PDC 19.

9.4.1.4.5 10 CFR 20.1406, "Minimization of Contamination"

FSAR Section 12.3.3.5, states the following:

> During normal operations, the normal control room HVAC system (CRVS) supplies conditioned air to the CRB, including the control room envelope (CRE), the technical support center, and the other areas of the CRB with outside air that is filtered (low and high efficiency) to maintain a suitable environment for personnel and equipment. The CRVS is designed to maintain a positive pressure inside the CRB with respect to adjacent spaces. Section 9.4.1, Control Room Area Ventilation System, contains additional details.
>
> If a high radiation indication is received from an outside air intake radiation monitor, the supply air is routed through the CRVS filter unit which provides additional HEPA and charcoal filtration. The CRVS is designed to maintain operator doses in the MCR and technical support center within PDC 19 limits.
>
> If power is not available, or if a high radiation indication is received from the radiation monitors in the CRE supply duct, the CRE isolation dampers close and the control room habitability system (CRHS) is initiated.

The staff finds that CRVS ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts and the AFUs are designed with features that minimize the time required for filter changes. Therefore, the staff concludes that the above-described design considerations constitute compliance with 10 CFR 20.1406.

9.4.1.4.6 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

GDC 60 requires that the nuclear power unit design include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

The CRVS has two AFUs located in the supply air side of the system. When gaseous or particulate radioactivity in the outside air duct exceeds the high setpoint, the normal outside air flowpath is isolated, and 100 percent of the outside air is bypassed through the AFUs to minimize radiation exposure to personnel within the CRB.

The staff considers that RG 1.140 is applicable to the CRVS AFUs because it describes the supply air atmospheric cleanup function. According to RG 1.140, standards acceptable to the NRC staff for the design and testing of the system include ASME N509, "Nuclear Power Plant Air-Cleaning Units and Components," and ASME AG-1, "Code on Nuclear Air and Gas Treatment."

The staff finds, at the SDA stage, that the CRVS is consistent with the guidance of RG 1.140 because the standards for the design and testing of the system include ASME N509 and ASME AG-1. Therefore, the staff concludes that the design and testing of the AFUs comply with GDC 60.

9.4.1.4.7 10 CFR 50.63, "Loss of All Alternating Current Power"

In FSAR Section 8.4, "Station Blackout," NuScale stated the following:

> The SBO duration for passive plant designs is 72 hours pursuant to Nuclear Regulatory Commission policy provided by SECY-94-084 and SECY95132 and the associated staff requirements memoranda. Passive plants are required to demonstrate that safety-related functions can be performed without reliance on AC power for 72 hours after the initiating event.
>
> The SBO does not pose a significant challenge to the plant, which does not rely on AC power for performing safety functions. A safe and stable shutdown is automatically achieved and maintained for 72 hours without operator actions… The control room remains habitable for the duration of the SBO event using the control room habitability system. The control room instrumentation to monitor the event mitigation and confirm the status of reactor cooling, reactor integrity, and containment integrity also remains available. The control room habitability system is described in Section 6.4.

In FSAR Section 9.4.1.3, NuScale stated the following:

> In a station blackout event, the CRE isolation dampers close to form part of the CRE boundary.

The staff finds that the design of the CRVS complies with 10 CFR 50.63 regarding the capability for responding to a station blackout (SBO), specifically maintaining acceptable environmental conditions to support operator access and egress and equipment functionality during the SBO and recovery period because the CRHS is consistent with the guidance of RG 1.155, Regulatory Position C.3.2.4, and remains operational. The CRHS is evaluated in Section 6.4 of this report. Therefore, the CRE room temperature would be expected to be maintained and would not challenge equipment operability or operator performance. After 72 hours, backup power is expected to be available, and the CRVS will then be utilized to provide air conditioning and building pressurization.

9.4.1.5 Conclusion

The staff evaluated the CRVS for the NuScale US460 design using the guidance of SRP Section 9.4.1. Based on the above evaluation, the staff finds that the CRVS design meets GDC 2, GDC 4, GDC 5, PDC 19, GDC 60, 10 CFR 20.1406, and 10 CFR 50.63.

9.4.2 Reactor Building and Spent Fuel Pool Area Ventilation System

9.4.2.1 Introduction

FSAR Section 9.4.2, "Reactor Building and Spent Fuel Pool Area Ventilation System," states that the Reactor Building HVAC system (RBVS) serves the Reactor Building (RXB), including the pool hall, which contains the reactor pool, refueling pool, spent fuel pool (SFP), dry dock, new fuel storage, and the NuScale Power Modules (NPMs) and their handling equipment. The RBVS is designed to maintain acceptable ambient conditions in the RXB to support personnel and equipment and to control airborne radioactivity in the area during normal operation and following events that have the potential to release radioactivity in the RXB, such as a fuel handling accident.

The RBVS includes four subsystems: the supply subsystem, the general area exhaust subsystem, the SFP exhaust subsystem, and the module-specific battery room, charger room, instrumentation and control room, and reserved area air handling units subsystem.

RBVS indoor design conditions are described in FSAR Table 9.4.2-1.

9.4.2.2 Summary of Application

FSAR Section 9.4.2, "Reactor Building and Spent Fuel Pool Area Ventilation System," provides information associated with this section. The RBVS serves no safety-related functions, is not credited for mitigation of design basis accidents, and has no safe-shutdown functions. The RBVS maintains the radiation exposure to operating and maintenance personnel as low as reasonably achievable.

In FSAR Section 9.4.2, the applicant states:

> During normal operation, the supply subsystem provides conditioned and filtered outside air to the RXB. The two exhaust subsystems deliver air to the plant exhaust stack for discharge from the plant. The SFP exhaust flows through a high-efficiency particulate air (HEPA) filter. In addition to air from the RXB, the RBVS general area exhaust subsystem receives exhaust air from the Radioactive Waste Building HVAC system (RWBVS).

The exhaust from the RBVS is monitored for radioactivity contamination. The RBVS includes air filtration and utilizes automatic realignment of the Spent Fuel Pool (SFP) area subsystem to limit release of airborne radioactivity contaminants to the environment. RBVS exhaust paths are monitored for radioactivity releases.

The RBVS moves air from areas that are not contaminated or that are expected to have low levels of contamination to areas that are likely to be more contaminated. The RBVS maintains air pressure in the RXB below that of the outside environment.

The general area exhaust subsystem collects exhaust air from each level of the RXB, including the battery rooms. The general area exhaust subsystem includes a standby fan, and each fan can be isolated from the others with dampers to allow inspection, testing, and maintenance with the remaining fans in operation. The general area exhaust subsystem maintains hydrogen concentrations in the battery rooms less than 1 percent by volume.

The SFP exhaust subsystem filters the exhaust air to reduce radioactive releases to the environment. The SFP exhaust subsystem includes a standby fan and filter set, each of which has isolation dampers that can be closed to isolate the equipment for inspection, testing, and maintenance with the remaining set in operation.

ITAAC: NuScale gave the ITAAC associated with FSAR Section 9.4, in SDAA Part 8, Section, Section 3.2, "Normal Control Room Heating Ventilation and Air Conditioning System." These ITAAC are evaluated in Section 14.3 of this SER.

9.4.2.3 Regulatory Basis

SRP Section 9.4.2, "Spent Fuel Pool Area Ventilation System," provides staff review guidance for the RBVS. The following are the relevant requirements of NRC regulations for this area of review:

- GDC 2, as it relates to system capability to withstand the effects of earthquakes.
- GDC 5, as it relates to shared SSCs among nuclear power units.
- GDC 60, as it relates to system capability to suitably control release of gaseous radioactive effluents to the environment.
- GDC 61, as it relates to providing appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment from the fuel storage facility under normal and postulated accident conditions.
- 10 CFR 20.1406, as it relates to the requirements that the facility design minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

9.4.2.4 Technical Evaluation

The staff reviewed the RBVS in accordance with the review procedure in SRP 9.4.2. The results of the staff's review are provided below.

9.4.2.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

FSAR Section 9.4.2.1 states that the RBVS serves no safety-related functions, is not risk-significant, is not credited for mitigation of design-basis accidents and has no safe-shutdown functions. The RBVS is not required to operate during or after a DBE.

The staff based its review of RBVS compliance with GDC 2 requirements, in part, on adherence to RG 1.29, Regulatory Position C.2. Based on its review of the NuScale US460 FSAR, the staff understands that the RBVS is a nonsafety-related, seismically Category III designed system.

FSAR Table 1.9-3 indicates the NuScale US460 RBVS design conforms with SRP 9.4.2 (acceptance criteria II.1), which refers to GDC 2 requirements.

RG 1.29, C.2 indicates the following:

> Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level or could result in incapacitating injury to occupants of the control room, should be designed and constructed so that the SSE would not cause such failure. Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.

The NRC staff notes that while most RBVS SSCs are seismic Category III, NuScale indicated in FSAR Section 9.4.2.3, that portions of the RBVS, in which structural failure could adversely affect the operability of seismic Category I SSCs, are designed to Seismic Category II standards. In addition, SSCs, including isolation dampers, that support the protection of mild environmental areas from high-energy line break events are designed to Seismic Category I standards.

The staff finds that the applicant followed the guidance of RG 1.29, Regulatory Position C.2, for nonsafety-related portions and therefore concludes that the RBVS complies with the requirements of GDC 2.

9.4.2.4.2 GDC 5, "Sharing of Structures, Systems, and Components"

Even though the RBVS is shared between multiple reactor modules, FSAR Section 9.4.2.3, states, "the RBVS does not have a function relative to shutting down an NPM or maintaining it in a safe shutdown condition. Operation of the RBVS does not interfere with the ability to operate or shut down an NPM."

According to GDC 5, SSCs important to safety shall not be shared among nuclear power plants unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

To satisfy GDC 5, the staff's review of the RBVS is to ensure that sharing of RBVS SSCs in multiple-NPM plants does not significantly impair their ability to perform their safety functions, including, in the event of an accident in one NPM, the ability of NuScale control room operators to shut down the remaining NPMs.

The NuScale US460 design does not credit any operator actions to mitigate DBEs. Specifically, in FSAR Section 15.0.0.6.4, NuScale stated the following:

> There are no operator actions credited in the evaluation of NuScale Power Plant US460 standard design DBEs. After a DBE, automated actions place the NPM in a safe state and it remains in the safe-state condition for at least 72 hours without operator action, even with assumed failures.

Based on the discussion above, the staff determined that sharing of RBVS SSCs in multiple-NPM plants does not significantly impair their ability to perform their safety functions, including, in the event of an accident in one NPM, an orderly shutdown and cooldown of the remaining

NPM(s). Therefore, the staff concludes that the design of the RBVS complies with the requirements of GDC 5.

9.4.2.4.3 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

GDC 60 requires that the nuclear power unit design include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences.

When normal power is available, exhaust from the RBVS is filtered. The general area RBVS exhaust is filtered by the RBVS general exhaust filter units, which utilize HEPA filters to remove contaminated particulate. The SFP exhaust is normally filtered through HEPA filters. Upon detection of radiation levels above the set design limit, the exhaust is designed to be diverted through HEPA filtration and charcoal adsorbers.

The staff finds that RG 1.140, Regulatory Positions C.2 and C.3, are applicable to the general area exhaust filter units and SFP exhaust charcoal and filter units. These units are not credited during a design-basis event (DBE).

According to FSAR Section 9.4.2.2.1, the general area exhaust filter units are designed, constructed and tested, to meet the applicable performance requirements of ASME N509, N510, "Testing of Nuclear Air Treatment Systems," and AG-1 in conformance with RG 1.140. The same FSAR section states that the SFP exhaust charcoal and filter units conform to RG 1.140.

The staff reviewed conformance to RG 1.140 as discussed below.

RG 1.140, Regulatory Position C.2 states that normal atmosphere cleanup systems should be designed in accordance with ASME N509 and ASME AG-1 with some modifications and supplements.

The staff reviewed the RBVS for satisfaction of Regulatory Positions C.2.1 through C.2.4 and concludes that NuScale's commitment to following RG 1.140, along with ASME AG-1 and ASME N509 provides assurance that the system design conforms to the guidance in RG 1.140, Regulatory Position C.2.

The RBVS general area exhaust filter units do not have charcoal adsorbers. The staff notes that RG 1.140, Section 3.1, indicates, "whenever a normal atmosphere cleanup system is designed to remove only particulate matter, a component for iodine adsorption need not be included." This reflects that gaseous iodine (elemental iodine and organic iodides) is not expected to be present in the air stream during normal operation and anticipated operational occurrences (startup, shutdown, and refueling). Therefore, charcoal adsorbers are not required in the RBVS general area exhaust filter units.

Other than standard components, the SFP exhaust charcoal and filter units are equipped with electric heating coils. As stated in RG 1.140, Section 3.1, heating coils may be used when the humidity is to be controlled before filtration. Since exhaust air from the SFP area may have higher humidity, installing heating coils is consistent with RG 1.140.

Based on the above finding, at the standard design approval stage, the staff finds that the provided RVBS components, HEPA filters, fans, ducts, dampers, instrumentation, and heating coils are sufficient to satisfy RG 1.140, Regulatory Position C.3.1.

FSAR Section 1.2.2.1, states the following:

> The RXB houses the NPMs and systems and components required for plant operation and shutdown. It is designed with considerations for the effects of aircraft impact, environmental conditions, postulated DBEs (internal and external), and design-basis threats. The RXB also provides radiation protection to plant operations and maintenance personnel.

Since the RBVS is located inside the RXB, this system is protected from missiles and designed to withstand the effects of natural phenomena.

FSAR Section 9.4.2.1, states, "[t]o maintain the radiation exposure to operating and maintenance personnel as low as reasonably achievable (ALARA), the RBVS is designed to facilitate maintenance, inspection, and testing in accordance with the guidance in RG 8.8."

Based on the above description, at the standard design approval stage, the staff finds that ALARA requirements for radiation exposure to operating and maintenance personnel are satisfied, and therefore, RG 1.140, Regulatory Position C.3.4, is satisfied.

The staff also reviewed piping and instrumentation diagrams, major component and system instrumentation diagrams, and inspection and testing for the system. Based on this review, the staff finds that the in-place testing (RG 1.140), the general area exhaust filter unit flow rate, and SFP exhaust charcoal and filter unit flow rate satisfy RG 1.140, Regulatory Position C.3.2; the instrumentation of each atmosphere cleanup system satisfies RG 1.140, Regulatory Position C.3.3; and the missile protection installed for the RXB outdoor air opening is sufficient to satisfy RG 1.140, Regulatory Position C.3.5.

Based on the above review, the staff concludes that the design of all HEPA filters and charcoal filter banks conforms to RG 1.140, Regulatory Position C.3.

Since the design of the RBVS conforms to RG 1.140 at the standard design approval stage, the staff concludes that the requirements of GDC 60 are satisfied.

9.4.2.4.4 GDC 61, "Fuel Storage and Handling and Radioactivity Control"

The staff reviewed the RBVS as it applies to RG 1.13, Regulatory Position C.4, which states that a controlled-leakage building should enclose the fuel to limit the potential release of radioactive iodine and other radioactive materials. If necessary to limit offsite dose consequences from a fuel handling accident or SFP boiling, the building should include an ESF filtration system that meets the guidelines outlined in RG 1.52.

According to FSAR Section 15.0.3.7.5, "Fuel Handling Accident," the activity released from the pool to the RXB is assumed to be instantaneously released to the environment without holdup or mitigation. Doses are determined at the exclusion area boundary, the low-population zone, and for personnel in the control room and TSC. That is, in calculating activity release to the exclusion area boundary, low-population zone, control room, and TSC during a fuel handling accident, NuScale does not take credit for the SFP charcoal and filter units. In FSAR Section 9.4.2.1, the applicant stated the RBVS is not safety-related. As such, the system design need not follow the guidelines outlined in RG 1.52. The staff agrees that the RBVS, including the SFP exhaust subsystem, can be considered not safety-related.

According to FSAR, Section 15.0.3.6.3, “Reactor Building Pool Boiling Radiological Consequences,” without available power, decay heat from the reactors and spent fuel would heat the pool water and could eventually cause the reactor pool to boil. It takes longer than 2.3 days for the pool to reach boiling after a loss of normal ac power event. However, if the pool were to boil, the dose would be less than 1 mSv (0.1 rem) total effective dose equivalent.

The SFP is located within the RXB, which is a controlled-leakage building. Exhaust from the SFP area passes through the RBVS exhaust charcoal and HEPA filter units. Upon detection of radiation within the SFP exhaust ductwork, three automatic actions occur: (1) RXB general exhaust is closed, (2) SFP exhaust air is diverted through HEPA filter and charcoal adsorber, and (3) supply fans reduce the capacity to accommodate the reduction in exhaust.

The staff finds that the fuel handling area HVAC system design complies with GDC 61 by conforming to the guidance in RG 1.13, Regulatory Position C.4.

9.4.2.4.5 10 CFR 20.1406, “Minimization of Contamination”

FSAR Section 12.3.3.3, states the following:

> During normal operation, the RBVS services the areas inside the RXB by providing conditioned and filtered outside air. The exhaust from the spent fuel pool area is filtered by a high-efficiency particulate air (HEPA) filter. If the spent fuel pool exhaust radiation monitors detect radioactivity above their setpoints, the exhaust flow from the spent fuel pool area is diverted to go through an additional HEPA filter and charcoal adsorbers. FSAR Section 9.4.2, Reactor Building HVAC, contains additional details.
>
> The dry dock area has dedicated exhaust vents to entrain airborne contamination that may result from air being exposed to NPM components during maintenance activities.
>
> Heating, ventilation, and air conditioning equipment drains are routed to the RWDS.
>
> The design provides adequate space for temporary shielding to minimize personnel exposures during maintenance of ventilation equipment, including filters, inspection, and testing. In addition, the filter units have design features that minimize the time required for filter changes.

Based on the above-described design considerations, the staff finds that RBVS ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts and the AFUs are designed with features that minimize the time required for filter changes, and therefore, constitutes compliance with 10 CFR 20.1406.

9.4.2.5 Conclusion

The staff evaluated the RBVS for the NuScale US460 design using the guidance of SRP Section 9.4.2. Based on the above evaluation, the staff finds that the RBVS design complies with GDC 2, GDC 5, GDC 60, GDC 61, and 10 CFR 20.1406.

9.4.3 Radioactive Waste Building Ventilation System

9.4.3.1 Introduction

The RWBVS is designed to support personnel access and equipment functions by maintaining a suitable operating environment in the RWB, including the waste management control room. The RWBVS maintains the temperature and humidity within ranges suitable for the comfort of personnel and to prevent the degradation of equipment during normal operation. The system directs airflow from areas of lower potential contamination to areas of higher potential contamination.

Exhaust from the RWBVS flows into the RBVS general area exhaust subsystem. The staff evaluates the RBVS in SER Section 9.4.2.

9.4.3.2 Summary of Application

FSAR Section 9.4.3, "Radioactive Waste Building Ventilation System," provides information associated with this section. The RBVS serves no safety-related functions, is not risk significant, is not credited for the mitigation of DBAs, and has no safe shutdown functions. The RWB has no safety-related components, and failure of the RWBVS to operate does not prevent SSCs from performing safety-related functions.

The RWBVS supply AHUs and provide filtered and heated or cooled air to various areas of the RWB. Dedicated units provide HVAC service to specific areas of the RWB, including the waste management control room, battery and battery charger rooms, and radiologically controlled area access control and hot shop areas. The AHUs and select fan coil units in the RWBVS have redundant units that automatically start if the running unit trips.

During normal operation, air enters the RWBVS through an intake located in an exterior wall of the RWB and then proceeds through a main supply AHU. The RWBVS main AHU supply airflow modulates to maintain the RWB at a negative pressure with respect to the outside air. Pressurization air ensures that air flows from clean spaces to potentially contaminated spaces. The RWBVS maintains the hydrogen concentration levels in the battery rooms below 1 percent by volume.

ITAAC: There are no proposed ITAAC related to the RWBVS.

Technical Specifications: No GTS are associated with the RWBVS.

9.4.3.3 Regulatory Basis

SRP Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System" provides staff review guidance for the RWBVS. The following are the relevant requirements of NRC regulations for this area of review:

- GDC 2, as it relates to the system's capability to withstand the effects of earthquakes.
- GDC 5, as it relates to the sharing of SSCs among multiple units not significantly impairing the SSC's ability to perform its safety function in the event one unit experiences an accident condition.

- GDC 60, as it relates to the capability of the system to suitably control the release of gaseous radioactive effluents to the environment.

- 10 CFR 20.1406, as it relates to the requirement that the facility design minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

9.4.3.4 Technical Evaluation

The staff reviewed the RWBVS in accordance with the review procedure in SRP Section 9.4.3. The results of the staff's review are provided below.

9.4.3.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

FSAR Section 9.4.3.1, states that the RWBVS is nonsafety-related, not risk-significant, does not perform a function to prevent a DBA, and has no safe shutdown functions. The RBVS is not required to operate during or after a DBE.

The staff based its review of RWBVS compliance with GDC 2 requirements, on adherence to RG 1.29, Regulatory Position C.2. Based on its review of the NuScale US460 FSAR, the staff understands that the RWBVS is a nonsafety-related, seismically Category III designed system. FSAR Table 1.9-3 indicates the NuScale US460 RWBVS design conforms with SRP Section 9.4.3 (acceptance criteria II.1), which refers to GDC 2 requirements.

RG 1.29, Regulatory Position C.2 indicates the following:

> Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room, should be designed and constructed so that the SSE would not cause such failure. Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.

The NRC staff notes that the RWBVS is fully contained in the RWB. There is no safety-related or seismic Category I equipment in the RWB and the failure of the RWBVS does not affect the performance of safety-related functions.

The staff finds that the guidance of RG 1.29, Regulatory Position C.2, for nonsafety-related portions has been followed and therefore concludes that the RWBVS complies with the requirements of GDC 2.

9.4.3.4.2 GDC 5, "Sharing of Structures, Systems, and Components"

FSAR Section 9.4.3.3, states, "the operation of the RWBVS does not affect the safe and orderly shutdown and cooldown of the NuScale Power Modules. The RWBVS does not have a function relative to shutting a module down or maintaining a module in a safe shutdown condition."

According to GDC 5, SSCs important to safety shall not be shared among nuclear power plants unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

To satisfy GDC 5, the staff's review of the RWBVS is to make sure that sharing of RWBVS SSCs in multiple-NPM plants does not significantly impair their ability to perform their safety functions, including, in the event of an accident in one NPM, the ability of NuScale US460 control room operators to shut down the remaining NPMs.

The NuScale US460 design does not credit any operator actions to mitigate DBEs. Specifically, in FSAR Section 15.0.0.6.4, the applicant stated the following:

> 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 it remains in the safe-state condition for at least 72 hours without operator action, even with assumed failures.

Based on the design-description above, the staff determined that the sharing of RWBVS SSCs in multiple-NPM plants does not significantly impair their ability to perform their safety functions, including, in the event of an accident in one NPM, an orderly shutdown and cooldown of the remaining NPM(s). Therefore, the staff concludes that the design of the RWBVS complies with the requirements of GDC 5.

9.4.3.4.3 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

GDC 60 requires that the nuclear power unit design include a means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including AOOs.

FSAR Section 9.4.3.3 state that the RWBVS design considered GDC 60. The RWBVS exhaust is monitored and filtered by the RBVS general exhaust filter units, which use HEPA filters to remove contaminated particulate. A monitor in the RBVS checks the plant exhaust stack discharge containing the RWBVS and RBVS exhaust air for radiation. These provisions ensure that the release of radioactive materials entrained in gaseous effluents during normal reactor operation, including AOOs, is controlled. FSAR Section 9.4.2, describes the RBVS.

Because the RBVS controls, monitors, and filters the RWBVS exhaust, the staff evaluates the RBVS in SER Section 9.4.2, which considers the requirements of GDC 60. Based on the evaluation in SER Section 9.4.2, the staff concludes that the RWBVS exhaust meets the requirements of GDC 60 as it relates to control of radioactive materials to the environment.

9.4.3.4.4 10 CFR 20.1406, "Minimization of Contamination"

FSAR Section 12.3.3.4, states the following:

> The RWBVS serves the RWB as a once-through system. Outside air is introduced by the main supply air handling unit (AHU) and is exhausted through the RBVS exhaust system. The main supply AHU contains both low and medium efficiency outside air filters. Supply air from the main RWBVS is distributed throughout the RWB. Exhaust air is collected and conveyed to the RBVS general area exhaust subsystem and exhausted through the main stack. The RWBVS maintains airflow from areas of lesser potential contamination to areas of greater potential contamination. The RWBVS also maintains

the RWB atmosphere at a slight negative pressure with respect to the outside. Section 9.4.3, Radioactive Waste Building HVAC, contains additional details.

The design will ensure that exhausted air from the RWBVS flows into the RBVS general area exhaust subsystem. The staff evaluated the RBVS in Section 9.4.2 of this report and finds that RBVS ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts and the AFUs are designed with features that minimize the time required for filter changes. Therefore, the staff concludes that the above-described design considerations constitute compliance with 10 CFR 20.1406.

9.4.3.5 Initial Test Program

Information regarding RWBVS initial testing is provided in FSAR, Section 14.2 (Test # 18).

The ITP is evaluated in Section 14.2 of this SER.

9.4.3.6 Conclusion

The staff evaluated the RWBVS for the NuScale US460 design using the guidance of SRP Section 9.4.3. Based on the above evaluation, the staff finds that the RWBVS meets GDC 2, GDC 5, GDC 60, and 10 CFR 20.1406.

9.4.4 Turbine Building Ventilation System

9.4.4.1 Introduction

The turbine building HVAC system (TBVS) is designed to support personnel access and equipment functions by maintaining a suitable environment in the TGB. The TBVS maintains environmental conditions within ranges suitable for personnel occupancy and equipment reliability.

9.4.4.2 Summary of Application

FSAR Section 9.4.4, “Turbine Building Ventilation System,” provides information associated with this section. The TBVS serves no safety-related functions, is not risk-significant, is not credited for the mitigation of design-basis accidents and has no safe shutdown functions. There are no safety-related components in the TGB, and failure of the TBVS to operate does not prevent SSCs from performing safety-related functions.

The TBVS serves the TGB including the turbine hall, battery room, battery charger room, and maintenance room.

An exhaust fan in the battery room maintains the hydrogen concentration in the room to less than 1 percent by volume.

The turbine building is not directly connected to the RXB, the RWB, or any other areas that may contain radioactive contaminants. The TBVS is independent of other HVAC systems and is not directly connected to other SSCs that may contain radioactive contaminants. The staff notes that FSAR Section 9.4.4, does not indicate that filtration or adsorption systems are present anywhere in the TBVS.

ITAAC: There are no proposed ITAAC related to the TBVS.

9.4.4.3 Regulatory Basis

SRP Section 9.4.4, "Turbine Area Ventilation System" provides staff review guidance for the RWBVS. The following are the relevant NRC regulatory requirements for this area of review:

- GDC 2, as it relates to system capability to withstand the effects of earthquakes

- GDC 5, as it relates to the sharing of SSCs in multiple unit plants and the impact on the ability of the SSCs to perform their safety function in the event one unit experiences an accident condition

- GDC 60, as it relates to the capability of the system to suitably control the release of gaseous radioactive effluents to the environment

9.4.4.4 Technical Evaluation

The staff reviewed the TBVS in accordance with the review procedure in SRP 9.4.4. The results of the staff's review are provided below.

9.4.4.4.1 GDC 2, "Design Bases for Protection against Natural Phenomena"

The staff based its review of TBVS compliance with GDC 2 requirements on adherence to RG 1.29, Regulatory Position C.2. Based on its review of the NuScale US460 FSAR, the staff understands that the TBVS is a nonsafety-related, seismically Category III designed system.

RG 1.29, Regulatory Position C.2, indicates the following:

> Those portions of SSCs whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room, should be designed and constructed so that the SSE would not cause such failure. Wherever practical, structures and equipment whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.

The NRC staff notes that the TBVS is fully contained in the TGB. There are no safety-related SSCs in the TGB, therefore there are no safety-related SSCs affected by natural phenomena such as earthquakes. Failure of the TBVS will not affect safety-related SSCs.

The staff finds that the applicant followed the guidance of RG 1.29, Regulatory Position C.2, for nonsafety-related portions and therefore concludes that the TBVS complies with the requirements of GDC 2.

9.4.4.4.2 GDC 5, "Sharing of Structures, Systems, and Components"

According to GDC 5, SSCs important to safety shall not be shared among nuclear power plants unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

The staff's review of the TBVS determined that sharing of TBVS SSCs in multiple-NPM plants does not significantly impair their ability to perform their safety functions, including, in the event of an accident in one NPM, the ability of NuScale US460 control room operators to shut down the remaining NPMs. Therefore, the staff concludes that the design of the TBVS complies with the requirements of GDC 5.

9.4.4.4.3 GDC 60, "Control of Releases of Radioactive Materials to the Environment"

GDC 60 requires that the nuclear power plant design include a means for suitable control of the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including AOOs.

FSAR Section 9.4.4.3 states that the TBVS design considered GDC 60. During normal operation, radioactive material is not expected to be present in the TGB and therefore, the TBVS does not include radioactivity monitoring or filtration. The only potential source of radioactive material in the TGB is from a postulated steam generator tube failure. FSAR Section 11.5 provides information on radiation monitors in the main steam system and the condensate polisher resin regeneration system that monitor the secondary system, and therefore the TGB, for contamination.

Since the design of the TBVS is not relied on to control airborne radioactivity concentrations in the Turbine Building and gaseous effluents during normal operations (including AOOs), and after accidents that result in a radioactive material release, the staff concludes that the requirements of GDC 60 are satisfied.

9.4.4.5 Conclusion

The staff evaluated the TBVS for the NuScale US460 design using the guidance of SRP Section 9.4.4. Based on the above evaluation, the staff finds that the TBVS meets GDC 2, GDC 5, and GDC 60.

9.5 <u>Other Process Auxiliaries</u>

9.5.1 Fire Protection Program

9.5.1.1 Introduction

In FSAR Section 9.5, NuScale stated that the primary objectives of the fire protection program (FPP) are to minimize both the probability of occurrence and the consequences of fire. Further, NuScale stated that to meet these objectives, the FPP is designed to provide reasonable assurance, through defense in depth, that a fire will not prevent the necessary safe-shutdown functions from being performed and that radioactive releases or hazardous chemical exposure to personnel and to the environment are minimized. The FPP consists of the integrated effort involving components, procedures, analyses, and personnel used in defining and performing activities of fire protection. It includes system and facility design, fire prevention, fire detection, annunciation, confinement, suppression, administrative controls, fire brigade organization, inspection and maintenance, training, quality assurance (QA), and testing. Further, the fire protection system (FPS) and fire detection system are part of the FPP and include the fire detection, notification, and suppression systems, as designed, installed, and maintained in accordance with applicable industry codes and standards. FSAR Figures 9.5.1 1 and 9.5.1 2, show the FPS, fire water supply and fire pump arrangement and the fire main loop in the yard. The FPS design uses National Fire Protection Association (NFPA) 804, "Standard for Fire

Protection for Advanced Light Water Reactor Electric Generating Plants," 2020 Edition, and other applicable industry codes and standards included in FSAR Table 9.5.1 1.

9.5.1.2 Summary of Application

The FPP uses the guidance of Regulatory Guide (RG) 1.189, "Fire Protection for Nuclear Power Plants," Revision 4, and the requirements of NFPA 804, "Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants," 2020 Edition. The design basis and description of the FPP are provided in FSAR Section 9.5.1.1. The structures, systems, and components (SSCs) associated with the FPS are not safety-related, and do not have a quality group classification. The FPS classification is Seismic Category III but has unique seismic requirements. Portions of the fire protection water supply standpipe systems in the Reactor Building (RXB) remain functional following a safe shutdown earthquake (SSE). The standpipe system piping in the RXB up to and including the isolation valves supplying the sprinklers conform to the American Society of Mechanical Engineers (ASME) code for pressure piping B31.1, "Power Piping," 2018 Edition and are seismically analyzed under SSE inputs (i.e., Seismic Category I). Additionally, components and associated supports with failures that could prevent a safety-related function from being performed conform, at a minimum, to Seismic Category II standards. FSAR Table 9.5.1-3 identifies SSC classifications for FPS in the RXB, Control Building (CRB), and Radioactive Waste Building (RWB).

In addition, FSAR Table 9.5.1-2, "NuScale Fire Protection Design Compliance with RG 1.189," is a point-by-point comparison of the conformance of the NuScale US460 standard design with the guidelines of RG 1.189, Revision 4. Many tasks identified in this table have been assigned to a future combined license (COL) applicant to be addressed in the combined license application (COLA). Footnote 2 of FSAR Section 9.5.1, Table 9.5.1-2 identifies the term, "Applicant – The Applicant/Licensee will (also) address the subject Regulatory Position," and the first Comment in Table 9.5.1-2, in part, states, "Applicant will be required to develop and maintain the site-specific elements of the FPP."

FSAR Chapter 9, Appendix 9A, "Fire Hazard Analysis," presents the fire hazards analysis (FHA) and the fire safe shutdown plan for the NuScale Power Plant US460 standard design. The fire hazards analysis provides information on the fire hazards by evaluating the potential for the occurrence of fire within the NuScale Power Plant US460 standard design and demonstrating that the plant maintains the capability to perform safe shutdown functions and minimize the release of radioactive material to the environment in the event of a fire. The FHA demonstrates that the plant maintains the ability to perform safe shutdown functions and to minimize radioactive material releases to the environment in the event of a fire. In the event of a fire in the main control room (MCR), the operators trip the reactors, initiate decay heat removal and initiate containment isolation prior to evacuating the MCR. These actions result in passive cooling that achieves and maintains the modules in a safe shutdown condition. Plant operators can also place the reactors in safe shutdown from outside the MCR in the instrumentation and control (I&C) equipment rooms within the reactor building. Following shutdown and initiation of passive cooling, the design does not rely on operator action, instrumentation, or controls outside of the MCR to maintain a safe stable shutdown condition. There are two MCR isolation switches for each NuScale Power Module (NPM) located outside the MCR that when repositioned isolate the module protection system (MPS) manual actuation switches, override switches and enable non-safety control switches for each NPMs MPS in the MCR to prevent spurious actuation of equipment due to fire damage.

In the event of a fire in the MCR, the operators evacuate the MCR. There are two MCR isolation switches for each NPM that when repositioned, isolate the MPS manual actuation switches and the enable non-safety switch for each NPM's module protection system in the MCR to prevent spurious actuation of equipment due to fire damage. An alarm is annunciated in the MCR when the MCR hard-wired switches are isolated using the MCR isolation switches located outside the MCR. The alternative shutdown capability (I&C equipment room) is independent of specific fire areas and accommodates post-fire conditions when offsite power is available and when offsite power is not available for 72 hours, dependent on the conditions described in the FHA as described in Section 9.5.1, Appendix 9A. The I&C equipment rooms are seismically qualified and located in separate fire zones. Division I MPS and neutron monitoring station (NMS) equipment are located in a different room than Division II MPS and NMS equipment. In addition, the MPS manual isolation switches are mounted in a Seismic Category I enclosure to allow them to remain functional following an earthquake. Controls are available outside the MCR in the associated I&C equipment rooms that provide the capability to trip the reactors, to initiate decay heat removal system (DHRS), and to initiate containment isolation, which initiates passive cooling, and places and maintains the NPMs in safe shutdown. The alternate operator workstations provide non-safety-related human-system interfaces (HSIs) and direct readings of process variables that allow operators to monitor the NPMs. The MPS equipment and cable routing is designed to meet the separation requirements of Institute of Electrical and Electronics Engineers (IEEE) 384-1992, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” as endorsed by RG 1.75, Revision 3, “Criteria for Independence of Electrical Safety Systems.” These design attributes also provide separate rooms and cable runs to prevent a fire or explosion from affecting more than one division of MPS and NMS-excore equipment.

FSAR Section 14.2 describes the verification programs included in the initial test programs. The applicant gave the ITAAC associated with FSAR Section 9.5.1, in SDAA Part 8, Section 3.7, “Fire Protection System.” These ITAAC are evaluated in Section 14.3 of this SER.

9.5.1.3 Regulatory Basis

SRP Section 9.5.1.1, “Fire Protection Program,” issued March 2009, gives the relevant regulatory requirements for this area of review and the associated acceptance criteria, as summarized below, as well as the review interfaces with other SRP sections.

- 10 CFR 50.48(a)(4) requires, in part, that each applicant for a design approval under 10 CFR Part 52 must have a description and analysis of the fire protection design features for the standard plant necessary to demonstrate compliance with GDC 3, “Fire Protection.”

- GDC 3, as it relates to the following:

    - SSCs important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

    - Noncombustible and heat-resistant materials are used wherever practical throughout the unit.

    - Fire detection and fighting systems of appropriate capacity and capability are provided and are designed to minimize the adverse effects of fires on SSCs.

- Firefighting systems are designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs.

- GDC 5, as it applies to shared SSCs important to safety shall not be shared among nuclear power units it can be shown that such shearing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- GDC 19, as it applies to the MCR 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.

  NuScale has provided rationale to support that an exemption request from GDC 19 to implement a design-specific Principal Design Criterion (PDC) 19 that maintains the reactor in a safe condition would be justified in lieu of the requirements for “design capability for prompt hot shutdown” and “potential capability for subsequent cold shutdown” as specified in GDC 19. The staff’s evaluation of the rationale that supports PDC 19 is provided in Section 6.4 of this report.

- GDC 23, “Protection system failure modes,” as it applies to the protection system being designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis, if the plant experiences adverse environments such as from a fire.

In accordance with the Regulatory Guide 1.189, fire protection for nuclear power plants uses the defense-in-depth approach to achieve the required degree of reactor safety by using echelons of administrative controls, FPSs and features, and post-fire safe-shutdown capability. The defense-in-depth approach includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.

The defense-in-depth approach uses the design and operation of nuclear power plants in a manner that prevents and mitigates accidents that release radiation or hazardous materials. The key is to create multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is relied upon exclusively.

RG 1.189, Revision 4 provides guidance and acceptance criteria for one acceptable approach for an FPP that meets the regulatory requirements in 10 CFR 50.48.

In addition to the regulatory requirements and guidance provided above, SRP Section 9.5.1.1 provides enhanced fire protection criteria for new reactor designs as documented in the following:

- SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” dated January 12, 1990 (ADAMS Accession No. ML003707849)

- SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 2, 1993 (ADAMS Accession No. ML003708021)

- SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” March 28, 1994 (ADAMS Accession No. ML003708068)

SECY-90-016 provides enhanced fire protection criteria for evolutionary LWRs. SECY-93-087 recommends that the enhanced criteria be extended to include passive reactor designs. The Commission approved SECY-90-016 and SECY-93-087 in staff requirements memoranda dated June 26, 1990, and July 21, 1993 (ADAMS Accession Nos. ML003707885 and ML003708056, respectively). SECY-94-084, in part, provides criteria defining safe-shutdown conditions for passive LWR designs. According to FSAR Section 9A.3.1, the NuScale design meets the intent of these SECYs through compliance with RG 1.189.

9.5.1.4 Technical Evaluation

9.5.1.4.1 Fire Hazards Analysis

The staff reviewed NuScale’s FSAR Section 9.5.1, and the FHA, as provided in Appendix 9A of the FSAR, to ensure that NuScale has demonstrated that the NuScale US460 standard design will have the ability to perform safe shutdown functions and minimize radioactive material release to the environment in the event of a fire in conformance with RG 1.189, Revision 4. FHA describes the fire hazards and provides the results for plant fire areas. NuScale stated that the NuScale US460 standard design FHA establishes and evaluates all fire areas for RXB, CRB, RWB. No other structures in the plant contain equipment necessary for safe shutdown or have the potential for a radiological release. FSAR, Chapter 1, “Introduction and General Description of the Plant,” identify fire areas and indicate fire barriers and ratings in RXB, CRB, and RWB as follows:

- Figure 1.2-8: Reactor Building 25’-0” Elevation,
- Figure 1.2-9: Reactor Building 40’-0” Elevation,
- Figure 1.2-10: Reactor Building 55’-0" Elevation,
- Figure 1.2-11: Reactor Building 70’-0" Elevation,
- Figure 1.2-12: Reactor Building 85’-0" Elevation,
- Figure 1.2-13: Reactor Building 100’-0" Elevation,
- Figure 1.2-14: Reactor Building 126’-0" Elevation,
- Figure 1.2-15: Reactor Building 146’-6” Elevation,
- Figure 1.2-18: Control Building 100’-0" Elevation,
- Figure 1.2-19: Control Building 125’-0” Elevation,
- Figure 1.2-22: Radioactive Waste Building 70’-0" Elevation,
- Figure 1.2-23: Radioactive Waste Building 82'-0" Elevation,
- Figure 1.2-24: Radioactive Waste Building 100'-0" Elevation,
- Figure 1.2-25: Radioactive Waste Building 119’-0” Elevation,
- Figure 1.2-26: Radioactive Waste Building 145’-0” Elevation.

- Table 9A-1, “Fire Hazard Analysis Elements and Attributes,” identifies the limitations of FHA in plant fire areas.
- Table 9A-8, “Reactor Building Fire Areas,”
- Table 9A-9, “Radioactive Waste Building,” Fire Areas, and
- Table 9A-10, “Control Building Fire Areas,” identify the fire areas evaluated by the FHA in Appendix 9A.

The purpose of the FHA is as follows:

- to consider in-situ and transient fire hazards.
- to determine the effects of a fire in any location in the plant on the ability to safely shut down the reactor or to minimize and control the release of radioactivity to the environment.
- to specify measures for fire prevention, detection, suppression, and containment for each fire area containing safety-related and risk-significant SSC, in accordance with the fire protection regulations and guidance.

The FHA is based on an assessment of every fire area, using the fire protection defense-in-depth approach from RG 1.189, Revision 4. The FHA is based on the introduction of transient combustible to any area of the plant, subject to administrative controls. The FHA approach used by the NuScale US460 standard design include following:

- Physical construction and layout of the buildings and equipment, including fire areas and the fire ratings of area boundaries.
- Inventory of the principal combustibles within each fire subdivision.
- Description of the fire protection equipment, including detection and alarm systems, and manual and automatic extinguishing systems.
- Analysis of the postulated fire in each fire area, including its effect on safe shutdown equipment, assuming automatic and manual fire protection equipment does not function.
- Analysis of the potential effects of a fire on life safety, release of contamination, impairment of operations, and property loss, assuming the operation of installed fire-extinguishing equipment.
- Analysis of the potential effects of an uncontained fire that may cause other problems not related to safe shutdown, such as a release of contamination or impairment of operations.
- Analysis of the post-fire recovery potential.
- Analysis of the protection of nuclear safety-related systems and components from the inadvertent actuation of or breaks in an FPS.

NuScale defined a fire area as a portion of a building or plant that is separated from other areas by 3-hour-rated fire barriers (i.e., walls, floors, and ceilings) which contain the effects of a fire to within a single fire area. Fire-rated barriers include components such as reinforced concrete walls, floors, beams, joists, and columns. All penetrations in fire-rated barriers are protected with 3-hour-rated components such as penetration seals, fire doors, and fire dampers.

NuScale also defined a fire zone as a division of a fire area, typically based on FPSs and structural features in the fire zone that provide an appropriate level of protection for the associated hazards. A fire zone is not necessarily isolated by complete fire barriers or fire-rated construction. A fire area may be divided into fire zones when it is not practicable or desirable to divide a fire area into multiple fire areas because of the plant design and layout such as inside containment.

NuScale FSAR states that the combustible loading, both in-situ and transient, in a fire area or fire zone are quantified to determine an equivalent fire severity in units of time.

FSAR Section 9A.3.2.1, “In-Situ Combustibles and Ignition Sources,” of the FHA identifies in-situ hazards and addresses fire protection features of the facility, including fire separation used to protect against the in-situ hazards. The plant, to the extent practicable, is to be built of non-combustible or limited-combustible materials. FSAR Table 9A-2, “In-Situ Combustible Material Classification,” and Table 9A-3, “In-Situ Ignition Sources,” identify the types of combustibles and ignition sources located in specific areas throughout the plant. Those listed are representative of the hazards and are not a comprehensive list. Self-ignition of electrical cables that are qualified in accordance with a nationally recognized standard fire test methodology, such as IEEE 1202 (Table 9A-2), is not credible as long as there are properly sized protective devices (fuses or circuit breakers) and there are cables appropriately derated for ampacity. Therefore, there are no postulated self-ignited cable fires from in-situ ignition sources.

FSAR Section 9A.3.2.2, “Transient Combustibles,” of the FHA states:

> Transient combustibles are those fire hazards that are not commonly found in a space, room, or other location, but may be present in various quantities due to movement of materials, temporary storage, testing, maintenance, or other conditions of normal operation, such as refueling, maintenance, or plant modifications. Fire Protection Program features control transient combustibles. FHA Table 9A-4, Typical Transient Combustibles, lists typical transient combustibles.
>
> Construction materials may involve assorted materials related to construction or installation of system(s) for additional modules. This construction may occur while one or more modules are operating. Dedicated operating areas may contain construction materials, but only before operation of that area. For example, an I&C equipment room could contain construction materials before the MPS equipment in that room operates, but once the MPS equipment is operational, construction materials are not expected to be present. Some areas that contain shared equipment contain redundant safe shutdown equipment in a different area. System construction or installation includes connections, terminations, and importation of relatively small equipment because the walls, floors, and ceilings are in place, preventing the importation of large equipment skids and tanks. The building design accommodates the NPM passage. Construction materials are typically indistinguishable from transient combustibles associated with repair and maintenance of plant systems.

FSAR Section 9A.3.2.3, “Transient Ignition Sources,” of the FHA states that transient ignition sources may be the result of maintenance, repair, or renovation work in the area that results in a temporary source that is brought into the fire area. Table 9A-5, “Transient Ignition Sources,” lists the transient ignition sources considered in the FHA.

FSAR Section 9A.3.3, “External Exposure Hazards,” of the FHA states that the protection of the RXB, CRB, and RWB from the effects of external fires from adjacent buildings, is in accordance with NFPA 80A, “Recommended Practice for Protection of Buildings from Exterior Fire Exposures,” 2017 Edition. Site-specific exposure hazards have not been considered in this analysis. Exposure hazards are plant-specific and vary depending on the final location of the plant and arrangement of the nearby structures and support buildings.

FSAR Section 9A.6.3, "Systems Required for Fire Safe Shutdown," of the FHA stated that although systems credited for safe shutdown following a fire event do not require a safety-related designation, safe shutdown uses only safety-related equipment. Defense-in-depth credits non-safety-related equipment, with specific examples in Section 9A.6.4. Safe shutdown following a fire is achieved through the successful operation of one division of each of the systems (MPS, DHRS, emergency core cooling system (ECCS) and reactor coolant system (CRS)) listed in Table 9A-7, "Safe Shutdown Plant Functions." Other systems (RCS, ultimate heat sink (UHS), control rod assembly, and control drive system (CRDS)) included in Table 9A-7 do not fail in a fire because they are fail safe, passive systems, or use passive components. System functions have also been identified.

FSAR Section 9A.5, "Fire Hazards Analysis," of the FHA addresses the NuScale US460 standard design specific fire areas in RXB, CRB, and RWB to determine combustibles present, to identify the consequences of fires, and to evaluate the FPSs and features. FHA Tables 9A-8, "Reactor Building Fire Areas," Table 9A-9, "Radioactive Waste Building Fire Areas," and Table 9A-10, "Control Building Fire Areas," identify fires areas within the plant. The staff's review considered the identification and adequacy of fire protection systems (e.g., detection, suppression, barriers, and separation of redundant safe shutdown systems), the credited safe shutdown paths and supporting systems, the analysis of fire scenarios and their potential impact on safety-related systems and structures, and the demonstration that sufficient defense-in-depth is maintained through fire prevention, control, and mitigation strategies.

Since section 9A.5 of the FSAR goes into detail on how the NuScale US460 standard design will have the ability to perform safe shutdown functions and minimize radioactive material release to the environment in the event of a fire in conformance with the fire protection regulations and consistent with the guidance in RG 1.189, Revision 4, the staff finds the FHA acceptable.

9.5.1.4.2 GDC 3, "Fire Protection"

The staff reviewed NuScale's FPP against the four requirements described in GDC 3 as follows: SSCs important to safety shall be designed and located to minimize the probability and effect of fire and explosions; noncombustible and heat resistant materials shall be used wherever practical; fire detection and fighting systems of appropriate capacity and capability shall be provided; and assurance is provided that rupture or inadvertent operation of firefighting systems does not impair the safety capability of these SSCs.

9.5.1.4.2.1 Minimizing the Probability and Effect of Fires and Explosions

The staff reviewed FSAR Section 9.5.1.1, "Design Bases", to ensure that it conforms to GDC 3 as it relates to the SSCs important to safety that are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Appendix 9A, FHA provides a fire safe-shutdown plan that demonstrates that the NuScale US460 standard design conforms to GDC 3. A method used to meet GDC 3 is to compartmentalize the buildings that contain equipment with safety-related or risk-significant functions into separate fire areas. Compartmentalization is achieved by using properly rated fire barriers, fire doors, fire dampers, and fire barrier penetration seals to prevent the spread of fire between fire areas. The FHA defines the locations of fire areas and fire barriers. NuScale US460 standard design fire protection building design and passive fire protection features conforms to RG 1.189, Revision 4, Regulatory Positions 4.1.1, 4.1.1.1, 4.1.1.2, 4.1.2, 4.1.2.1, 4.1.2.3, 4.1.3.1, and 4.1.3.2, as listed in FSAR Table 9.5.1-2.

NuScale provided an FHA for the RXB, CRB, and RWB. The NuScale US460 standard design includes only these three buildings. NuScale stated that no other structures in the plant contain equipment necessary for safe shutdown or have the potential for a radiological release.

The FHA demonstrates how fire areas meet the following objectives for fire protection defense-in-depth:

- Prevent fires from starting.
- Promptly detect, rapidly control, and extinguish fires that occur.
- Provide protection for SSCs required for safe shutdown so that a fire that is not promptly extinguished by fire suppression activities does not prevent the safe shutdown of the plant.

NuScale stated that structural fire barriers separate redundant cables and equipment required for safe shutdown following a fire. Structural fire barriers include walls, floors, and supports, as well as beams, joists, columns, penetration seals, fire doors, and fire dampers. Structural steel forming part of or supporting fire barriers conforms to RG 1.189, Revision 4, Regulatory Position 4.2.2 as listed in Table 9.5.1-2 of FSAR, Chapter 9.

Door openings are protected to maintain the fire rating of the barrier. An independent testing laboratory has tested the fire doors to meet the desired fire resistance characteristics. NFPA 80, “Standard for Fire Doors and Other Opening Protectives,” provides requirements for fire doors (fire barriers openings). In accordance with NFPA 80, NuScale US460 standard design conforms to RG 1.189, Revision 4, Regulatory Position 4.2.1.2 as listed in Table 9.5.1-2 of FSAR, Chapter 9.

Fire dampers in the plant ventilation openings through fire barriers seal off the opening in the event of a fire. The fire-resistance rating of fire dampers is equivalent to the rating of the fire barrier in which it is installed. NFPA 90A, “Standard for the Installation of Air Conditioning and Ventilating Systems,” provides guidelines for installation of fire dampers. In addition, Underwriters Laboratories (UL) Standard 555, “Fire Dampers,” provides criteria for the design, fabrication, and testing of fire dampers. Based on the above information, in accordance with NFPA 90A and UL 555, NuScale US460 standard design conforms to RG 1.189, Revision 4, Regulatory Position 4.2.1.3 as listed in Table 9.5.1-2 of FSAR, Chapter 9.

Openings in fire barriers for pipes, conduits, and cable trays that separate fire areas are sealed (penetration seals) to provide a fire resistance rating equivalent to the fire barrier rating. Penetration seals are tested in the configuration in which they are intended to be used or in a configuration that bounds the intended installation. Testing is conducted in accordance with NFPA 251, “Standard Methods of Tests of Fire Endurance of Building Construction and Materials,” and American Society for Testing and Materials (ASTM) E-119, “Standard Test Methods for Fire Tests of Building Construction and Materials.” Additional guidance documents include ASTM E814, “Standard Test Method for Fire Tests of -Through Penetration Fire Stops,” and Institute of IEEE 634, “IEEE Standard Cable Penetration Fire Stop Qualification Test.”

As part of the FHA, NuScale provided information on the NFPA hazard classification; the expected in situ combustibles and ignition sources; the expected transient combustible and ignition sources; installed fire protection detection and suppression systems; and the impact of fire and smoke on the emergency response, postfire operations, and potential for radiological releases for each fire area. Based on the FSAR and above information, the staff concludes, NuScale US460 standard design conforms to NFPA 251, IEEE standards, and RG 1.189, Revision 4, Regulatory Position 4.2.1.4 as listed in Table 9.5.1-2 of FSAR, Chapter 9.

Switchgear rooms containing equipment with safety-related or risk-significant functions are separated from the remainder of the plant by barriers having a three-hour fire rating. Redundant switchgear safety divisions are separated from each other by three-hour fire rated barriers.

Automatic fire suppression for switchgear rooms is based on the FHA. Fire hose stations and portable fire extinguishers are outside the area and are readily available. Adequate floor drainage removes water from firefighting activities and suppression system actuation. The NRC staff notes that NuScale US460 standard design does not contain a cable spreading room.

NuScale FSAR states, plant battery rooms associated with the redundant separation trains are separated from each other and other areas of the plant by fire barriers having a minimum three-hour fire rating. Battery rooms housing batteries that produce flammable off-gases have ventilation systems designed to maintain the concentration of the gas as defined in Table 9.5.1-2, RG 1.189, Revision 4, Regulatory Position 6.1.7. Automatic fire detection alarms annunciate in the control room and alarm locally. Loss of ventilation alarms are located in the control room. Battery rooms do not contain direct current (DC) switchgear or inverters. Standpipes, hose stations, and portable extinguishers are readily available outside the room.

NuScale provided information on how safe shutdown is achieved following a fire in a single fire area where the placement of redundant equipment required for safe shutdown cannot be avoided. RG 1.189, Revision 4, Regulatory Position 8.2, “Enhanced Fire Protection Criteria,” describes the control room and the reactor containment building as such areas. Based on the above information, the staff concludes, NuScale US460 standard design conforms to RG 1.189, Revision 4, Regulatory Position 8.2 as listed in Table 9.5.1-2 of FSAR, Chapter 9.

NuScale stated that the MCR has redundant equipment necessary for safe shutdown. This configuration is acceptable, as a fire in the MCR is unlikely, and a fire that does occur should be promptly detected and extinguished. If a fire requires an evacuation of the MCR, the controls in the MCR are isolated, and safe shutdown can be monitored from the remote shutdown room.

NuScale stated that the RXB houses the NPM and maintains them partially immersed in the UHS. Fire suppression or detection is not provided inside containment. The containment interior remains inaccessible while operating. During operations, the containment for each NPM is partially immersed in the UHS pool and maintained at vacuum conditions by the containment evacuation system. The evacuated state provides insufficient oxygen to sustain combustion in the unlikely event that combustion initiation conditions occur.

Electrical conductors within the containment vessel (CNV) are noncombustible or routed in conduit, and result in no intervening combustible loading for an exposure fire impacting other cable or components in the containment. The reactor coolant system relies on natural circulation, and therefore, there are no pumps with associated lube oil systems located inside containment.

In addition, fire suppression or detection is not necessary during refueling outages. During a plant shutdown for refueling, the containment floods at the same time containment pressure increases to atmospheric. The fire safe shutdown equipment in the CNV is the ECCS valves, the control drive mechanisms, and safety-related pressurizer heaters.

NuScale stated that the US460 standard design includes a fire-rated, 3-hour barrier at the top of each module under the bioshield. This barrier either fully encloses the area or prevents fire spread through other means, creating a separate fire area to ensure module separation.

The top of the module area is inaccessible during reactor operation, which precludes introduction of transient combustibles at any time that the reactor is operating. After reactor shutdown is complete, removal of the bioshield is permissible. Once the bioshield is removed, transient combustibles could be introduced; at the same time, manual fire suppression is available in the area at the top of the module. The applicant has taken the following measures under the bioshield to ensure that one division of safe-shutdown equipment remains available:

- Maintain divisional separation to the extent practicable given the physical restraints of the area. Safe-shutdown SSCs are safety-related; at a minimum, the plant follows the separation guidance of RG 1.75, Revision 3, February 2005.
- Eliminate in situ combustibles except for fire-rated cable inside steel conduit. Cable not in conduit is noncombustible.
- Use redundant, hydraulically operated valves for safe shutdown that are not dependent on power cables in the bioshield fire area.
- Divisionally separated hydraulic control units are located outside of the bioshield fire area in separate 3-hour-rated structural fire areas.
- Provide smoke detection in the ventilation exhaust from each individual fire area enclosed by the bioshield.

The applicant provided a fire safe-shutdown-plan in FHA Appendix 9A, Section 9A.7, "Safe Shutdown Plant Functions," and identified safe-shutdown components for which fire-induced circuit faults could directly or indirectly prevent safe shutdown. FHA Table 9A-11, "Multiple Spurious Operations Challenging Safe Shutdown," identifies potential scenarios where multiple spurious operations could occur simultaneously and challenge the plant's ability to achieve safe shutdown. It highlights combinations of failures or malfunctions that could impact key systems and functions, requiring mitigation strategies to ensure the safe shutdown process is not compromised. The NuScale stated that the FHA and fire safe-shutdown plan address possible fire-induced failures, including multiple spurious actuations. Consistent with RG 1.189, Revision 4, Regulatory Position 5.3.1.1, the applicant used the methodology described in Chapter 4 of Nuclear Energy Institute (NEI) 00-01, "Guidance for Post Fire Safe Shutdown Circuit Analysis," which relies on the expert panel process and the generic list of multiple spurious operations (MSOs) in Appendix G to NEI 00-01, for the analysis of MSOs for protection of SSCs important to safe shutdown. In the NuScale internal review process the expert panel reviewed the safe-shutdown equipment list, plant drawings, and other plant-specific documents to develop a list of possible plant-specific MSOs. The pressurized water reactor (PWR) generic MSO list in Appendix G to NEI 00-01 was used as guidance, and a potential MSO scenario encountered during the review of plant documents was considered. Other possible scenarios were identified and, if determined to be applicable, were added to the multiple actuation scenarios list. Further, the staff concludes, NuScale US460 standard design complies with the RG 1.189, Revision 4, Regulatory Position 5, "Safe-Shutdown Capability," and Regulatory Position 5.3.1, "Identification and Evaluation of Post-Fire Safe-Shutdown Circuits."

Based on the above information, the staff finds that the proposed design conforms to GDC 3, in that SSCs important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.

The applicant stated that compressed or cryogenic gas should not be stored inside or near safety-critical buildings or systems during any mode of plant operation. Flammable gas is stored outdoors or in detached buildings to prevent fire or explosion from affecting safety-important equipment.

NuScale FSAR states that the protection of the RXB, CRB, and RWB from the effects of external fires from adjacent buildings, is in accordance with NFPA 80A 2019 Edition. Exposure hazards are plant-specific and vary depending on the final location of the plant and arrangement of the nearby structures and support buildings. Intervening combustibles have a 50-foot spatial separation in accordance with NFPA 804,2020 Edition, Section 8.1 and Section 8.9, and RG 1.189, Revision 4, Regulatory Positions C.7.3 and C.7.4. Based on the above information, the staff concludes that the NuScale US460 standard design conforms to NFPA 804 and RG 1.189, Revision 4, Regulatory Position 8.2 as listed in Table 9.5.1-2 of FSAR, Chapter 9.

9.5.1.4.2.2 Use of Noncombustible and Heat-Resistant Materials

The staff reviewed Section 9.5.1.1, "Design Bases", and Appendix 9A, to ensure that it conforms to GDC 3 as it relates to the use of noncombustible and heat resistant materials wherever practical throughout the unit. The applicant stated that the RXB, CRB, and RWB floors, walls, and ceilings are constructed almost entirely of reinforced concrete.

According to the Regulatory Guide 1.189, risk-significant or safety-related SSCs are designed and located to minimize the probability and effect of fires and explosions. Noncombustible and fire resistance materials are used throughout the plant where fire is a potential risk to safety-related systems. Passive fire barriers compartmentalize the plant into separate areas or zones. Compartmentalization separates redundant, safety-related systems and components to ensure that a fire in one area does not prevent the redundant systems and components in an adjacent area from performing their safety functions. The primary purpose of these fire areas or zones is to confine the effects of fires to a single compartment or area, thereby minimizing the potential for adverse effects from fires on redundant risk-significant or safety-related SSCs. Compartmentalization is achieved by using properly rated fire barriers, fire doors, fire dampers, and penetration seals to prevent the spread of fire between areas. Based on the staff review, adequate equipment and cable separation meet the enhanced fire protection criteria as described in RG 1.189, Revision 4, Regulatory Position 8.2 by the design of these divisions and subdivisions.

Transformers installed inside buildings containing SSCs that are safety-related or have risk-significant functions are dry type or insulated and cooled with noncombustible liquids to prevent fires from adversely impacting the ability to safely shut down the plant.

Outside transformers are either 50 ft from plant buildings, or a three-hour fire barrier with no openings that separate outside transformers from the plant buildings. NuScale FSAR Appendix 9A state that the fire barriers used for the outside transformers conform with NFPA 804, 2020 Edition. The transformer area provides oil spill confinement and confines used fire water suppression.

Electrical conductors within the CNV are noncombustible or routed in conduit, which results in no intervening combustible loading. Cable not in conduit is noncombustible.

Only metal is used for cable trays and metallic tubing is used for conduit. Thin wall metallic tubing is not used and flexible metallic tubing is used only in short lengths to connect components to equipment.

The liquid, gaseous, and solid radioactive waste processing and storage systems described in FSAR Chapter 11, Sections 11.2, 11.3, and 11.4 rely almost exclusively on metal tanks or containers. Exceptions may include storage of radioactive wastes that are packaged for shipping in approved (nonmetal) high integrity containers.

The staff finds that the proposed design conforms to GDC 3, in that noncombustible and heat -resistant materials are used wherever practical.

9.5.1.4.2.3 Fire Detection and Fighting Systems

The staff reviewed Section 9.5.1.1, "Design Bases", and Appendix A, to ensure that it conforms to GDC 3 as it relates to fire detection and firefighting systems that are of the appropriate capacity and capability and designed to minimize the adverse effects of fires on SSCs.

The FHA Appendix 9A of NuScale FSAR identifies the extent to which fire detection and automatic or manual fire suppression systems are required. Fire detection and fire suppression systems are installed in accordance with applicable industry codes and standards.

Areas that contain or present a fire exposure to equipment with safety-related or risk-significant functions have fire detection alarms that sound in the MCR. Fire detection and alarm systems comply with the requirements of Class A systems, as defined in NFPA 72, "National Fire Alarm Code," 2019, Edition and Class I circuits, as defined in NFPA 70, "National Electrical Code," According to the FHA Appendix 9A of NuScale FSAR.

The fire detector's location and installation are in accordance with NFPA 72, 2019 Edition, NFPA 804, 2020 Edition, and RG 1.189, Revision 4 and the FHA Appendix 9A of NuScale FSAR. The type of detection used, and the location of the detectors are the most suitable for the particular type of fire hazard identified by the FHA.

Primary and secondary power supplies exist for the fire detection and alarm system as well as electrically operated valves in the fire suppression system. According to NuScale the primary and secondary power supplies meet the requirements of with NFPA 72, 2019 Edition. Control room fire detection and alarms are in accordance with the guidance in Regulatory Position 6.1.2 of RG 1.189, Revision 4.

Areas that contain or present a fire exposure to equipment with safety-related or risk-significant functions have fire detection that alarms in the MCR. The following areas are provided with automatic detection.

- Plant computer rooms
- Switchgear rooms
- battery rooms
- diesel generator areas
- pump rooms
- new and spent fuel areas

- radioactive waste and decontamination areas

Air-sampling systems (i.e., Very Early Warning Fire Detection Systems) or individually addressable intelligent smoke detectors are used in areas to eliminate the need for continuous line-type heat detectors. Complete design to be addressed by the COL applicant as part of site-specific design (Item 4.1.3.3, Table 9.5.1-2 of FSAR, Chapter 9. This item will be addressed by an applicant who references the NuScale US460 SDAA standard design in a COL application).

The FPS water supply system is designed in accordance with NFPA 22, "Standard for Water Tanks for Private Fire Protection," and NFPA 24, "Standard for the Installation of Private Fire Service Mains and Their Appurtenances." The water supply meets the following criteria:

- Two 100-percent system capacity freshwater tanks are available independent of other water systems. Tanks are installed and interconnected so that the fire pumps can take suction from either or both tanks. A failure in one tank does not cause both tanks to drain. The tanks connect to a water supply capable of refilling the tank in eight-hours or less.
- The water supplies are sized to provide the largest expected flow rate for a minimum of 2 hours, but the size of the supplies is not less than 300,000 gallons.
- Two 100percentcapacity tanks are installed and interconnected in accordance with NFPA 22, "Standard for Water Tanks for Private Fire Protection," 2018 Edition and NFPA 24, "Standard for the Installation of Private Fire Service Mains and Their Appurtenances," 2019 Edition. Fire pumps can take suction from either or both tanks. A failure in one tank will not cause both tanks to drain. The tanks are connected to a water supply capable of refilling the tank in 8 hours or less. Based on the above information, the staff concludes the NuScale US460 standard design conforms to RG 1.189, Revision 4, Regulatory Position 3.2.1 as listed in Table 9.5.1-2 of FSAR, Chapter 9. Based on the above information NuScale US460 standard design conforms to NFPA 804 and RG 1.189, Revision 4, Regulatory Position 3.2.1 as listed in Table 9.5.1-2 of FSAR, Chapter 9.
- Fire water supplies are filtered and treated as necessary to prevent and control bio-fouling or microbiologically induced corrosion of the fire water systems.
- Two Fire pump (electric motor-driven and diesel engine-driven) installations conform to NFPA 20, "Standard for the Installation of Stationary Pumps for Fire Protection," 2019 Edition, consistent with NFPA 13, "Standard for the Installation of Sprinkler Systems," 2019 Edition each pump is capable of delivering the demand from the largest sprinkler or deluge system plus an additional 1900 L/min (500 F) for fire hoses. Fire pump status alarms are in the MCR. The motor-driven fire pump has 480 VAC power from the low-voltage alternating current electrical distribution system. The fuel tank for the diesel engine-driven pump holds enough fuel to operate the pump for at least eight hours. A motor-driven jockey pump keeps the firewater system full of water and pressurized when the main pumps are not operating. The jockey pump design and operation are in accordance with the NFPA 20 guidance for pressure maintenance pumps. Based on the information above, NuScale US460 standard design conforms to RG 1.189, Revision 4, Regulatory Position 3.2.2 as listed in Table 9.5.1-2 of FSAR, Chapter 9.
- Fire hydrants in the FPS are of an approved type for FPS. Hydrants allow for

pressurization of the fire main from an external source. There are hydrants every 250 feet along the yard main. One hydrant on each of the four RXB sides is at least 300 feet from the RXB to satisfy the loss of large area requirements. Based on the above information, the staff conclude the NuScale US460 standard design conforms to RG 1.189, Revision 4, Regulatory Position 3.4.2 as listed in Table 9.5.1-2 of FSAR, Chapter 9.

Automatic fire suppression systems are designed to detect fires and provide the capability to extinguish them. Automatic fire suppression systems are used where necessary to protect redundant systems or components required for safe shutdown and SSCs with safety-related or risk-significant functions.

Automatic sprinkler and water spray systems are used to protect against a variety of hazards, such as those related to cable areas, lubrication oil hazards, computer rooms, and transformers. Automatic sprinkler systems are installed in accordance with NFPA 13, 2019 Edition. Automatic water spray systems are installed in accordance with NFPA 15, “Standard for Water Spray Fixed Systems for Fire Protection,” 2017 Edition. Based on the above information, the staff concludes the NuScale US460 standard design conforms to RG 1.189, Revision 4, Regulatory Position 3.2.1,b as listed in Table 9.5.1-2 of FSAR, Chapter 9. Table 9A-6, “Hazard Classifications,” lists hazard classifications designated in Section 9A.5; these classifications are in alignment with Chapter 5 of NFPA 13, 2019 Edition, and Chapter 6 of NFPA 101, “Life Safety Code,” 2021 Edition.

Manual firefighting capability is provided throughout the plant to give the fire brigade the ability to limit fire damage to SSCs with safety-related or risk-significant functions. Outside fire hydrants and hose installations allow manual firefighting for outside hazards that could impact equipment with safety-related or risk-significant functions. Fire hydrants are provided every 76 m (250 ft) along the yard main system. Floor drains design are in accordance with RG 1.189 Regulatory Position 4.1.5 as described in Table 9.5.1-2. FSAR Chapter 3, Section 3.4.1 discusses flood protection for equipment required to perform a safety function.

The FPP addresses implementation plans to establish an organizational structure, train, and equip the site fire brigade to ensure adequate manual firefighting capability for areas for SSCs with safety-related or risk-significant functions in accordance with RG 1.189, Revision 4, Regulatory Positions 1.6.4, 1.6.4.1, 1.6.4.2, 1.6.4.3, 1.6.4.4, 3.5.1, 3.5.1.1, 3.5.1.2, 3.5.1.3, 3.5.1.4, 3.5.2, 3.5.2.1, 3.5.2.2, and 3.5.2.3. The organizational structure includes training, qualification, and documentation and maintenance of training and qualification records.

Where provided, interior hose installations can reach areas with 30 m (100 ft) of hose and an effective hose stream. Standpipe and hose systems are designed and installed in accordance with NFPA 14, “Standard for the Installation of Standpipe and Hose Systems,” for sizing, spacing, and pipe support requirements for Class III standpipes. Based on the above information, the staff concludes the NuScale US460 standard design conforms to RG 1.189, Revision 4, Regulatory Position 3.4.1 as listed in Table 9.5.1-2 of FSAR, Chapter 9.

At least two standpipes and hose connections are provided for manual firefighting in areas containing equipment required for safe plant shutdown in the event of an SSE. The piping is analyzed for SSE loading and provided with supports to ensure system pressure integrity. The piping and valves for these seismically analyzed standpipes satisfy ASME B31.1. For the purpose of supplying fire water to the seismically analyzed standpipes, the piping system serving the RXB from the fire water storage tanks to the diesel fire pump, then from the diesel fire pump to the RXB’s seismically analyzed piping up to and including sectional isolation valves supplying buildings and systems other than the RXB, are designed to the requirements of ASME B31.1. Based on the above information, the staff concludes the NuScale US460 standard design conforms to RG 1.189, Revision 4, Regulatory Position 3.2.1,j as listed in Table 9.5.1-2 of FSAR, Chapter 9.

Fire extinguishers are provided in areas that could present a fire exposure hazard to equipment with safety-related or risk-significant functions. Fire extinguishers are the appropriate size and type for the fire hazards in the area. NFPA 10, "Standard for Portable Fire Extinguishers," provides guidance on the installation of portable fire extinguishers. Based on the above information, the staff concludes the NuScale US460 standard design conforms to RG 1.189, Revision 4, Regulatory Position 3.4 as listed in Table 9.5.1-2 of FSAR, Chapter 9.

The staff finds that the proposed design conforms to GDC 3 and is therefore acceptable, because the design provides fire detection and firefighting systems of appropriate capacity and capability to minimize the adverse effects of fires on SSCs. Further, firefighting systems are designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of SSCs.

9.5.1.4.2.4 Rupture and Inadvertent Operation of Firefighting Systems

The staff reviewed Section 9.5.1.1, "Design Bases", and FHA Appendix 9A, to ensure that it conforms with GDC 3 as it relates to confirming that firefighting systems are designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of SSCs. Additionally, Section 3.4, "Water Level (Flood) Design," of NuScale US460 standard design, FSAR, Chapter 3, Design of Structures, Systems, and Components and Equipment (ML25099A247), evaluates the impact of inadvertent actuation or breaks in the FPS water supply piping. No credit was taken for the floor drains of the radioactive waste drain system (RWDS) or the balance of plant drain system (BPDS) in removing fire water. The effect of fire suppression system operation, either in response to a fire or a spurious discharge, is minimized by providing suitable protection for equipment that may be compromised by the operation of the fire suppression system.

Redundant divisions of safe-shutdown equipment for the NPMs are located in separate fire areas where practicable so that fires, a spurious discharge, or a failure of the FPS can affect only one division of safe-shutdown equipment per NPM. Facility design ensures that fire water discharge in one area does not impact safety-related equipment in adjacent areas.

Internal flooding analyses were performed by NuScale for the RXB and CRB to confirm that flooding from postulated failures of tanks and piping or actuation of fire suppression systems does not cause loss of equipment required to perform safety functions. These SSCs are equipment subject to flood protection.

The staff finds that the proposed design conforms to GDC 3, because the firefighting systems are designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of SSCs and is therefore acceptable.

*9.5.1.4.3 GDC 5, "*Sharing of Structures, Systems, and Components"

The staff reviewed Section 9.5.1.1, "Design Bases", and Appendix 9A, to ensure that it conforms to GDC 5, as it applies to shared FPSs and potential fire impacts on shared SSCs important to safety. The applicant stated that the NPMs are all located in the RXB, which is serviced by a common, shared FPS. Automatic fire detection and suppression systems are provided to protect redundant systems or components required for safe shutdown and SSCs with safety-related or risk-significant functions.

The independence of redundant safe-shutdown circuits is such that a fire in a fire area will not prevent the redundant circuits in a separate fire area from performing their safe-shutdown functions. Redundant divisions of safe-shutdown equipment for the NPMs are located in separate fire areas where practicable so that fires, a spurious discharge, or a failure of the FPS can affect only one division of safe-shutdown equipment per module. There are fire areas in the RXB where one fire could affect multiple modules, although only one division per module would be affected, leaving an alternative division intact. With one success path of safe-shutdown equipment available for each module, safe-shutdown functions can still be performed for all modules. Because the FPS provides protection to all modules and redundant safe-shutdown circuits and equipment are located in separate fire areas, the staff finds that the design conforms to GDC 5.

9.5.1.4.4 PDC 19, "Control Room"

The staff reviewed Section 9.5.1.1, "Design Bases", and Appendix 9A, to ensure that it conforms to PDC 19, as it relates to the design providing the capability both inside and outside the control room to operate plant systems necessary to achieve and maintain safe-shutdown conditions. The applicant stated that the FPS protects the CRB, which houses the MCR. By protecting the CRB, the FPS protects the cables, switching and transmitting equipment, and display components from fire damage, allowing the MCR to function. In the RXB, the FPS protects sensing, switching and transmitting equipment, as well as cabling, which contributes to the functionality of the control room in case of fire in the RXB. Within the MCR, the FPS provides automatic fire detection in the cabinets and consoles. The FPS also provides manual fire suppression capability within the control room by providing portable fire extinguishers and hose stations.

NuScale US460 standard design MCR is designed with the ability to place the reactors in safe shutdown in case of a fire requiring an MCR evacuation and for safe shutdown to be maintained without operator action thereafter. Before evacuating the MCR, operators trip the reactors, initiate decay heat removal, and initiate containment isolation. These actions result in passive cooling that achieves safe shutdown of the reactors. Operators can also achieve safe shutdown of the reactors from outside the MCR in the I&C equipment rooms within the RXB. Following shutdown and initiation of passive cooling from either the MCR or the I&C equipment rooms, the design does not rely on operator action, instrumentation, or controls outside the MCR to maintain the safe shutdown condition. There are no remote displays, alarms, or controls necessary to monitor or maintain the modules in a safe shutdown condition. Emergency lighting is provided for access to and illumination of equipment necessary to implement the shutdown from the remote shutdown room. This emergency lighting has at least an 8-hour battery backup power. Because the FPS protects the building and associated SSCs that house the MCR and itself and provides for a remote shutdown station, the staff finds that the design conforms to PDC 19.

9.5.1.4.5 GDC 23, "Protection System Failure Modes"

The staff reviewed NuScale US460 standard design FSAR Chapter 9, Section 9.5.1.1, "Design Bases", and Appendix 9A, to ensure that it conforms to GDC 23, as applied to designing the protection system to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if the plant experiences adverse environments such as from a fire. The applicant stated that, consistent with GDC 23, functional requirements have been imposed on the design of the MPS that addresses safe failure states when exposed to the effects of fire and water. The MPS is designed, with sufficient functional diversity to prevent loss of a protection function, to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if the plant experiences adverse environments such as those from a fire.

NuScale US460 standard design FSAR Chapter 7, “Instrumentation and Control,” (ML25099A257), Section 7.1.1.1, “Design Bases,” consistent with GDC 23 for the MPS, has sufficient functional diversity to prevent the loss of a protection function, to fail into a safe state or into a state demonstrated to be acceptable if conditions such as disconnection of the system, loss of power, or postulated adverse environments are experienced.

NuScale demonstrated that its protection systems are designed to fail safely or to an acceptable state, even under adverse conditions such as fire. This approach aligns with Regulatory Guide 1.189, which provides acceptable fire protection methods for nuclear plants. The staff concludes that NuScale’s design meets the intent of GDC 23 by ensuring protection systems maintain their safety functions during fire-induced conditions.

9.5.1.5 Initial Test Program

The preoperational test related to the FPS for design certification is Fire Protection Systems Test #22. This test is performed in accordance with FSAR Table 14.2-22 and fire codes and standards listed in FSAR Table 9.5.1-1. Section 14.2 of this SER provides the staff’s evaluation of the plant’s ITP.

9.5.1.6 Combined License Information Items

Table 9.5.1-1 lists COL information item numbers and descriptions related to fire protection from FSAR, Table 1.8-1.

Table 9.5.1-1 NuScale COL Information Items for Section 9.5.1

COL Item No.	Description	FSAR Section
13.4-1	An applicant that references the NuScale Power Plant US460 standard design will provide site-specific information, including implementation milestones, for Operational Programs […Fire Protection Program (Section 9.5.1) …]	9.5.1

FSAR Table 9.5.1-2, “NuScale Fire Protection Design Compliance with RG 1.189,” is a point-by-point comparison of the conformance of the NuScale US460 SDAA with the guidelines of RG 1.189, Revision 4. The table lists items and descriptions related to FPS design compliances with Regulatory Positions in RG 1.189, Revision 4. These regulatory position items conform with RG 1.189, Revision 4, and will be addressed by an applicant who references the NuScale US460 SDAA standard design in a COL or an operating license application. FSAR Table 9.5.1-2, Note 2 states, "Applicant - The Applicant/Licensee will (also) address the subject Regulatory Position.” The staff finds this acceptable, because it is consistent with RG 1.189.

In addition, FSAR, Section 9.5.1.4, “Inspection and Testing Requirements,” states that the periodic inspection and testing to ensure system functionality is in accordance with applicable codes and standards. The staff finds this acceptable, because the NuScale US460 standard design follows the fire protection industry codes and standards included in FSAR Table 9.5.1-1.

9.5.1.7 Conclusion

Based on the review above, the staff concludes that the FPP for the NuScale US460 design is in accordance with the guidance provided in RG 1.189, Revision 4 and applicable industry codes and standards. Consistency with RG 1.189, Revision 4, ensures that the fire detection and fighting systems provided have the capacity and capability to minimize the adverse effects of fires and that their rupture or inadvertent operation does not impair the safety capability of other SSCs. In addition, the staff concludes that the applicant has adequately addressed the need for COL information items for this review section.

9.5.2 Communication Systems

9.5.2.1 Introduction

The communications systems (COMS) discussed in this SER primarily involve verbal communication functions between personnel and organizations, although there may also be physical communication links in some cases to transmit limited data communications (e.g., Web page or facsimile transmission over the telephone lines). DSRS Chapter 9 and SRP Section 13.3, Revision 3, "Emergency Planning," issued March 2007, address the review of systems for communicating data among portions of the instrumentation systems and among site-related facilities such as the MCR, TSC, operations support center (OSC), emergency operations facility (EOF), meteorological stations, and security stations.

This review of NuScale's COMS is limited to that portion of the system used in intraplant (including among multiple modules, units, and control rooms at a single plant site) and plant-to-offsite communications during normal operation; transients; fire; accidents; off-normal phenomena including tornado, hurricane, flood, tsunami, lightning strike, and earthquake and declared emergencies; and security-related events.

As stated in FSAR Section 9.5.2, NuScale's COMS comprise the following systems:

- telephony
- wide area mass notification system (WAMNS)
- distributed antenna
- satellite telephony
- health physics network

9.5.2.2 Summary of Application

FSAR Section 9.5.2, "Communication System," provides information associated with this section. The COMS serves no safety-related or risk-significant functions, is not credited for mitigation of DBAs, and does not interfere with safety-related or risk-significant structures, systems, or components.

ITAAC: No ITAAC are associated with FSAR Section 9.5.2.

9.5.2.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," particularly Part IV.E(9), as it relates to the provision of at least one onsite and one offsite COMS, each with a backup power source

- 10 CFR 50.34(f)(2)(xxv), for TMI Action Plan Item III.A.1.2, as it relates to the provision for communications made to support an onsite TSC, an onsite OSC, and a near-site EOF

- 10 CFR 50.47(b)(6) and 10 CFR 50.47(b)(8), as they relate to the provision for communications provided and maintained in the emergency facilities and control room to support emergency response

- GDC 1, as it relates to the provision that communication equipment and related support equipment important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed

- GDC 2, as it relates to the provision that communication equipment and related support equipment important to safety are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches

- GDC 3, as it relates to the provision that communication equipment and related support equipment important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires, smoke effects from fires, and explosions

- GDC 4, as it relates to the provision that communication equipment and related support equipment important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs

- GDC 19, as it relates to the provision that communication equipment is provided at appropriate locations inside the control room and designed with the capability to support all normal and emergency operations

  NuScale has provided rationale to support that an exemption request from GDC 19 to implement a design-specific Principal Design Criterion (PDC) 19 that maintains the reactor in a safe condition would be justified in lieu of the requirements for "design capability for prompt hot shutdown" and "potential capability for subsequent cold shutdown" as specified in GDC 19. The staff's evaluation of the rationale that supports PDC 19 is provided in Section 6.4 of this report.

- 10 CFR 73.55(e)(9)(vi)(B), and 10 CFR 73.55(j), as they relate to physical protection communication requirements.

The guidance in DSRS Section 9.5.2 lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

In addition, DSRS Section 9.5.2 notes that the applicant should ensure that communications equipment will be compatible with the electromagnetic interference (EMI) and radiofrequency interference (RFI) environments of the plant and that design measures have been taken such that there will be no interference between wireless communications systems and other plant equipment, including application of the appropriate guidance from RG 1.180, "Guidelines for

Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems."

9.5.2.4 Technical Evaluation

Sections 3.2.2, 17.4, and 19.1 of this report describe the basis for the acceptability of the safety-significance categorization for the COMS function. The COMS serve no safety-related or risk-significant functions. The COMS are not credited for the mitigation of DBAs nor do they have any safe-shutdown functions. The failure of any COMS does not adversely affect safe-shutdown capability. The B2 (not safety-related and not risk significant) functions of the COMS are reflected in FSAR Table 9.5.2-1. The NRC staff also performed its review to ensure that the COMS will not adversely impact any safety-related functions. Section 7.0.4.2 of this report provides the NRC staff's evaluation of NuScale's I&C system architecture.

9.5.2.4.1 Compliance with 10 CFR Part 50, Appendix E, Part IV.E(9)

In 10 CFR Part 50, Appendix E, Part IV.E(9), the NRC requires that adequate provisions shall be made and described for emergency facilities and equipment including at least one onsite and one offsite communication system; each system shall have a backup power source. NuScale's WAMNS, telephony system, and the plant radio system provide onsite communications. NuScale's telephony and plant radio system provide offsite communications. The low voltage ac electrical distribution system that is not safety-related supplies power to these systems. Three independent voice communications systems provide onsite communications. The failure of any or all of them does not affect safety-related equipment. FSAR Section 9.5.2, contains COL Information Item 9.5-1 and is reflected in FSAR Table 1.8-1, which states the following:

> An applicant that references the NuScale Power Plant US460 standard design will provide a description of the offsite communication system, how that system interfaces with the onsite communications system, as well as how continuous communications capability is maintained to ensure effective command and control with onsite and offsite resources during both normal and emergency situations.

The NRC staff considers this COL information item acceptable because offsite COMS are unique to the COL applicant and can be addressed at the time of COL application. Because COL Information Item 9.5-1 is acceptable and there is at least one onsite and offsite COMS with backup power sources, the NRC staff finds that the requirements of 10 CFR Part 50, Appendix E, Part IV.E(9), have been met.

9.5.2.4.2 Compliance with 10 CFR 50.34(f)(2)(xxv), 10 CFR 50.47(b)(6), and 10 CFR 50.47(b)(8)

In 10 CFR 50.34 (f)(2)(xxv) and TMI Action Plan Item III.A.1.2, the NRC requires that applicants provide for an onsite TSC, an onsite OSC, and for construction permit applications only, a near-site EOF.

Section 13.1 of this report provides the NRC staff's evaluation of the design details for the technical support center (TSC) and plant, local, and offsite emergency response facilities. Because the design includes an onsite TSC and plant, local, and offsite emergency response facilities, the NRC staff finds that the applicant has met the requirements of 10 CFR 50.34(f)(2)(xxv) with respect to COMS.

In 10 CFR 50.47(b)(6), the NRC requires that provisions exist for prompt communications among principal response organizations to emergency personnel and to the public. In 10 CFR 50.47(b)(8), the NRC requires that adequate emergency facilities and equipment to support emergency response are provided and maintained.

FSAR Section 9.5.2.3, "Safety Evaluation," states, in part, the following:

> [A]dequate provisions for communications are provided and maintained in the emergency facilities and MCR to support the emergency response, including prompt communication among principal response organizations to emergency personnel and to the public.

The TSC and plant, local, and offsite emergency response facilities provide prompt communications among principal response organizations. Section 13.1 of this report provides the NRC staff's evaluation of the design details for the TSC and plant, local, and offsite emergency response facilities. Section 13.3 of this report also states, in part, that the design of the TSC complies with NUREG-0696. FSAR Section 9.5.2.3, also states, in part, the following:

> The TSC has voice communications such as the telephony system, WAMNS, and the plant radio system, which provide communications between the TSC and plant, local, and offsite emergency response facilities, the Nuclear Regulatory Commission, and local and state operations centers.

Because the design provides for a TSC that is equipped with voice COMS capable of providing both onsite and onsite-to-offsite communications during normal operating conditions as well as for emergency response, the NRC staff finds that the requirements of 10 CFR 50.47(b)(6) and 10 CFR 50.47(b)(8) have been met.

9.5.2.4.3 Compliance with General Design Criteria

GDC 1 requires SSCs important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. The COMS is not an important-to-safety or risk-significant SSC. FSAR Section 9.5.2.3, states, in part, the following:

> Consistent with GDC 1, COMS structures, systems, and components are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. The COMS is classified as a non-Class 1E system and serves no safety-related function.

The COMS is a non-Class 1E system that does not serve any safety-related function. Because the COMS is designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed, the NRC staff finds that the requirements of GDC 1 have been met.

GDC 2 requires that SSCs important to safety withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without the loss of capability to perform their safety functions. FSAR Section 9.5.2.3, states that consistent with GDC 2, the COMS is not required to function during or after natural phenomena. Therefore, the NRC staff finds that the COMS does not have to meet GDC 2.

GDC 3 requires that SSCs important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. FSAR Section 9.5.2.3, states, in part, the following:

> Consistent with GDC 3, the COMS design minimizes the probability and effect of fires and explosions. The COMS provides two-way voice communications to support safe shutdown and emergency response in the event of fire. The plant radio system complies with Regulatory Guide 1.189, Regulatory Position 4.1.7, in that the COMS design provides effective communications between plant personnel in vital areas during fire conditions under maximum potential noise levels.

As the COMS is designed in accordance with RG 1.189, Regulatory Position 4.1.7, the NRC staff finds that the requirements of GDC 3 have been met.

GDC 4 requires that SSCs important to safety accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. These SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. The COMS is not an important-to-safety or risk-significant SSC. Hence, the COMS does not have to meet GDC 4. FSAR Section 9.5.2.3, states, in part, “the COMS is not required to function during or after events that result in the generation of missiles, pipe whipping, or discharging fluids.”

> FSAR Section 9.5.2.3 states that the WAMNS, telephony system, and plant radio system are physically independent. These systems serve as a backup to one another in the event of
>
> system failure as a result of natural phenomena, environmental or dynamic effects, and fires. The independence of the voice communications systems ensures any single event does not cause a complete loss of intra-plant communication.

FSAR Section 9.5.2.2, also states, in part, that the COMS meets the practices for limiting EMI and RFI provided by RG 1.180, which identifies electromagnetic environment operating envelopes, design, installation, RFI, and power surges on I&C systems and components. However, because the COMS is not an important-to-safety or risk-significant SSC, the NRC staff finds that the COMS does not need to be credited for evaluating compliance with GDC 4.

PDC 19 requires that an MCR be provided from which actions can be taken to operate the plant safely under normal conditions and to maintain it in a safe condition under accident conditions. 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 I&Cs 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.

The NuScale US460 design provides a control room to operate the plant safely under normal and accident conditions. However, FSAR Section 9.5.2.3, states, in part, “The design allows for safe shutdown without operator action.” Therefore, the operators do not need the COMS to take actions to shut down the plant safely. FSAR Section 9.5.2.3, also states the following:

> However, the various independent and diverse communications systems located in the MCR increase the overall command and control the reactor operators have over the plant by providing the ability to communicate and direct activities with operating, maintenance, health physics, firefighting, security, and rescue personnel.

As the control room operators do not need the COMS to take actions for safe shutdown during normal and accident conditions, the NRC staff finds that the COMS does not need to be credited for evaluating compliance with PDC 19.

9.5.2.4.4 Compliance with 10 CFR 73.55(e)(9)(vi)(B), and 10 CFR 73.55(j)

The staff evaluated compliance with the above regulations from 10 CFR Part 73, "Physical Protection of Plants and Materials," in Section 13.6 of this SER.

9.5.2.4.5 Electromagnetic Interference and Radiofrequency Interference Compatibility

DSRS Section 9.5.2 calls, in part, for verification that communications equipment will be compatible with the EMI and RFI environment of the plant and that design measures have been taken such that there will be no interference between wireless communications systems and other plant equipment. Control of EMI and RFI from these systems that are not safety-related is necessary to ensure that safety-related I&C systems can continue to perform properly in the nuclear power plant environment.

FSAR Section 9.5.2.2, states, in part, the following:

> The COMS meets the practices for limiting electromagnetic interference and radio frequency interference provided by Regulatory Guide 1.180, which identifies electromagnetic environment operating envelopes, design, installation, and test practices for addressing the effects of electromagnetic interference, radio frequency interference, and power surges on instrumentation and controls systems and components.

Because NuScale committed to conform to RG 1.180, the NRC staff finds that the COMS adequately addresses EMI and RFI testing to ensure that EMI and RFI effects from the COMS do not adversely impact safety systems.

9.5.2.5 Initial Test Program

FSAR Table 14.2-61, Test #61 provides the ITP test for communication. The staff evaluates the ITP in Section 14.2 of this report.

9.5.2.6 Combined License Information Items

Table 9.5.2-1 lists the COL information item number and description related to COMS from FSAR, Table 1.8-1.

Table 9.5.2-1 NuScale COL Information Item for Section 9.5.2

COL Item No.	Description	FSAR Section
9.5-1	An applicant that references the NuScale Power Plant US460 standard design will provide a description of the offsite communication system, how that system interfaces with the onsite communications system, as well as how continuous communications capability is maintained to ensure effective command and control with onsite and offsite resources during both normal and emergency situations	9.5

9.5.2.7 Conclusion

Based on the review above, and to the extent the application addressed the use of the COMS in intra-plant and plant-to-offsite communications in support of the plant in normal, emergency, and security functions, the staff finds that the COMS designs, with the exception of requirements relating to 10 CFR Part 73, are acceptable and meet the applicable requirements described in the Regulatory Basis for this section. Note that the staff's evaluation of COMS requirements associated with 10 CFR Part 73 is provided in Section 13.6 of this report.

9.5.3 Lighting Systems

9.5.3.1 Introduction

FSAR Section 9.5.3, "Lighting Systems," states that the plant lighting system (PLS) provides artificial illumination for buildings, rooms, spaces, and outdoor areas of the plant and under plant operating conditions including normal, transient, fire, accident, and station blackout. The PLS includes the normal plant lighting, the main control room (MCR) normal lighting, the emergency plant lighting, and the emergency MCR lighting.

Furthermore, the PLS includes security lighting, which includes the exterior plant lighting within the protected area that receives power from the security power system. Chapter 13 of this report provides the staff's evaluation of the lighting for physical security.

9.5.3.2 Summary of Application

FSAR, Section 9.5.3.1, "Design Bases," states

> Normal and emergency plant lighting are not required to function in response to a design-basis accident. The PLS is not essential for reactor shutdown, containment isolation, or containment and reactor heat removal. The PLS is not essential in preventing release of radioactive material to the environment. Failure of normal and emergency lighting does not compromise automatic actuation of nuclear safety-related systems, nor does it prevent safe shutdown of the reactor. Therefore, normal and emergency plant lighting are non-safety-related, not risk-significant, and non-Class 1E. …
>
> The PLS includes lighting transformers which receive power from the 480 [volt alternating current] VAC Electrical Low Voltage Systems (ELVS). The secondary side of the lighting transformer is the 120-VAC that provides 120 VAC to the lighting fixtures in the plant.

> The plant illumination levels provided by the PLS are in accordance with the applicable lighting levels specified in NUREG-0700, Revision 3, section 12.1.2.4-4 and Table 12.8 in section 12.2.2.3-1. The emergency lighting system conforms with applicable guidance of Regulatory Guide 1.189.
>
> Lighting fixtures in the MCR and areas containing safety-related structures, systems, and components are mounted to meet Seismic Category II requirements.

FSAR, Section 9.5.3.2, "System Description," describes the normal, emergency, and normal and emergency MCR plant lightings, and states, in pertinent part:

> Normal Plant Lighting
>
> Normal plant lighting provides artificial illumination for outdoor areas outside the protected area and within the owner-controlled area, and for plant buildings.
>
> The low voltage alternating current (AC) electrical distribution system, described in Section 8.3.1, Alternating Current Power Systems, provides power to the lighting panel boards that feed the plant's light fixtures with the exception of [the] security lighting.
>
> Normal MCR Lighting
>
> The PLS cabinets provide 120 V AC power to the normal lighting fixtures during normal, operating, maintenance, and testing conditions. The 120 VAC power provides the normal lighting fixtures to the required illumination levels as stated in Revision 3 of NUREG-0700 Sections 12.1.2.3-1 through 12.1.2.3-4. In the event that AC power is not available, the normal MCR lighting fixtures will no longer provide illumination.
>
> Emergency Plant Lighting
>
> Emergency lighting fixtures, outside of the MCR, have self-contained batteries that are powered from the low voltage AC electrical distribution system. Upon a loss of AC power to the plant, the batteries provide power to their associated fixtures. The PLS provides emergency lighting outside the MCR.
>
> Emergency MCR Lighting
>
> Two divisions of the common augmented direct current power system [EDAS-C] , described in Section 8.3.2, Direct Current Power Systems, provide power to the emergency lighting fixtures in the MCR. Emergency lighting fixtures in the MCR are continuously on during normal, operation, maintenance, testing, transient, and emergency conditions. Upon loss of AC power, the MCR emergency lighting fixtures will continue to be supplied power (125 VDC) via EDAS-C batteries.

FSAR Table 9.5.3-1, "Classification of Structures, Systems, and Components," identifies SSC classification and seismic classification for the PLS. The PLS is assigned SSC classification B2. The SSC classification is described in FSAR, section 3.2, "Classification of Structures, Systems, and Components." The PLS is classified as seismic Category III in Table 9.5.3-1, but portions of the PLS can be designed as seismic Category II in the as-built plant. Note 4 in Table 9.5.3-1 states: "Where SSC (or portions thereof) as determined in the as-built plant that are identified as

Seismic Category III in this table could, as the result of a seismic event, adversely affect Seismic Category I SSC or result in incapacitating injury to occupants of the control room, they are categorized as Seismic Category II consistent with Section 3.2.1.2 and analyzed as described in Section 3.7.3.8." Seismic Category II SSC is defined in FSAR, Section 3.2.1.2, "Seismic Category II."

FSAR Table 1.9-3, "Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard," indicates that NuScale US460 design conforms to SRP section 9.5.3 for normal and emergency lighting and partially conforms to NUREG-0700, Revision 3.

ITAAC: NuScale gave the ITAAC associated with FSAR Section 9.5.3, in SDAA Part 8, Section 3.8, "Plant Lighting System." These ITAAC are evaluated in Section 14.3 of this SER.

9.5.3.3 Regulatory Basis

There are no specific GDCs or other requirements that directly apply to the performance of the lighting systems. However, 10 CFR 50.34(f)(2)(iii) states, in part, that an application shall provide "a control room design that reflects state-of-the-art, human-factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts." A control room design includes lighting for operators to perform actions, and NUREG-0700 provides detailed acceptance criteria for human factors engineering design attributes, including lighting.

The following NRC guidance applies to the review of lighting:

- SRP Section 9.5.3, Revision 3, "Lighting Systems," issued March 2007, provides acceptance criteria for the lighting systems. The lighting systems must (1) provide adequate lighting in all areas of the plant during normal plant operations, (2) provide adequate emergency lighting during all plant operating conditions, including fire, transient, and accident conditions, (3) address the effect of the loss of all AC power (i.e., during an SBO) on the emergency lighting system, and (4) have adequate illumination levels that conform to the illumination levels recommended in NUREG-0700.

- NUREG-0700, Revision 3, "Human-system Interface Design Review Guidelines," as it relates to acceptable lighting levels.

- RG 1.75, Revision 3, "Criteria for Independence of Electrical Safety Systems," applies as it relates to the physical separation and electrical isolation that must occur between safety-related and not safety-related circuits to maintain the independence of safety-related circuits and equipment so that the safety functions required during and following any DBE can be accomplished.

- RG 1.189, Revision 4, "Fire Protection for Nuclear Power Plants," as it relates to the emergency lighting necessary to support fire suppression actions and safe-shutdown operations, including access and egress pathways to safe-shutdown areas during a fire event.

9.5.3.4 Technical Evaluation

The staff reviewed the information in FSAR, section 9.5.3, to determine whether the plant lighting systems provide adequate lighting during all plant operating conditions and whether the lighting systems can operate without adversely impacting the operation, control, and maintenance of safety-related SSCs.

9.5.3.4.1 Normal Plant Lighting and Normal MCR lighting

SRP section 9.5.3 recommends that the integrated design of the normal lighting system provide adequate station lighting in all plant areas from onsite power sources that are used for control and maintenance of equipment and access routes during normal plant operations. The normal lighting system illumination levels should be conformed with the applicable lighting levels recommended in NUREG-0700.

FSAR Section 9.5.3.2.1, "Normal Plant Lighting," states that the normal plant lighting provides artificial illumination for outdoor areas outside the protected area and within the owner-controlled area, and for plant buildings during normal operation, maintenance, and testing conditions. FSAR Section 9.5.3.1 states that the low voltage AC electrical distribution system (ELVS), which is described in FSAR, Section 8.3.1 with a nominal voltage of 480 VAC, supplies the PLS lighting transformers whose secondary side provides 120 VAC to the lighting fixtures in the plant. Section 9.5.3.1 also states that the plant illumination levels provided by the PLS are in accordance with the applicable lighting levels recommended in NUREG-0700, Revision 3, section 12.2.2.3-1, "Illumination Levels," Table 12.8, "Range of Recommended Illuminances." Table 12.8 provides the illumination levels for the normal lighting in areas where inspection/assembly activities are performed and in other areas inside the plant.

FSAR Section 9.5.3.2.2, "Normal MCR Lighting," states that the normal lighting fixtures in the MCR provide illumination during normal, operating, maintenance, and testing, conditions. The normal lighting fixtures in the MCR are supplied by the PLS 120 VAC to provide the illumination levels in accordance with the applicable lighting levels recommended in NUREG-0700, Revision 3, section 12.1.2.3-1, "General Illumination Levels," through section 12.1.2.3-4, "Task-Specific Illumination Levels." The illumination levels for worksurfaces and for specific tasks in the control room are provided in NUREG 0700, Revision 3, section 12.1.2.3-1 and section 12.1.2.3-4, which includes Table 12.1, "Nominal Illumination Levels For Various Tasks And Work Areas," respectively.

The staff finds that the illumination levels of the normal plant lighting and the normal MCR are acceptable since they are conformed to NUREG-0700, Revision 3, section 12.2.2.3-1, Table 12.8, and section 12.1.2.3-1 through section 12.1.2.3-4.

Based on the above evaluation, the staff finds that the integrated design of the normal plant lighting and normal MCR lighting provide adequate illumination in all plant areas that are used for control and maintenance of equipment and plant access routes during normal plant operations, as recommended by SRP section 9.5.3. Therefore, the staff finds that the normal plant lighting and normal MCR lighting for the NuScale US460 design are acceptable.

9.5.3.4.2 Emergency Plant Lighting and Emergency MCR Lighting

SRP section 9.5.3 recommends that the integrated design of the emergency lighting system provide adequate emergency station lighting in all plant areas required for firefighting, control and maintenance of equipment used for implementing safe shutdown of the plant during all plant operating conditions, and the access routes to and from these areas. The plant operation conditions include fire, transient and accident conditions. SRP section 9.5.3 recommends that the emergency lighting system illumination levels conform with the applicable lighting levels recommended in NUREG-0700, and the effect of an SBO event on the emergency lighting system be addressed.

FSAR Section 9.5.3.2.3, “Emergency Plant Lighting,” states that the emergency lighting outside and inside the MCR provide lighting for accident, transient, and fire conditions. Also, FSAR Section 9.5.3.2.3 states that the design does not require 8-hour emergency lighting fixtures outside of the MCR because no credit is taken for operator actions in the SBO analysis and no post-fire safe shutdown activities requiring operation of safe shutdown equipment is identified in the post-fire safe shutdown analysis. RG 1.189, Revision 4, Position 4.1.6.2, “Post-Fire Safe shutdown,” recommends that fixed, self-contained lighting consisting of units with individual 8-hour minimum battery power supplies be provided in areas needed for operation of safe-shutdown equipment and for access and egress routes to these areas. The staff finds that since the plant design requires no operator actions outside of the MCR for an SBO event and no operation of safe shutdown equipment outside of the MCR for post-fire shutdown activities, the 8-hour battery powered emergency lighting fixtures recommended by RG 1.189, Revision 4, Position 4.1.6.2 are not applicable to the emergency plant lighting outside of the MCR.

Furthermore, FSAR Section 9.5.3.2.3 states that the emergency lighting fixtures outside of the MCR are emergency egress light fixtures with 1.5-hour battery backup for exiting the area. NuScale stated that the emergency lighting for egress conforms to NFPA 101, “Life Safety Code,” 2021 Edition, which is provided in FSAR Table 9.5.1-1, “List of Applicable Codes, Standards and Regulatory Guidance for Fire Protection.” RG 1.1.89, Revision 4, Position 4.1.6.1, “Egress Safety,” recommends that an emergency lighting be provided in support of the emergency egress design guidelines outlined in Position 4.1.2.3, “Access and Egress Design,” of the RG. NFPA 101, 2021 Edition, Section 7.9, “Emergency Lighting, provides guidance on backup power supply for egress lighting. In emergency conditions, NFPA 101 recommends that the emergency lighting for the path of egress provide illumination for a minimum of 1.5 hours in the event of failure of normal lighting. Since the emergency egress lighting fixtures for the NuScale US460 design have a 1.5-hour battery backup for exiting areas, the staff finds that the emergency plant lighting for the NuScale US460 design will provide adequate lighting for exiting areas during loss of normal AC power. Therefore, the staff finds that NuScale has addressed the effect of loss of all AC power (i.e., SBO) on the emergency plant lighting outside of the MCR. The staff also finds that the design provides emergency lighting for egress pathways, as recommended by RG 1.189, Revision 4, Position 4.1.6.1.

FSAR, Section 9.5.3.2.4, “Emergency MCR Lighting,” states that the emergency lighting fixtures in the MCR are continuously on during normal operation, maintenance, testing, transient, and emergency conditions and are supplied by two divisions of the EDAS-C. FSAR, Section 9.5.3.2.4 also states that the emergency lighting fixtures in the MCR will continue to be on during a loss of AC power and the EDAS-C batteries in either division can maintain the emergency lighting at the illumination level of 10 footcandles, as stated in NUREG-0700, Revision 3, for a minimum of 72 hours. NUREG-0700, Revision 3, section 12.1.2.4-4, “Emergency Lighting Levels,” and Table 12.1 provide the illumination of 10 footcandles for a control room emergency lighting. In addition, FSAR section 8.4, “Station Blackout,” states that the SBO coping duration is 72 hours. The staff finds that the emergency illumination level for the MCR lighting system is acceptable since it conforms to NUREG-0700, Revision 3. The staff also finds that NuScale addressed the effect of an SBO event on the emergency MCR lighting system because battery backed lighting is provided in the MCR where operators perform actions critical to plant safe shutdown operation. The staff’s evaluation for an SBO is provided in section 8.4 of this report.

Based on the above evaluation, the staff finds that the integrated designs of the emergency lighting outside of the MCR (egress lighting) and inside of the MCR provide adequate lighting in all plant areas and the access routes to and from these areas during all plant operating conditions such as fire, transient and accident conditions, as recommended by SRP section 9.5.3. Therefore, the staff finds that NuScale US460 emergency plant lighting and emergency MCR lighting are acceptable.

9.5.3.5 Initial Test Program

The staff evaluates the ITP in Section 14.2 of this SER.

9.5.3.6 ITAAC

The staff evaluates ITAAC in Section 14.3 of this SER.

9.5.3.7 Conclusion

The staff reviewed the normal and emergency plant lightings and the normal and emergency MCR lightings in the PLS for conformance with the guidelines of SRP section 9.5.3, NUREG-0700, and RG 1.189, Revision 4. Based on the above technical evaluation, the staff concludes that the normal plant integrated design of the normal plant lighting, the emergency plant lighting, the normal MCR lighting, and the emergency MCR lighting designs provide adequate illumination in all areas of the plant and access routes to these areas under all plant operating conditions such as normal, transient, fire, accident, and SBO conditions, as recommended by SRP Section 9.5.3 and RG 1.189, Revision 4. Therefore, the staff concludes that the designs of the normal and emergency plant lightings and the normal and emergency MCR lightings are acceptable.

9.5.4 Backup Diesel Generator Auxiliary Systems

9.5.4.1 Introduction

The onsite power systems for NuScale Power Plant (NPP) US460 design consist of a normal power distribution system and a backup power supply system (BPSS). The BPSS includes two backup diesel generators (BDGs). NuScale addressed the BDGs in FSAR Section 8.3, “Onsite Power Systems.” SRP sections 9.5.4 through 9.5.8 provide guidance for the emergency diesel generators and their support systems for light water reactors. To maintain consistency with the applicable SRP numbers, the staff elected to review the BDGs under Section 9.5.4 of this safety evaluation report.

9.5.4.2 Summary of Application

10 CFR 50 Appendix A Criterion 17 mandates in part that the onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure. NUREG-0800 (SRP), Sections 9.5.4 through 9.5.8 provide guidance for the review of various essential elements of the emergency diesel engine sets designated as safety-related equipment. NuScale used the terminology “backup diesel generators (BDGs)” for the onsite diesels described in Section 8.3 “Onsite Power Systems.” FSAR Table 1.3-1 indicates that NPP US460 design has no emergency diesel generators. FSAR Section 8.3.1.1, System Description, states that BPSS consists of two backup diesel generators. FSAR Section 8.3.1.1.1, BPSS, states that the principal function of the nonsafety-related BPSS is to provide electrical power to the plant when the normal sources of AC power are not available and further describes the functions it performs in conjunction with the BDGs as follows:

- The BDGs provide backup electrical power to the augmented DC power system and selected loads from various plant systems.
- The BPSS can provide backup electrical power to loads supporting beyond design basis accident mitigation and performing a black start to recover from total shutdown of all turbine generators without reliance on an external transmission grid.
- The BPSS delivers backup power to heating, ventilation, and air conditioning systems serving the battery and associated charger rooms to avoid prolonged periods of high ambient temperature.
- The BPSS can support other select non-safety-related, non-risk significant loads that provide asset protection and operational flexibility.

The BDGs functions are not safety-related, are not risk-significant, are not credited for the mitigation of design-basis accidents and do not include any safe shutdown functions. There are no safety-related components in the BDGs, and failure of the BDGs to operate does not prevent structures, systems, and components (SSCs) from performing their safety-related functions.

There are no proposed inspections, tests, analyses, and acceptance criteria (ITAAC) related to the BDGs. Information regarding initial testing of the BPSS, including the BDGs, is provided in the FSAR Section 14.2 (Test # 52).

The BPSS is not safety-related or risk-significant.

9.5.4.3 Regulatory Basis

The staff reviewed the BDGs in accordance with the following Regulatory Guides (RGs), and SRP Sections, which provide relevant guidance for the safety-related onsite emergency diesel generators and their support systems:

- RG 1.9, Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants, Revision 4 March 2007 (ADAMS Accession Number ML070380553)
- RG 1.137, Fuel Oil Systems for Emergency Power Supplies, Revision 2 June 2013 (ADAMS Accession Number ML12300A122)
- SRP 9.5.4, Emergency Diesel Fuel Oil Storage and Transfer System, Revision 3 March 2007 (ADAMS Accession Number ML070680388)
- SRP 9.5.5, Emergency Diesel Engine Cooling Water System, Revision 3 March 2007 (ADAMS Accession Number ML070550035)
- SRP 9.5.6, Emergency Diesel Generator Starting System, Revision 3 March 2007 (ADAMS Accession Number ML070550034)
- SRP 9.5.7, Emergency Diesel Engine Lubrication System, Revision 3 March 2007 (ADAMS Accession Number ML070460354)
- SRP 9.5.8, Emergency Diesel Engine Combustion Air Intake and Exhaust System, Revision 3 March 2007 (ADAMS Accession Number ML070550033)

The following criteria provide relevant guidance for this area of review:

- GDC 2 stipulates in part that design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

- GDC 4 stipulates in part that environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.
- GDC 5 stipulates that sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.
- GDC 17, as it relates to the BDG Auxiliary Systems, stipulates in part, that electric power systems. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

  The onsite electric power supplies, including the batteries, and the onsite electric distribution system shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

  Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

- GDC 44, as it relates to the BDG cooling water system, stipulates in part that cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

  Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that 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) the system safety function can be accomplished, assuming a single failure.

- GDC 45, as it relates to the BDG cooling water system stipulates that inspection of cooling water system. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

9.5.4.4 Technical Evaluation

FSAR Section 1.1 states that the design features of the Nuclear Power Modules do not require alternating current (AC) or direct current (DC) for safe shutdown and cooling. FSAR Section 8.3, “Onsite Power Systems”, states the plant safety-related functions are achieved and maintained without reliance on electrical power; therefore, neither the AC power systems nor the DC power systems are safety-related (Class 1E). FSAR Section 8.3 furthers states that the onsite power systems do not perform any risk-significant functions.

FSAR Table 1.9-2, Conformance with Regulatory Guides, states that RG 1.137 is not applicable because the design does not rely on or include safety-related diesel generators.

FSAR Table 1.9-3, “Conformance with NUREG-0800, Standard Review Plan and Design Specific Review Standard”, states that the design does not require or include safety-related emergency diesel generators that are subject to SRP sections 9.5.4 through 9.5.8 and that no AC or DC power is relied upon for the performance of NPP safety functions. Accordingly, all GDCs mentioned in the SRPs are not applicable.

The staff reviewed the FSAR to determine if the failure of the BDGs could potentially have an adverse impact on important-to-safety SSCs, or on the plant’s ability to achieve and maintain shutdown. Based on FSAR Figures 1.2-1 and 1.2-2, the BDGs are located within their own structures. All SSCs important to safety are located inside the Reactor Building (RXB) and Control Building (CRB). The staff finds that, since the BDGs are not safety-related and serves no safety-related functions, the requirements for diesel support systems operation, including the referenced GDCs in the SRP Sections 9.5.4 through 9.5.8 are not applicable to the BDGs.

Technical Specifications: There are no specific technical specification requirements associated with the BDGs.

9.5.4.5 Conclusion

Based on the evaluation, the staff notes that the guidance provided in SRP Sections 9.5.4 through 9.5.8 is not a regulatory requirement for the Backup Diesel Generator Auxiliary Systems of the NPP US460 design.

As stated in Section 9.5.4.3 above:

> The BDGs functions are not safety-related, are not risk-significant, are not credited for the mitigation of design-basis accidents and do not include any safe shutdown functions. There are no safety-related components in the BDGs, and failure of the BDGs to operate does not prevent structures, systems, and components (SSCs) from performing their safety-related functions.

Therefore, it follows that the BDG Auxiliary Systems are not required to meet GDC 2, GDC 4, GDC 5, GDC 17, GDC 44 and GDC 45.