9 AUXILIARY SYSTEMS

This chapter of the final safety evaluation report (FSER) 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), Design Certification Application (DCA), Part 2, "Final Safety Analysis Report (FSAR)." The staff's regulatory findings documented in this report are based on Revision 5 of the DCA, dated July 29, 2020 (Agencywide Document Access and Management System (ADAMS), Accession No. ML20225A071). The precise parameter values, as reviewed by the staff in this safety evaluation, are provided by the applicant in the DCA 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 DCA and not converted.

In this chapter, the NRC staff uses the term "nonsafety-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." However, among the nonsafety-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

The staff reviewed DCA Part 2, Tier 2, Revision 3, Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and the associated technical report TR-0816-49833, Revision 1, "Fuel Storage Rack Analysis," issued November 2018 (ADAMS Accession No. ML18310A154), to ensure that subcriticality will be maintained when storing and handling new and spent fuel in the onsite fuel storage and handling facility, which is located in the reactor building (RXB). New and spent fuel will be stored in the fuel storage racks in the borated spent fuel pool (SFP). The SFP is part of the ultimate heat sink (UHS) and is partially separated from the other parts of the UHS (the reactor pool and refueling pool (RFP)) by a weir, which is designed to maintain at least 3 meters (m) (10 feet (ft)) of water above the fuel storage racks. The fuel storage racks incorporate neutron absorber plates and geometric spacing to maintain subcriticality and are designed without any requirements for burnup, zoning, or loading patterns.

To assess whether the fuel storage racks maintain subcriticality for all credible storage conditions, the staff focused its review primarily on the inputs, assumptions, and methodology used in the applicant's criticality analysis and the acceptability of the analysis results. The staff also reviewed storage rack materials, including neutron absorber effectiveness, as well as criticality during fuel handling.

9.1.1.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 3.5, "Fuel Storage System," contains the DCA Part 2, Tier 1, information associated with this section. DCA Part 2, Tier 1, Section 3.5, includes a design commitment that the fuel storage racks maintain the effective neutron multiplication factor (k-effective or k_{eff}) within limits that are consistent with 10 CFR 50.68(b)(4), which states, in part, the following:

> If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

The Tier 1 information related to DCA Part 2, Tier 2, Section 9.1.1, is evaluated in Section 14.3 of this SER.

DCA Part 2, Tier 2: The applicant provided a system description in DCA Part 2, Tier 2, Section 9.1.1. Technical report TR-0816-49833 documents the detailed criticality analysis of the fuel storage racks and is incorporated by reference into the DCA in DCA Part 2, Tier 2, Table 1.6-2, "NuScale Referenced Technical Reports." The paragraphs below summarize the information in DCA Part 2, Tier 2, and TR-0816-49833 relevant to criticality safety of new and spent fuel storage and handling.

DCA Part 2, Tier 2, Section 9.1.1, provides the design basis for new and spent fuel storage and handling, a fuel storage facility description, and high-level information from the criticality analysis for new and spent fuel storage. The applicant stated that it considered the following requirements and guidance, which the staff discusses in Section 9.1.1.3 of this SER, 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"
- American National Standards Institute (ANSI)/American Nuclear Society (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"
- ANSI/ANS 57.3-1983, "Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants"

The applicant also stated that geometrically safe configurations and plant programs and procedures prevent inadvertent criticality in the fuel storage racks and during fuel handling.

The fuel storage racks prevent criticality through geometric spacing of fuel assemblies, use of boron carbide-aluminum metal matrix composite fixed neutron absorber plates, and chemical additions of boron to UHS water. The 14 fuel storage racks are freestanding and consist of 11 x 11 arrays of square, stainless steel fuel storage tubes with one neutron absorber plate on

the outside of each tube surface, or four neutron absorber plates per tube. Stainless steel spacers placed at three different elevations separate each fuel storage tube from the adjacent fuel storage tubes. The stainless steel spacers are kept in place by welding the ends of each spacer to connecting bars on the outside of the fuel storage rack. The spacers and the weldments securing the spacers ensure that the fuel tubes, the neutron absorbers, and the flux traps are maintained in the geometric configuration described in the technical report. A neutron absorber monitoring program verifies the ability of the neutron absorber material to provide the credited criticality control throughout its lifetime. In general, the design of the NuScale fuel rack is similar to that of operating plants, the only significant difference being the height of the fuel rack.

Up to 1,404 fuel assemblies can be stored in the 1,694 total locations in the SFP because of the travel limitations of the fuel handling machine (FHM). However, due to difficulty reaching the 11 storage locations nearest the weir wall, the applicant considers only 1,393 locations accessible. The criticality analysis conservatively assumes that all rack locations are full, even if they are not accessible. New and spent fuel with initial enrichment up to 5 weight percent uranium-235 may be stored in the racks without restrictions on burnup, zoning, or loading patterns.

For criticality prevention during fuel handling, the applicant stated that the geometrically safe designs of the fuel handling equipment (FHE) allow each piece of equipment to move only one fuel assembly at a time. In addition, Combined License (COL) Information Item 9.1-1 directs a COL applicant to develop plant programs and procedures for safe operations during the handling and storage of new and spent fuel assemblies, including criticality control.

TR-0816-49833 contains the detailed criticality analysis for normal and abnormal conditions in the fuel storage racks and describes the applicant's analytical methodology, methodology validation, analysis assumptions, determination of bias and uncertainty, and analysis results. The applicant used the SCALE 6.1.3 computer code system with nuclear data from the Evaluated Nuclear Data File (ENDF)/B-VII.0 238-group cross section library for its calculations and stated that it had validated the code and data following NRC guidance. The applicant concluded that the calculated k_{eff} values, including bias and uncertainty, remain within the limits in 10 CFR 50.68(b)(4).

ITAAC: The applicant gave the inspections, tests, analyses, and acceptance criteria (ITAAC) associated with DCA Part 2, Tier 2, Section 9.1.1, in DCA Part 2, Tier 1, Table 3.5-1, "Fuel Storage System Inspections, Tests, Analyses, and Acceptance Criteria." These ITAAC are evaluated in Section 14.3 of this SER.

Initial Test Program: No initial tests are proposed related to DCA Part 2, Tier 2, Section 9.1.1.

Technical Specifications: The generic technical specifications (GTS) associated with DCA Part 2, Tier 2, Section 9.1.1, appear in the NuScale DCA Part 4, "Generic Technical Specifications—NuScale Nuclear Power Plants," and include GTS 3.5.3, "Ultimate Heat Sink," GTS 4.3, "Fuel Storage," and GTS 5.5.12, "Spent Fuel Storage Rack Neutron Absorber Monitoring Program."

Technical Reports: TR-0816-49833, Revision 1, provides the criticality analysis, including inputs, assumptions, methodology, methodology validation, and analysis results.

9.1.1.3 Regulatory Basis

The relevant requirements for the Commission regulations for criticality safety of fresh and spent fuel storage and handling, and the associated acceptance criteria, are identified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [light-water reactor] Edition" (SRP), Section 9.1.1, "Criticality Safety of Fresh and Spent Fuel Storage and Handling," and are summarized below. Review interfaces with other SRP sections can be found in NUREG-0800, Section 9.1.1.

- 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 52.47(a)(17), which requires the applicant to provide information demonstrating how it will comply with the requirements for criticality accidents in 10 CFR 50.68(b)(2)–(b)(4)

The related acceptance criteria are as follows:

- The criteria for GDC 62 are specified in ANSI/ANS 57.1, 57.2, and 57.3, 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 that a criticality accident will occur.

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

- DSS-ISG-2010-01, Revision 0, "Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools," dated September 29, 2011 (ADAMS Accession No. ML110620086)
- 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

The staff reviewed DCA Part 2, Tier 2, Section 9.1.1, and TR-0816-49833 in accordance with SRP Section 9.1.1 to ensure compliance with the regulatory requirements listed in Section 9.1.1.3 of this SER. The staff evaluated important aspects of the applicant's criticality analysis, including fuel assembly and storage rack design data, computational methods and data, computational method validation, bias and uncertainty analysis, the scope of normal and abnormal conditions analyzed, analysis models and assumptions, and analysis results. The staff also reviewed aspects of storage rack materials, including neutron absorber effectiveness

and fuel handling. In addition, as discussed in Section 9.1.1.5 of this SER, the staff evaluated the GTS as they relate to DCA Part 2, Tier 2, Section 9.1.1.

9.1.1.4.1 Fuel Assembly Modeling

The staff reviewed the fuel assembly design information in TR-0816-49833, including dimensions and materials, and determined that the information is consistent with the design information for the NuScale fuel assembly (NuFuel HTP2). Because NuFuel HTP2 is the only assembly design proposed for use in the NuScale Power Module (NPM), there are no other types of fuel design for the applicant to consider in the criticality analysis.

The criticality analysis conservatively assumes all fuel rods contain the maximum uranium-235 enrichment allowed by regulations, 5.0 weight percent. In addition, the fuel assemblies are conservatively modeled without axial blankets, fuel pellet dishes and chamfers, and burnable poisons. These assumptions maximize reactivity and are therefore acceptable to the staff. In addition, the staff audited some of the applicant's calculation input files and confirmed that the fuel assembly inputs are consistent with the methodology and data described in TR-0816-49833 (see the audit documentation at ADAMS Accession Nos. ML18025A857 and ML18025B026).

For the reasons given above, the staff finds that the applicant used appropriate fuel assembly design information and conservative modeling assumptions for the fuel assemblies.

9.1.1.4.2 Storage Rack Modeling

The staff reviewed the fuel storage rack design information in TR-0816-49833, including dimensions, materials, and drawings, to ensure the information is complete and conservatively incorporated into the criticality analysis. The staff determined that the fuel storage rack information is comprehensive and the listed material properties are consistent with known material compositions. The staff confirmed through an audit (see the audit documentation at ADAMS Accession Nos. ML18025A857 and ML18025B026) that a selection of the applicant's original criticality analysis input files incorporates rack inputs that are consistent with the rack design information in TR-0816-49833, Revision 0, except for a few modeling errors, such as incorrect rack spacing and incorrect storage tube thickness for some input files. The applicant corrected the errors and updated the analysis in Revision 1 of TR-0816-49833 accordingly. These actions acceptably addressed the staff's concern.

In addition, the staff determined that, with the exception of certain assumptions associated with structural components, the modeling assumptions are appropriately conservative. For example, the applicant did not credit the full design boron (B)-10 areal density in the neutron absorber plates, and the model boundary conditions account for the phenomenon of neutron reflection by the concrete SFP walls. However, the applicant simplified the storage rack model by omitting some structural components, which the staff noted may be nonconservative. To assess the reactivity effect of the omission, the applicant performed sensitivity studies that showed that omitting the structural material above and below the fuel had a negligible effect, but omitting the support tubes in the flux trap region was nonconservative. The applicant used the results of the sensitivity studies to determine a bias and bias uncertainty related to structural materials and incorporated the structural material bias and bias uncertainty into the criticality calculations in TR-0816-49833. The staff finds this approach acceptable because the applicant accounted for the reactivity effects of omitted structural materials in the final calculations of k_{eff} at a 95-percent probability and 95-percent confidence level ($k_{95/95}$). In addition, the staff audited the sensitivity studies and confirmed that the applicant's models were sufficient to determine the reactivity

effect of structural materials, and the applicant's calculation results were consistent with the associated bias and bias uncertainty reported in TR-0816-49833 (see the audit documentation at ADAMS Accession Nos. ML18025A857 and ML18025B026).

For the reasons given above, the staff finds that the applicant used appropriate storage rack design information and reasonable modeling assumptions for the storage racks.

9.1.1.4.3 Storage Rack Materials

9.1.1.4.3.1 Conformance with Industry Standards

The fuel racks are fabricated from [[ ]] and [[ ]]. Both stainless steel types have extensive operational history in SFP and reactor environments. The applicant used computer codes that appropriately model the neutronic behavior of [[ ]], as further discussed in SER Section 9.1.1.4.4. The [[ ]] bearing plates have no impact on the criticality evaluation because the bearing plates are located below the active fuel region of the fuel racks.

The applicant will procure the [[ ]] material as plate using the American Society of Mechanical Engineers (ASME) [[ ]]. The staff reviewed [[ ]], which contains standard dimensional tolerances for plate material. The staff determined that the criticality evaluation incorporates the expected uncertainties of materials procured to those standards; therefore, certification that the procured stainless steel meets the material specification is sufficient to conclude that the material conforms to the assumptions in the criticality analysis.

The staff reviewed the quality assurance measures associated with the fabrication of the fuel racks. The applicant committed to conforming to ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications," during fabrication. Compliance with ASME NQA-1 includes meeting ASME NQA-1, Subpart 2.1, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants." The staff previously endorsed ASME NQA-1 with conditions in Regulatory Guide (RG) 1.28, "Quality Assurance Program Criteria (Design and Construction)." The staff concludes that ASME NQA-1, with the conditions described in RG 1.28, provides sufficient quality assurance controls for the fuel racks, and ASME NQA-1, Subpart 2.1, contains sufficient requirements to prevent contamination of neutron absorber and stainless steel materials.

9.1.1.4.3.2 Stainless Steel Materials

As discussed in SER Section 9.1.2, the neutron absorber and the chemistry of the UHS do not have a deleterious effect on the stainless steel. Contact between the neutron absorber and stainless steel materials will not degrade the stainless steel material because the stainless steel is more "noble" than the aluminum matrix of neutron absorber. Controlling the chemistry of the UHS prevents the corrosion of stainless steel, as described in SER Section 9.1.3. For these reasons, the staff finds the use of [[ ]] stainless steels in the spent fuel racks acceptable. The material selection and fabrication controls as well as chemistry control are sufficient to prevent degradation of the stainless steel in the spent fuel racks in all normal, off-normal, and design-basis accident (DBA) conditions.

9.1.1.4.3.3 Neutron Absorber Material

As discussed below, the staff reviewed the neutron absorber material to determine whether the fabrication, qualification, monitoring, and modeling of the material are adequate. The staff also reviewed the applicant's use of lessons learned from boral and Boraflex degradation in the design of the spent fuel racks and agrees that, in general, this class of material is not expected to experience the issues related to boral or Boraflex, but material testing, as specified in COL Item 9.1-9, will confirm the performance of the material as it relates to degradation in the UHS.

9.1.1.4.3.3.1 Neutron Absorber Material—Fabrication and Licensing Basis

According to DCA Part 2, Tier 2, Section 9.1.2.3.5, the design uses an aluminum and boron carbide metal matrix composite material for criticality control. The applicant did not identify a specific neutron absorber material within this class but provided a set of performance and physical criteria derived from assumptions used in the criticality safety analysis. This class of material has been used successfully in operating plants. For example, the material called Metamic™ has been approved by the staff as a neutron absorber in fuel storage racks, subject to an evaluation of the compatibility and chemical stability with the exposure conditions; specification of uncertainties and tolerances of the rack design; justification for the assumption of uniform boron distribution; and demonstration of neutron attenuation capability through a monitoring program, in accordance with SRP Section 9.1.1.

TR-0816-49833 describes the applicant's design requirements for the neutron absorber material, including the chemical content, mechanical properties of the material, dimensions of the plates, and the characteristics of the boron carbide component. These requirements provide high-level acceptance criteria corresponding to compatibility with the exposure conditions, adequate strength and ductility in processing and exposure, sizing consistent with the criticality analysis, and neutron attenuation consistent with the criticality analysis. The staff finds the design requirements for the neutron absorber material acceptable because the criticality analysis shows that the specified values are sufficient to meet the criticality requirements in 10 CFR 50.68.

The design includes COL Item 9.1-9 to demonstrate that the specified properties of the selected material and the as-manufactured neutron absorber products meet the acceptance criteria described in the criticality analysis. COL Item 9.1-9 directs a COL applicant to provide a qualification report demonstrating that the neutron absorber material can meet the neutron attenuation and environmental compatibility functions described in TR-0816-49833. COL Item 9.1-9 also directs a COL applicant to establish procedures to inspect the as-manufactured material for contamination and manufacturing defects and evaluate the neutron attenuation uncertainty corresponding to the material lot variability.

Based on its review of the information in the DCA, including TR-0816-49833, the staff finds that the selection and control of the neutron absorber material offers reasonable assurance that the neutron absorber can provide the neutron attenuation assumed in the criticality analysis.

9.1.1.4.3.3.2 Neutron Absorber Material—Monitoring Program

Operational experience with boral and Boraflex has shown that a neutron absorber monitoring program is necessary to ensure that the spent fuel racks continue to meet the requirements of GDC 62 and 10 CFR 50.68. A coupon monitoring program must provide the ability to assess whether the neutron-absorbing element (B-10) is being lost and whether the neutron absorber is experiencing dimensional changes.

The applicant stated that the neutron absorber monitoring program will be established in accordance with ASTM C1187-2007, "Standard Guide for Establishing Surveillance Test Program for Boron-Based Neutron Absorbing Material Systems for Use in Nuclear Fuel Storage Racks in a Pool Environment." Currently operating nuclear power plants have used this consensus standard, which is applicable to the NuScale design. The staff reviewed the coupon examination schedule and found it to be consistent with previously accepted coupon monitoring programs. The quantity of coupons is sufficient for the 40-year licensing period of the NuScale design certification. Two spare coupons are provided for license renewal purposes; however, the staff notes that the withdrawal schedule for the coupons is based on the first irradiated fuel insertion into the rack. It is feasible that a COL applicant will acquire 12 NuScale modules in two different procurements, resulting in a set of staggered start dates for a multimodule site. If this is the case, the design life of the fuel rack should not be tied to a single particular module but rather the time from which spent fuel was first loaded into the fuel rack.

The applicant described the coupon monitoring program in TR-0816-49833, including (1) the parameters to be inspected, (2) the acceptance criteria for the inspections, and (3) the actions to be taken if the inspected parameters do not meet the acceptance criteria. The coupon inspections allow a licensee that references the NuScale design to detect a reduction in neutron attenuation (via the loss of B-10 or the existence of an air bubble that could decrease neutron moderation) or corrosion. Photography of the coupons allows for the tracking of changes in the material's appearance throughout the lifetime of the spent fuel rack. The staff finds that the coupon inspection parameters are sufficient to ensure that the method of controlling criticality in the fuel rack is appropriately monitored during the lifetime of the plant.

The staff reviewed the acceptance criteria for the coupon monitoring program to determine whether the minimum and maximum acceptable values would result in continued compliance with the licensing basis. In comparison to other certified designs that have credited 75 percent of the design B-10 areal density in the neutron absorber, [[

]] The staff finds that the tradeoff between a stricter monitoring program acceptance criterion and less conservatism in the amount of B-10 credited in the criticality analysis is acceptable because, as described in the coupon monitoring program, a licensee that references the NuScale design will have to perform an engineering evaluation if the B-10 areal density is below the 95-percent acceptance criteria. The higher amount of B-10 credited in the neutron absorber was reviewed with respect to maintaining overall safety margins in the fuel rack design. The staff concluded that the fuel rack analysis has significant criticality margin inherent in the design calculations and assumptions. SER Sections 9.1.1.4.1, 9.1.1.4.2, and 9.1.1.4.5 through 9.1.1.4.8 discuss the conservatism in the criticality analysis. Additionally, the reported as-fabricated B-10 uses the lower bound 95/95 confidence limit of the entire lot of material, thereby ensuring that the mean material has additional B-10 that is not credited in the design. The staff concludes that examinations for pitting, general corrosion, dimensional changes, and B-10 areal density are sufficient to initiate an engineering evaluation if evidence of degradation is found.

In addition, GTS 5.5.12 states that the neutron absorber monitoring program shall be in accordance with Nuclear Energy Institute (NEI) 16-03-A, Revision 0, "Guidance for Monitoring of Fixed Neutron Absorbers in Spent Fuel Pools," issued May 2017 (ADAMS Accession No. ML17263A133). The staff finds this approach for GTS 5.5.12 acceptable because the staff found in its review of NEI 16-03-A that a neutron absorber monitoring program that implements NEI 16-03-A provides reasonable assurance that degradation of neutron-absorbing material can

be detected, thereby maintaining the ability of the neutron-absorbing material to provide the credited criticality control.

The staff concludes that the neutron absorber monitoring program is sufficient to detect, track, and trend changes to the neutron absorber material that could indicate an active degradation mechanism. The staff finds that the structure of the coupon monitoring program described in DCA Part 2, Tier 2, Section 9.1.2, "New and Spent Fuel Storage," is consistent with monitoring programs in currently operating plants and considers operating experience from the LWR fleet. Based on this, the staff concludes that the neutron absorber monitoring program will maintain compliance with 10 CFR 50.68, GDC 4, "Environmental and Dynamic Effects Design Bases," and GDC 62 during the lifetime of the fuel rack.

9.1.1.4.4 Computational Methods and Data

The applicant used the SCALE 6.1.3 computer code system and nuclear data from the ENDF/B-VII.0 238-group cross-section library for its criticality analysis. The applicant used the CSAS5 sequence to drive the cross-section processing modules and the KENO V, a Monte Carlo solution method for calculating k_{eff}.

Oak Ridge National Laboratory developed the SCALE code package with funding from the NRC. The staff notes that the use of SCALE to demonstrate compliance with the regulatory criteria outlined in SRP Section 9.1.1 is within the code's intended applications and is similar to the confirmatory analysis methods used by the staff, as well as the analysis methods for other licensing applications, including fuel storage criticality analyses. In addition, the cross-section library is the most recent version of the ENDF library included in the SCALE 6.1.3 code package. For these reasons, the staff finds the applicant's computational methodology and data acceptable.

9.1.1.4.5 Computational Method Validation

The purpose of validating the computational methods is to determine the bias and bias uncertainty resulting from the methods, code, and cross-section library. The bias in a criticality code is determined by comparing benchmark experiment measurements to calculated results for those experiments. The bias may be correlated to parameters such as fuel enrichment, neutron absorber content, and fuel pin pitch; if so, the trends must be properly accounted for in the bias. The code bias and bias uncertainty are then used to conservatively adjust the calculated k_{eff}. TR-0816-49833 describes the applicant's code validation and cites the guidance in NUREG/CR-6698.

The applicant selected a total of 101 benchmark experiments for the validation: 99 from the International Handbook of Evaluated Criticality Safety Benchmark Experiments (ICSBEP Handbook) and 2 from NUREG/CR-6361. The ICSBEP Handbook contains some of the most complete, peer-reviewed experimental data available for benchmarking purposes, and the experiments in NUREG/CR-6361 are intended for use in benchmarking for LWR fuel applications. In addition, the staff reviewed the experiments chosen for the validation and determined they are representative of the materials, conditions, and parameters modeled in the NuScale criticality analysis. For these reasons, the staff finds the selected experiments appropriate for the NuScale criticality code validation.

The applicant selected subsets of the 101 experiments to analyze for possible trends. As recommended by DSS-ISG-2010-01, the applicant performed a trend analysis for each parameter used to define the area of applicability of the validation: enrichment, fuel pin pitch,

fuel assembly separation, soluble boron content, boron areal density in the neutron absorber plates, moderator-to-fuel ratio, and neutron spectrum (represented by the energy of average lethargy of fission). The staff reviewed each subset to ensure the experiments chosen represent a comprehensive range for the parameter of importance in the subset.

As described in NUREG/CR-6698, regression fits to the datasets can be used to identify trends. The statistical method used to determine the code bias and bias uncertainty depends on the characteristics of the dataset. If the dataset follows a normal distribution, a single-sided tolerance limit should be used in the absence of an apparent trend, but a tolerance band should be used when a valid trend exists. These limits, above which the true population of k_{eff} is expected to lie, are denoted as K_L and inherently capture both the bias and the bias uncertainty from the code. If the dataset is not normally distributed, a nonparametric statistical treatment should be used to determine K_L.

As a result of staff questions about the applicant's original conclusion that no additional bias is warranted by any trend, the applicant included a more rigorous statistical analysis in TR-0816-49833, Revision 1. The applicant identified statistically significant trends for enrichment, assembly separation, and B-10 areal density in separator plates. Of those, only the B-10 areal density dataset was normally distributed. In addition, the applicant concluded that the set of all 101 experiments was not normally distributed. Based on a confirmatory statistical analysis, the staff agrees with these conclusions.

Although NUREG-6698 recommends calculating tolerance bands only for normally distributed datasets that exhibit statistically significant trends, the applicant calculated tolerance bands at a 95-percent confidence level for each trending parameter regardless of normality and presence of a trend. The applicant determined that the most limiting result came from the tolerance band for assembly separation. The applicant claimed that the assembly separation tolerance band is valid because the dataset failed the expanded Shapiro-Wilk normality test only by a small amount (indicating a slightly lower confidence that the data are normally distributed than provided by meeting the selected significance level for the test), and the tolerance band clearly bounds the data points by a significant margin. Although a nonparametric analysis is desired for nonnormal datasets, the staff notes that the applicant's calculated tolerance band visually bounds the data, and some tests (including the Chi-square test and the D'Agostino-Pearson omnibus test) did not provide sufficient evidence to reject the hypothesis of normality, suggesting the data may be reasonably approximated as normal. In addition, the applicant took the assembly separation K_L as the minimum value across the entire tolerance band, which accounts for varied assembly separation distances. This K_L is conservative relative to the higher K_L at the NuScale design value of fuel assembly separation. Finally, if a nonparametric treatment were used and the nonparametric margin suggested in NUREG/CR-6698, Table 2-2, were interpolated, the K_L selected by the applicant would bound the nonparametric K_L. For these reasons, the staff accepts the applicant's K_L resulting from the assembly separation trend.

Consistent with the guidance in NUREG/CR-6698, the applicant performed a nonparametric analysis for the set of all 101 experiments. The applicant stated that this also covered the nonparametric analysis for the enrichment dataset. The staff disagrees, noting that the full set of 101 experiments uses the second smallest sample to determine K_L, while the smaller size of the enrichment dataset requires use of the smallest sample, resulting in a less conservative K_L. However, this is of no consequence in the final determination of code bias because the applicant used the most limiting K_L from the entire validation study (0.97654), which resulted from the assembly separation trend. For the reasons stated above, the staff concludes that the applicant has calculated a reasonably conservative code bias.

The staff reviewed the area of applicability of the validation in Table 3-25 of TR-0816-49833. The staff finds that the applicant correctly determined the area of applicability because the ranges described for the parameters are consistent with the parameter ranges for the experiments used to determine the code bias and bias uncertainty. In addition, the staff notes that the NuScale design and models are within the defined area of applicability.

The guidance in NUREG/CR-6698 cautions against using code options that are not validated in criticality safety analyses. Although the applicant did not validate the water and concrete albedo boundary conditions used in its SFP criticality models, in its letter dated May 22, 2018 (ADAMS Accession No. ML18142C221), the applicant described the results of sensitivity studies that investigated the effects of three different cases using validated boundary conditions in its SFP criticality models. The differences in k_{eff} from using the albedo boundary conditions compared to the other three options were small and generally conservative, and the difference for the one nonconservative case was less than 1 percent of the manufacturing tolerance calculated for the unborated SFP. The staff concludes that the differences are conservative or negligible, or both; therefore, use of the albedo boundary conditions is acceptable for the NPM criticality analyses.

The staff concludes that the applicant's code validation methodology is generally consistent with the guidance in NUREG/CR-6698, and the staff has reasonable assurance that the resulting code bias and bias uncertainty values are conservative.

9.1.1.4.6 Bias and Uncertainty Analysis

A criticality analysis must account for all relevant sources of bias and uncertainty to conclusively demonstrate compliance with 10 CFR 50.68. To calculate $k_{95/95}$, the applicant used Equation 1 in Section 3.3.1 of TR-0816-49833. The bias terms in the equation include bias from variation in the system and base case modeling parameters (in particular, moderator temperature and structural material) and computer code bias. The uncertainty terms include the Monte Carlo calculation uncertainty, the bias uncertainty associated with variation of system and base case modeling parameters, uncertainty because of manufacturing tolerances, and the Monte Carlo uncertainty resulting from the manufacturing tolerance calculations. As noted in Section 9.1.1.4.5 of this SER, the code bias is calculated at 95/95, eliminating the need for a separate code bias uncertainty term. The staff notes that Equation 1 includes a comprehensive set of biases and uncertainties and is adequate to provide an accurate calculation of $k_{95/95}$.

The applicant included the bias and bias uncertainty from variation in moderator temperature because the applicant performed the criticality analysis at a temperature of 19.85 degrees Celsius (°C) (67.73 degrees Fahrenheit (°F)), corresponding to a moderator density of 0.9982 grams per cubic centimeter, although the actual normal operation condition extends down to 4.4 °C (40 °F) and a corresponding density of 1.0 gram per cubic centimeter. The staff notes that assuming the highest moderator density possible is limiting for k_{eff} because of enhanced moderation. Although the staff would normally expect the criticality analysis to be performed at the limiting temperature and density, the staff accepts the adjustment of k_{eff} with a bias and bias uncertainty to account for the temperature and density difference because the bias effectively serves as a penalty based on sensitivity calculations using varying temperatures and the corresponding densities. The staff reviewed the moderator temperature bias and bias uncertainty calculations and finds them acceptable because they use appropriate mathematical and statistical methods.

The applicant chose to incorporate a bias and bias uncertainty because of the nonconservative effect of neglecting storage rack structural material in the models, as discussed in Section 9.1.1.4.2 of this SER. The staff confirmed the appropriateness of the bias and bias

uncertainty values through an audit of the applicant's sensitivity calculations that formed the basis for those values (see the audit documentation at ADAMS Accession Nos. ML18025A857 and ML18025B026).

To account for manufacturing tolerances, the staff considers it acceptable to either (1) analyze a worst case combination of mechanical tolerances and material conditions to maximize k_{eff} or (2) perform a sensitivity study of the reactivity effects of the various tolerances. The applicant chose the second approach. The applicant's model for examining tolerances consists of a single fuel storage rack with periodic boundary conditions on the x- and y-faces, water above and below the rack structures, and a water albedo on the z-faces. This arrangement is conservative because it effectively creates an infinite array of storage racks. The applicant varied mechanical parameters one at a time to determine the reactivity effects for both unborated and borated moderator. The staff finds the tolerances analyzed, as listed in Tables 3-8 and 3-9 of TR-0816-49833, to be comprehensive and therefore acceptable. In addition, the staff confirmed that the applicant used suitable methods to combine the reactivity effects of the tolerances.

The staff concludes that the applicant's criticality analysis conservatively accounts for all relevant sources of bias and uncertainty. Furthermore, the staff independently confirmed that the applicant combined the aforementioned biases and uncertainties correctly and applied them to the nominal k_{eff} values to determine $k_{95/95}$ for each analysis. Therefore, the staff finds the applicant's treatment of bias and uncertainty acceptable.

9.1.1.4.7 Normal Conditions

The applicant used its normal-conditions model to analyze nominal conditions in the SFP, including the storage of up to five damaged fuel assemblies in the racks. The normal-conditions model is a finite array model of the whole SFP and consists of 14 individual racks with minimum spacing between racks and nominal spacing to the SFP walls. The model includes a concrete albedo on the x- and y-faces to model the SFP walls and a water albedo on the z-faces. This model is reflective of the actual SFP arrangement except that it conservatively assumes that all 1,694 rack locations are full, even though not all rack locations are accessible. All fuel assemblies are assumed to be enriched to 5 percent uranium-235 by weight, and no credit is taken for burnup of the fuel assemblies.

As discussed in Section 9.1.1.4.5 of this SER, the staff concludes that the applicant's use of water and concrete albedos is acceptable. Also, as mentioned in Section 9.1.1.4.6 of this SER, the normal-conditions model does not use the most limiting, lowest normal operating temperature, but the applicant included a bias and bias uncertainty due to moderator temperature. The staff accepts this treatment because the effects of the temperature difference are adequately taken into account.

For normal conditions, k_{eff} must remain below 1.0 at 95/95 when flooded with unborated water, and k_{eff} must not exceed 0.95 at 95/95 when flooded with borated water. The applicant conservatively credited 800 milligrams per milliliter (mg/ml) (800 parts per million (ppm)) of boron for borated water analyses, compared to the normal boron concentration of at least 1800 mg/ml (1800 ppm) given in DCA Part 2, Tier 2, Table 9.1.3-2, "Water Chemistry Parameters Monitored for the Ultimate Heat Sink Pools," and the design-basis refueling boron concentration of 2000 mg/ml (2000 ppm) listed in DCA Part 2, Tier 2, Table 4.3-2, "Nuclear Design Parameters (for Equilibrium Cycle)."

As shown in Table 3-26 of TR-0816-49833, the $k_{95/95}$ for nominal conditions without damaged fuel is 0.96623, assuming unborated moderator and 0.92191 assuming borated moderator. The applicant also analyzed three different configurations of five damaged fuel assemblies in the fuel storage racks. The applicant conservatively modeled damaged fuel as fresh fuel enriched to 5 weight percent with the fuel-clad gap in all fuel rods completely flooded with water, simulating cladding failure. The limiting case was five damaged assemblies in the center of a storage rack, resulting in a $k_{95/95}$ of 0.96642 assuming unborated moderator and 0.92220 assuming borated moderator. In addition, the applicant analyzed a more conservative case assuming all damaged fuel assemblies, which resulted in a $k_{95/95}$ of 0.97429 assuming unborated moderator and 0.92857 assuming borated moderator. The staff does not consider all damaged fuel assemblies to be a normal condition as the NuScale design specifies storage of only five damaged fuel assemblies; however, this scenario bounds multiple damaged fuel storage configurations.

In addition, the staff conducted confirmatory analyses of the normal-conditions model using Version 6.1 of the Monte Carlo N-particle (MCNP) code, developed by Los Alamos National Laboratory, to assess the validity of the applicant's results. The staff built its model independently using the design information, such as dimensions and materials specifications, provided in the DCA. Like the applicant, the staff also omitted some rack structural materials from the model. The staff's calculated k_{eff} values for the unborated and borated normal-conditions model are higher than the applicant's without adjusting for any bias or uncertainty. This likely results from the staff's use of the limiting water temperature and density and different cross-section treatment; MCNP uses continuous-energy cross sections, but the applicant used multigroup cross sections. However, the applicant's results plus the moderator temperature bias adjustment are more conservative (i.e., result in a higher k_{eff}) than the staff's calculations. Because the applicant's calculated k_{eff} is higher than the staff's when factoring in the moderator temperature bias, the staff has reasonable assurance that the applicant's calculation results are acceptable.

The applicant did not provide a criticality analysis for normal conditions of fuel handling in the fuel storage facility. However, DCA Part 2, Tier 2, Section 9.1.1, states that each piece of FHE, including the new fuel jib crane (NFJC), the new fuel elevator, and the FHM, may move only one fuel assembly at a time. In addition, DCA Part 2, Tier 2, Section 9.1.4, "Fuel Handling Equipment," states that the FHE is designed in accordance with ANSI/ANS 57.1 and describes characteristics of the equipment that preclude system malfunctions or failures that could cause criticality accidents. Such features include an FHM control system that monitors and verifies each fuel movement and position according to the fuel movement plan. In addition, the applicant has included COL Information Item 9.1-1, which requires a COL applicant to develop plant programs and procedures for safe operations during fuel storage and handling, including criticality control. The staff also notes that the conservative assumption of fuel assemblies occupying even inaccessible rack locations bounds normal conditions of fuel handling. Given that and the controls on fuel handling, the staff has reasonable assurance that criticality safety will be maintained during normal fuel handling operations.

Based on the results presented in TR-0816-49833 and the staff's confirmatory analysis, the staff has reasonable assurance that the NuScale design meets 10 CFR 50.68 during normal conditions.

9.1.1.4.8 Abnormal Conditions

The goal of the abnormal conditions analysis is to demonstrate that k_{eff} will remain within the specified limits during abnormal or accident conditions. Such conditions include a fuel assembly

dropped on top of a rack, a fuel assembly misloaded into an improper position in the rack, a fuel assembly dropped outside of a rack in the fuel elevator area, a seismic event, and a boron dilution event.

The structural analysis of the fuel racks, including vertical fuel assembly drop scenarios, is to be provided by a COL applicant as part of COL Item 9.1-8. Therefore, a COL applicant will be responsible for assessing any impact of resulting storage rack deformation and relocation on the criticality analysis. The applicant did not provide a quantitative criticality analysis for a fuel assembly dropped horizontally on top of a rack because the distance between the dropped assembly and the active fuel region of the assemblies in the rack precludes neutronic coupling. Based on the number of neutrons mean free paths corresponding to this distance, the staff concludes that this accident is not limiting and accepts the qualitative criticality analysis.

Because the fuel storage racks have no loading restrictions, a misloaded fuel assembly is not a credible accident and requires no analysis.

The applicant used the whole-pool model discussed in Section 9.1.1.4.7 of this SER to analyze the fuel assembly dropped outside the rack. The applicant analyzed several locations for the fuel assembly dropped outside of the rack in the new fuel elevator area. The most limiting location resulted in a $k_{95/95}$ of 0.92219, which is less than the limit of 0.95 when crediting soluble boron and represents a minimal increase over the nominal condition. Furthermore, in accordance with SRP Section 9.1.1, the spaces between racks are not large enough to accommodate a fuel assembly, and the travel limitations of the FHM prevent placement of a fuel assembly between the racks and the SFP wall. Therefore, dropping a fuel assembly between racks or between a rack and a wall is physically precluded, and therefore, no analysis is necessary for this scenario.

The applicant used the single-rack model discussed in Section 9.1.1.4.6 of this SER for the seismic event analysis, changing the distance between the outside of the rack and the model's periodic boundaries to simulate the racks moving closer together. The applicant calculated the reduction in spacing that would need to occur to cause $k_{95/95}$ to reach 0.95. The structural analysis of the fuel racks, including seismic response, is to be provided by a COL applicant as part of COL Item 9.1-8; therefore, a COL applicant will be responsible for demonstrating that the limits in 10 CFR 50.68 are met in the event of a safe-shutdown earthquake (SSE).

DCA Part 2, Tier 2, Section 9.1.1, states that the large volume of water (about 26,500 cubic meters (m^3) (7 million gallons (gal)) in the UHS prevents an undetected boron dilution accident from the normal boron concentration of about 1800 mg/ml (1800 ppm) to below the 800 mg/ml (800 ppm) credited in the analysis, because approximately 26,500 m^3 (7 million gal) of unborated water would need to be added to the UHS. Because the pool level alarm would warn operators when the UHS water level rises by 0.3 m (1 ft) (about 352 m^3 (93,000 gal)), operators would detect or notice a potential UHS boron dilution well before the boron concentration would drop to 800 mg/ml (800 ppm).

The staff finds the scope of the abnormal conditions identified by the applicant comprehensive. Based on the results presented in TR-0816-49833, the staff has reasonable assurance that the NuScale design meets the requirements of 10 CFR 50.68 during abnormal conditions, except for structural evaluation aspects to be provided by a COL applicant.

9.1.1.5 Technical Specifications

GTS 3.5.3 enforces limits on UHS parameters, including a minimum level, a maximum bulk average temperature, and a bulk average boron concentration within the limits specified in the core operating limits report (COLR). Of these limits, the boron concentration has direct implications for criticality safety. Surveillance Requirement (SR) 3.5.3.3 calls for verification of the concentration in accordance with the surveillance frequency control program. According to the bases for GTS 3.5.3, the minimum boron concentration limit in the COLR is based on maintaining adequate shutdown margin during module refueling. Technical Specification (TS) 5.6.3, "Core Operating Limits Report," states that DCA Part 2, Tier 2, Section 4.3, "Nuclear Design," contains the methodology used to determine UHS limits in the COLR. According to DCA Part 2, Tier 2, Table 4.3-2, the design-basis refueling boron concentration is 2000 mg/ml (2000 ppm), much greater than the 800 mg/ml (800 ppm) credited in the criticality analysis. Therefore, GTS 3.5.3 and SR 3.5.3.3 provide reasonable assurance that the boron concentration will remain at a value above that credited in the criticality analysis.

GTS 4.3 places limitations on fuel storage, including the maximum enrichment of fuel assemblies, the maximum allowable k_{eff}, the center-to-center spacing between fuel assemblies in the fuel storage racks, and the capacity of the SFP. The staff notes that the information in GTS 4.3 is consistent with the information in the criticality analysis.

GTS 5.5.12 sets forth requirements for the neutron absorber monitoring program. As discussed in Section 9.1.1.4.3 of this SER, the staff finds GTS 5.5.12 acceptable.

For the reasons stated above, the staff finds the TS related to the criticality safety of new and spent fuel storage and handling acceptable. Chapter 16 of this SER contains additional evaluation of the TS.

9.1.1.6 Conditions and Limitations

The staff based its review and approval of the NuScale fuel storage rack criticality analysis on the fuel designs, rack designs, rack configurations, and models presented in TR-0816-49833, Revision 1. Any changes to the analysis inputs and assumptions, such as neutron absorber material or storage rack spacing, would be considered changes to the approved methodology and would require additional analysis, review, and approval.

9.1.1.7 Combined License Information Items

Table 9.1.1-1 below provides the COL information items related to criticality safety of new and spent fuel storage and handling and their descriptions. The staff reviewed these proposed COL items and finds them acceptable. 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. The applicant did not provide information on the structural qualification of the rack and its potential for interaction with other commodities during operation. The staff has reviewed COL Item 9.1-8 and finds that it provides adequate information regarding overall issues certified, while the COL applicant evaluates the structural integrity of the rack and other potential interactions. The structural analysis described by COL Item 9.1-8 is also needed to ensure that mechanical accidents and seismic events will not affect the conclusions the applicant has reached regarding criticality safety. Finally, COL Item 9.1-9 is needed to demonstrate that the specified properties of the selected neutron absorber material and the as-manufactured neutron absorber products are met, as described in more detail in SER

Section 9.1.1.4.3.3.1. The staff concludes that no other COL information items are needed for this review section.

Table 9.1.1-1 NuScale COL Information Items for Section 9.1.1

COL Item No.	Description	DCA Part 2, Tier 2, Section
9.1-1	A COL applicant that references the NuScale Power Plant design certification will develop plant programs and procedures for safe operations during handling and storage of new and spent fuel assemblies, including criticality control.	9.1.1
9.1-8	A COL applicant that references the NuScale Power Plant design certification will provide a structural evaluation of the spent fuel storage racks, and fuel assemblies located in the racks, and confirm the thermal-hydraulic, criticality, and material analysis aspects of the design remain valid. This evaluation is dependent on the vendor-specific spent fuel storage rack design.	9.1.2
9.1-9	A COL applicant that references the NuScale Power Plant design certification will provide a neutron absorber material qualification report, which demonstrates that the neutron absorber material can meet the neutron attenuation and environmental compatibility design functions described in Technical Report TR-0816-49833. The COL applicant will establish procedures to evaluate the neutron attenuation uncertainty associated with the material lot variability and will establish procedures to inspect the as-manufactured material for contamination and manufacturing defects.	9.1.2

9.1.1.8 Conclusion

The staff reviewed the fuel storage rack criticality analysis, storage rack materials, and information on criticality during fuel handling described in DCA Part 2, Tier 2, Section 9.1.1, Revision 3, and TR-0816-49833, Revision 1, to evaluate whether new and spent fuel will remain subcritical in all credible storage conditions and during fuel handling. Based on the review of the applicant's proposed design criteria and design bases for the fuel storage facilities and the provisions necessary to maintain a subcritical array, the staff concludes that the design of the fuel storage facilities and supporting systems is in conformance with the Commission's regulations in GDC 62 and in 10 CFR 50.68. This conclusion is based on the following:

The applicant has met the requirements of GDC 62 pertaining to criticality because the fuel will remain subcritical under all normal and credible abnormal conditions analyzed in the scope of the DCA. Structural analysis aspects are to be evaluated by a COL applicant.

The applicant has met the requirements of 10 CFR 50.68 by performing criticality analyses for normal and abnormal conditions and demonstrating that the design maintains the calculated k_{eff} values within the limits specified in 10 CFR 50.68 at a 95-percent probability and 95-percent confidence level.

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 has 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. The fuel racks ensure that stored fuel is maintained in a suitable geometry to prevent criticality and provide cooling for all design conditions.

The applicant described the SFP structures, systems, and components (SSCs) related to fuel storage in DCA Part 2, Tier 2, Section 9.1.2. The staff evaluates the spent fuel pool cooling system (SFPCS), fuel handling system, and ventilation separately.

9.1.2.2 Summary of Application

DCA Part 2, Tier 1: The applicant provided a general description of the new and spent fuel storage systems in DCA Part 2, Tier 1, Section 3.5.

DCA Part 2, Tier 2: The applicant described new and spent fuel storage in DCA Part 2, Tier 2, Section 9.1.2. This facility provides for the storage of new and spent fuel assemblies. The system functions are to maintain the fuel assemblies in a safe and subcritical array during all storage conditions.

ITAAC: DCA Part 2, Tier 1, Table 3.5-1, provides ITAAC information for fuel storage. These ITAAC are evaluated in Section 14.3 of this SER.

Initial Test Program: No initial tests are proposed for the SFP.

Technical Specifications: No GTS are explicitly evaluated in this section. GTS addressing criticality in the SFP are evaluated in Section 9.1.1 of this SER. The SFP is part of the UHS and shares the same volume of water. The staff evaluates the UHS GTS in Section 9.2.5 of this SER.

Technical Reports: TR-0816-49833, Revision 1, "Fuel Storage Rack Analysis," issued November 2018 (ADAMS Accession No. ML18310A154).

9.1.2.3 Regulatory Basis

Design Specific Review Standard (DSRS) for NuScale SMR 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, as it relates to 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). These SSCs shall be appropriately protected 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 units 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 monitoring systems for detecting conditions that could cause 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 DCA Part 2, Tier 2, Section 9.1.2, against the agency’s regulatory guidance to ensure that the DCA represents the complete scope of information relating to this review topic.

The SFP contains 14 fuel storage racks with 121 locations each, for a total of 1,694 storage locations. Because of the FHM’s travel limitations (evaluated in Section 9.1.4 of this SER), only 1,404 fuel storage locations are accessible to the FHM. However, due to possible difficulty in reaching the 11 fuel storage locations closest to the weir wall, the applicant considered only 1,393 fuel storage locations to be accessible. Therefore, the applicant’s evaluation for minimum storage capacity credits only 1,393 locations as available. The 1,393 fuel storage locations include storage for five damaged fuel assemblies and for nonfuel core components such as a control rod assembly (stored within a fuel assembly). Each refueling batch consists of 13 assemblies. In DCA Part 2, Tier 2, Section 9.1.3.3.4, “Residual Heat Removal Capability,” the applicant stated that, with a 24-month cycle (one refueling every 2 months), the SFP has capacity for more than 10 years of operation.

Based on a 24-month cycle, the discharge batch of 13 spent fuel assemblies, a core size of 37 fuel assemblies, and the total number of storage locations available, the staff finds that the applicant’s fuel storage facility will have sufficient space to accommodate the fuel discharged from more than 5 years of operation and a full core offload. This storage capacity meets the recommendation in SRP Section 9.1.2(iii)(1), which indicates that the minimum SFP storage capacity should equal or exceed the amount of spent fuel from 5 years of operation at full power plus one full-core discharge.

9.1.2.4.1 GDC 2, “Design Bases for Protection against Natural Phenomena”

DCA Part 2, Tier 2, Section 9.1.2, states that the SFP is located within the RXB, which is a seismic Category I structure designed to provide protection from the effects of natural phenomena, including earthquakes, tornadoes, hurricanes, floods, and external missiles. The fuel storage racks are designed as seismic Category I structures, in order to maintain K_{eff} below limits during and following an SSE. The racks are freestanding structures that are designed to slide during an SSE, without making contact with the pool walls.

The SFP and the UHS concrete structure and its liners are designed to meet seismic Category I requirements; therefore, they meet 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 necessary for radiation shielding should be designed to seismic Category I requirements), and RG 1.29, Revision 5, “Seismic Design Classification,” issued July 2016, which states that all SSCs that must remain functional following a design-basis seismic event should be designed to seismic Category I criteria).

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.

DCA Part 2, Tier 2, Section 3.7.3, “Seismic Subsystem Analysis,” states that any nonseismic Category I SSC whose failure could adversely affect seismic Category I SSCs is designed to meet seismic Category II requirements. DCA Part 2, Tier 2, Section 9.1.2.3, “Safety Evaluation,” states that the FHE is designed to seismic Category II requirements, except the FHM, which is seismic Category I. The reactor building crane (RBC) is designed to seismic Category I requirements.

The staff finds that by designing the equipment located in the vicinity of the SFP to seismic Category I or II requirements, the applicant followed the guidance in DSRS Section 9.1.2.III.4.C. Based on the above discussion, 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 anticipated normal operating and postulated accident conditions. This requirement 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.

DCA Part 2, Tier 2, Section 9.1.2.1, “Design Bases,” states that the fuel storage racks, SFP, liner, and RXB protect the stored fuel assemblies from the effects of natural phenomena hazards, including earthquakes, hurricanes, tornadoes, floods, tsunami, seiches, external missiles, internal missiles, pipe whip, and discharging fluids.

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 the applicant's system description in the DCA and found that the stored fuel assemblies are located below grade and protected by several seismic Category I structures (the fuel storage racks, SFP, liner, and RXB). This protection ensures that the stored fuel is adequately protected against natural phenomena hazards.

DCA Part 2, Tier 2, 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 evaluation of the missile protection methodology (including acceptability of the barriers) is discussed in Section 3.5.1 of this SER.

Following the staff acceptance of the applicant's missile protection methodology as discussed in Section 3.5.1 of this SER, 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 and 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 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 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.

In DCA Part 2, Tier 2, Section 9.1.2.3, the applicant 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.

DCA Part 2, Tier 2, 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 UHS and the NPM by a weir. Based on the separation between the SFP and 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 SER, 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, the staff finds that the design meet 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 for 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 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.

DSRS Section 9.1.2.III.4.B states that the new and spent fuel storage racks are designed such that a fuel assembly can be inserted only in a designed location. The design also should prevent placement of fuel assemblies in the adjacent regions external to the racks. DCA Part 2, Tier 2, Section 9.1.2, describes the SFP configuration that prevents placing an assembly between the storage racks and between the storage racks and the wall. The FHM has travel limitations that also prevent the movement of fuel assemblies outside the storage racks.

The staff evaluated the SFP layout described in the DCA and the travel limitations of the FHM and determined that the proposed design meets the design recommendations of DSRS Section 9.1.2. Section 9.1.4 of this SER presents additional discussion of the FHM and the movement limits.

DSRS Section 9.1.2 recommends the use of low-density storage, at a minimum, for the most recently discharged fuel to enhance the capability for cooling. If low-density storage is not used, the use of high-density storage racks need to be evaluated by the staff on a case-by-case basis.

The DCA does not specify the design rack density (low or high); however, the rack design described in the DCA conforms to that of high-density racks. The applicant's DCA provides criticality and thermal evaluations of the racks to demonstrate design adequacy. Section 9.1.1 of this SER discusses the staff's evaluation of the criticality analysis for the SFP, and Section 9.1.3 of this SER gives the staff's evaluation of the applicant's thermal analysis report, which demonstrates adequate cooling in the SFP. Therefore, the staff finds the use of high-density racks for storage of new and spent fuel, including the most recently discharged fuel, acceptable.

DSRS Section 9.1.2.III.4.G states that all the wetted surfaces in the SFP should be of chemically compatible and stable materials. DCA Part 2, Tier 2, Section 9.1.2.3.2, describes the materials used in the fabrication of the racks. The fuel racks are constructed from three materials: [[ ]], an aluminum and boron-carbide neutron-absorbing material, [[ ]], and [[ ]]. The neutron-absorbing material is not credited with a structural capacity.

The staff reviewed the selection of materials in the fuel storage system to determine if the applicant had selected materials that were compatible with and stable in the UHS environment. Control of the UHS water chemistry and selection of proper materials and fabrication controls demonstrate reasonable assurance that the materials are compatible and stable.

The UHS water chemistry, described in Table 9.1.3-2, "Water Chemistry Parameters Monitored for the Ultimate Heat Sink Pools," is identical to the recommended values in Electric Power Research Institute (EPRI) Report 1014986, "Pressurized Water Reactor Primary Water Chemistry Guidelines," Appendix B, Section B.7, "Spent Fuel Pool Cooling and Cleanup System." Appendix B to the EPRI Pressurized Water Chemistry Guidelines is a "recommended" element and, as such, is industry best practice. The applicant's commitment to control the UHS water chemistry consistent with the EPRI recommended practices, which has been shown to be sufficient to prevent corrosion in SFPs, is a reasonable basis for the prevention of corrosion in the NuScale UHS.

The applicant specified that the fuel storage system will meet the requirements of ASME Code Section III-NF. ASME Code Section III-NF provides quality assurance requirements that ensure that the procured material has sufficient documentation and that the critical characteristics of the base material are verified. In the technical report, the applicant stated that "the principal structural load-carrying material used in the design is ASTM…stainless steel." The use of ASTM grade material for ASME Code Section III-NF construction is permitted by ASME Code Section III, subparagraph NCA-1221.1, "Metallic Material, ASTM Specification." Because the applicant committed to ASME Code Section III-NF, the staff finds that the quality assurance of the materials is sufficient for the important-to-safety function of the fuel racks.

The applicant has selected common grades of [[ ]], and [[ ]]), which have extensive history in LWRs. The [[ ]] material is used for the support leg bolts and is a generic "type 300" grade. The [[ ]] material is utilized for the threaded fastener on the support leg of the rack. The fuel racks are designed to slide in the event of an earthquake and as such are always in compression. The common degradation mechanisms for "type 300" stainless steels (such as stress-corrosion cracking) occur in tension or when the material is exposed to sensitizing temperatures. Considering that the material will not be welded during fabrication and that there is no expected degradation mechanism other than generalized corrosion, the staff finds that the [[ ]] material as described in the technical report is sufficient for these threaded fasteners.

[[ ]] is utilized for the fuel assembly tube, the support bars, and the baseplate. [[ ]] has significant operating experience in nuclear reactors and is acceptable for service, provided that sensitization is prevented, and weldments provide sufficient delta ferrite. [[ ]] conforms to NRC guidance found in RG 1.44, "Control of the Processing and Use of Stainless Steel," and is sufficient to prevent sensitization. The applicant committed to meeting NF-2433, "Delta Ferrite Determination," as part of the commitment to meeting ASME Code Section III-NF. The delta ferrite requirements in the ASME Code, Section III-NF, are consistent with RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and are therefore acceptable.

The [[ ]] material is utilized for the support legs. [[ ]], and thus, the guidance for austenitic stainless steel is not applicable. This [[ ]] material has significant operating experience in nuclear reactors, including experience in this application, and is acceptable for service.

Corrosion-resistant materials may degrade in service if the material is not controlled adequately during fabrication. The applicant stated that the fuel storage system will meet ASME NQA-1, which includes Subpart 2.1, which provides contamination controls. ASME NQA-1, Subpart 2.1, contains cleanness controls sufficient to protect the fuel rack materials from coming in contact with harmful chemicals and requires that mechanical cleaning (such as grinding) cannot introduce surface contamination.

The discussion of materials compatibility for the neutron absorber is provided in SER Section 9.1.1. The staff concludes that the UHS water chemistry, along with the use of [[ ]] stainless steel, is sufficient to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents as required by GDC 4.

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 of the gates should be above the top of the fuel assemblies, and the volume of the adjacent fuel handling areas should be limited so that leakage into these areas while drained would not reduce the coolant inventory to less than 3 m (10 ft) above the top of the fuel assemblies.

In DCA Part 2, Tier 2, Section 9.2.5, the applicant described the spent fuel handling area, in conjunction with the reactor pool and the 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 leaktight after an SSE. The bottom of the weir that leads from the SFP to the RFP is 3 m (10 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 gates. In DCA Part 2, Tier 2, Section 9.2.5, the applicant stated that a failure of the dry dock gate while the dry dock is empty would cause the UHS (and the SFP) water level to drop by 3.7 m (12 ft), but the SFP water level would remain more than 3 m (10 ft) above the top of the fuel assemblies.

The staff evaluated the applicant's description of the failure of the dry dock gate and finds that the total volume of the dry dock area meets the recommendations of DSRS Section 9.1.2.III.4.H.i because the SFP water level would remain more than 3 m (10 ft) above the top of the stored fuel assemblies.

DCA Part 2, Tier 2, Section 9.1.2, states that piping penetrations to the SFP are at least 3 m (10 ft) above the top of the fuel assemblies seated in the spent fuel storage racks, which complies with the recommendation in DSRS Section 9.1.2.III.4.H.ii.

DCA Part 2, Tier 2, Section 9.1.2.3.2, states that the fuel storage racks have adequate natural circulation cooling flow into the baseplate holes and through the fuel assemblies. The racks provide sufficient decay heat removal to maintain the peak temperature for a fuel rod below the local saturation temperature and prevent nucleate boiling. The applicant submitted TR-0816-49833, Revision 1, which includes the thermal analysis report that demonstrates adequate natural circulation cooling flow through the storage racks.

The staff evaluated the referenced technical report and found that the report demonstrates that the storage racks have adequate flow area through the baseplate to allow for the natural circulation of cooling water through the stored fuel and prevent nucleate boiling.

Based on the design features reviewed above, the staff finds that the 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 FHM refueling machines 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.

The applicant discussed the SFP level and temperature instrumentation in DCA Part 2, Tier 2, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," and the staff evaluates it in Section 9.1.3 of this SER. The applicant described normal and accident operation of the SFP area ventilation system in DCA Part 2, Tier 2, 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 SER.

DSRS Section 9.1.2.III.4.D states that the spent fuel racks need to be designed to withstand the maximum FHE uplift forces without an increase in k_{eff} or damage to the watertight integrity of the SFP liner. DCA Part 2, Tier 2, Section 9.1.2.2, "Facilities Description," states that the storage fuel racks are designed to withstand the FHM uplift force.

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.

DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 9.1.2.3.6, "Monitoring," states that the RXB (including the SFP area) has radiation monitors to detect both general area radiation levels and airborne contamination levels. DCA Part 2, Tier 2, Section 12.3, "Radiation Protection Design Features," provides additional information on the radiation area monitors. The staff evaluation of the applicant's radiation protection design features is evaluated in Section 12.3 of this SER.

Based on the design features reviewed above, the staff finds that the NuScale design meets GDC 63 and provides assurance that a loss of residual heat removal capability and high

radiation levels will be detected and that the release of radioactive materials to the environment will 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 and storage racks) avoid unnecessary buildup of radioactive material. Appropriate shielding of spent fuel and new recycled fuels also ensures compliance with the ALARA principle.

DCA Part 2, Tier 2, Section 9.1.2.3.7, "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. The PLDS is discussed and evaluated in Section 9.1.3 of this SER. The surface finishes of the components for the fuel storage racks and SFP liner are smooth to minimize the accumulation of radioactive materials and to facilitate surface decontamination. The baseplate of the fuel storage racks is elevated in order to provide space to allow cleaning under the baseplate for the removal of buildup and debris.

Section 12.1 of this SER contains the staff's complete evaluation of the ALARA design and decontamination details. The staff finds that the features discussed above meet the recommendation of SRP Section 9.1.2, "New and Spent Fuel Storage," and therefore are in compliance 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

The applicant has not identified any GTS evaluated in this section of the report. The staff reviewed the DCA and concluded that no TS are required for the SFP. GTS related to criticality of the SFP are evaluated in Section 9.1.1 of this SER, and GTS related to the SFP water level and temperature are evaluated in Section 9.2.5 of this SER.

9.1.2.7 Combined License Information Items

DCA Part 2, Tier 2, Table 1.8-2, describes COL Item 9.1-2, which directs a COL applicant that references the NuScale Power Plant design certification to demonstrate that an NRC-licensed cask can be lowered into the dry dock and used to remove spent fuel assemblies from the plant.

The staff evaluated the proposed COL information item and determined that the demonstration that an NRC-licensed cask exists and can be used in the NuScale plant is adequate to ensure there are means to offload fuel from the SFP.

Table 9.1.2-1 NuScale COL Information Items for Section 9.1.2

COL Item No.	Description	DCA Part 2, Tier 2, Section
9.1-2	A COL applicant that references the NuScale Power Plant design certification will demonstrate that an NRC-licensed cask can be lowered into the dry dock and used to remove spent fuel assemblies from the plant.	9.1.2

9.1.2.8 Conclusion

The staff evaluated the SFPCS for the NuScale design in accordance with the guidance of SRP Section 9.1.2. 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."

9.1.3 Spent Fuel 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.

DCA Part 2, Tier 2, Section 9.1.3, discusses the design and performance of the pool support systems. Each of these systems performs a different function. The active SFPCS removes decay heat from the stored fuel during normal operation. The active reactor pool cooling system (RPCS) cools the NPMs located in the reactor pool and RFP. The pool cleanup system (PCUS) removes impurities to reduce radiation dose rates and to maintain water chemistry and clarity in the UHS pools and dry dock. The pool surge control system (PSCS) 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 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 SER.

9.1.3.2 Summary of Application

DCA Part 2, Tier 1: The applicant has not proposed any Tier 1 sections related to the SFPCS.

DCA Part 2, Tier 2: The applicant described the system in DCA Part 2, Tier 2, Section 9.1.3. The SFP and the UHS are connected and share the volume of water above the weir. The SFPCS and the RPCS share the cooling function during normal conditions and are not safety-related systems. DCA Part 2, Tier 2, Section 9.2.5, discusses the safety-related function of cooling the stored fuel during and after an accident scenario. The PCUS removes impurities from the UHS pools and the dry dock and is not a safety-related system.

ITAAC: The applicant has not proposed any ITAAC related to the SFPCS.

Initial Test Program: Inspection and testing of the SFPCS are performed before plant operation, as described in DCA Part 2, Tier 2, Table 14.2-4, "Pool Surge Control System Test #4." The ITP is evaluated in Section 14.2 of this SER.

Technical Specifications: DCA 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.

Technical Reports: The applicant did not reference any technical report related to the SFPCS.

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 shared SSCs important to safety being capable of performing required 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 detect conditions that could result in the loss of decay heat removal capability, to detect excessive radiation levels, and to initiate appropriate safety actions.

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

9.1.3.4 Technical Evaluation

The SFPCS consists of two 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 active RPCS cools the NPMs located in the reactor pool and RFP. The RPCS consists of two suction headers, each with a strainer, that supply three cooling trains, each with a pump and a heat exchanger.

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 expected 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.1.i, 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.1.i, 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 SFP cooling and purification 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 DCA Part 2, Tier 2, 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 SFPCS and the RPCS are not safety-related systems. 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 SER.

DCA Part 2, Tier 2, Section 9.2.5, identifies the minimum safety water level needed to provide safety-related cooling to the NPMs as 16.8 m (55 ft) from the bottom of the pool

(elevation 24.4 m (80 ft)). The staff confirmed that elevations of all pipe openings or antisiphon devices on the piping are above the 16.8 m (55 ft) pool water level. DCA Part 2, Tier 2, Section 9.1.3.2.1, "Spent Fuel Pool Cooling System," states that the elevation of the SFPCS piping penetration through an SFP or RFP wall, and the open ends of the suction and discharge piping in the SFP and RFP, are above the 16.8 m (55 ft) pool water level. DCA Part 2, Tier 2, Section 9.1.3.2.2, states that the bottom of each RPCS piping penetration through a wall of the RFP, reactor pool, or SFP, and the open ends of the suction and discharge piping in the pools, are above the 16.8 m (55 ft) pool water level. PSCS piping penetrates the dry dock wall (UHS) above the 16.8 m (55 ft) pool water level. The piping continues deeper into the pool, but it is equipped with antisiphoning devices, which are located above the 16.8 m (55 ft) pool water level. The PCUS receives flow from the SFPCS, the RPCS, and the PSCS; therefore, the PCUS has no penetrations into the UHS/SFP.

DCA Part 2, Tier 2, Table 9.2.5-1, "Relevant Ultimate Heat Sink Parameters," states that the actual penetration height for SFPCS and RPCS suction piping in the SFP and RFP is 18.3 m (60 ft) above the pool floor. DCA Part 2, Tier 2, Section 9.1.3, states that the SFP cooling and purification subsystem piping, which has the potential to interact with seismic Category I SSCs, is 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 provided in DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 9.1.3.1, states that the SFPCS, RPCS, and PCUS are located inside the seismic Category I RXB. The PSCS 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.

DCA Part 2, Tier 2, Section 3.7.3, states that nonseismic Category I SSCs that could adversely affect seismic Category I SSCs are categorized as seismic Category II.

The DCA states that the FHE meets seismic Category II requirements, except the FHM, which is designed to seismic Category I requirements. The staff evaluates the FHE and its operation in Section 9.1.4 of this SER. The RXB crane is prevented from traveling above the SFP area,

thereby preventing heavy loads from traveling above the stored fuel. The staff evaluates the overhead heavy load handling system (OHLHS) in Section 9.1.5 of this SER.

The staff finds that failure of the SFP cooling and purification system components that are not safety related would not adversely impact safety-related SSCs. The design of the FHE and the OHLHS prevents the drop of heavy loads in the SFP area. Therefore, the staff finds that the applicant'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 unit.

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 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.

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 DCA Part 2, Tier 2, Section 9.1.2.3, the applicant 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.

DCA Part 2, Tier 2, 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 will 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 SER. 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. 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 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

DCA Part 2, Tier 2, Section 9.1.3, states that the pool support systems are designed with isolation valves to facilitate the inservice inspection of trains or major components. The applicant stated that laydown areas are provided for the maintenance of pumps and heat exchangers. Space is sufficient to allow for the removal of heat exchanger tube bundles and head. DCA Part 2, Tier 2, Section 14.2, "Initial Plant Test Program," discusses initial plant testing for the pool support systems. These tests ensure that the support systems are capable of performing their design functions. The staff finds that the design features discussed in DCA Part 2, Tier 2, 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.

DCA Part 2, Tier 2, Section 9.1.3, discusses the SFPCS, which consists of two trains, each with an inlet strainer, a pump, and a heat exchanger. The SFPCS is not a safety-related system, and during normal plant operation, it runs continuously using either one or two trains. The suction and discharge lines are on opposite corners of the SFP. The SFPCS can be aligned to connect with the RPCS. During normal operation, these two systems can operate in conjunction to cool the stored fuel in the SFP and the NPM in the reactor pool. The RPCS intakes are in the north and south walls of the RFP, and they discharge into each of the NPM bays in the reactor pool.

DCA Part 2, Tier 2, Section 9.1.3.3.4, states that the pool cooling systems (SFPCS and RPCS) are designed to maintain the pool bulk temperature below 43.3 °C (110 °F). The applicant described the pool cooling capability under several scenarios. These scenarios include normal operation (12 NPMs in operation and a full SFP), refueling (11 NPMs in operation, and a full SFP including one full core offload), and abnormal scenarios (0.3 percent of the total plant thermal output). For each of these scenarios, the applicant stated that the pool cooling systems are capable of maintaining the pool water temperature at or below the normal temperature (37.8 °C (100 °F)), with at least four of the five trains in operation between the two cooling systems.

DCA Part 2, Tier 2, Table 9.1.3-1a, "Equipment Parameters for the Spent Fuel Pool Cooling System," and DCA Part 2, Tier 2, Table 9.1.3-1b, "Equipment Parameters for the Reactor Pool Cooling System," describe the capacity of the cooling systems and specify that each of the five heat exchangers has a capacity to remove 5.47×10^6 kilojoules per hour (kJ/h) (5.18 million British thermal units per hour (MMBtu/h)), which results in a total heat removal capacity of 2.73×10^7 kJ/h (25.9 MMBtu/h), or 2.18×10^7 kJ/h (20.7 MMBtu/h) assuming a single train out of operation.

DSRS Section 9.1.3.III.3.D recommends that the cooling system 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 presented in the DCA and the system design capability and found that the pool cooling systems have 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 are (1) 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."

In DCA Tier 2, Section 3.8.4 and Section 9.2.5.2, the applicant stated that to prevent overpressurization in the UHS area of the RXB during abnormal conditions, an overpressurization vent (OPV) is included in the RXB system. If power is available, the applicant 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 OPV opens and releases the RXB pressure and prevents overpressurization of the building.

The staff evaluated the applicant's description of the OPV passive system, which is described as a safety-related, seismic Category I system. The staff finds that the OPV system conforms to the guidance provided 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 DCA Part 2, Tier 2, Section 9.2.5, the applicant identified that the minimum safety water level for the UHS is at a pool water level of 16.8 m (55 ft). In DCA Part 2, Tier 2, Section 9.1.3.3.5, the applicant stated that the elevation of the bottom of each of the piping penetrations through the walls of the UHS pools and the dry dock is above the 16.8 m (55 ft) pool water level. Additionally, the elevations of the open ends of the piping in the pools or the antisiphon devices on the piping are above this elevation.

The staff evaluation of the accident scenarios that determined the 16.8 m (55 ft) water level as the minimum safety limit is presented in Section 9.2.5 of this SER. 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. Therefore, the staff finds that these features meet the guidance of DSRS Section 9.1.3.

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 DCA Part 2, Tier 2, 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 SFPCS, with the liquid radioactive waste system (LRWS) providing alternate makeup. Both systems are capable of providing at least 380 liters per minute (L/min) (100 gallons per minute (gpm)). 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.

DCA Part 2, Tier 2, 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 15-centimeter (6-inch) diameter line is sloped and has the capacity to provide more than the credited 380 L/min (100 gpm) of water makeup.

The staff evaluated the design of the NuScale SFP, which is different from that of a typical large pressurized 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 applicant 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 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 SFPCS design is capable of providing suitable shielding for radiation protection and appropriate containment, confinement, and residual heat removal and has the capability to prevent a significant reduction in fuel storage coolant inventory under normal conditions. Therefore, the staff finds that the SFPCS conforms to the applicable requirements of GDC 61.

The Pool Cleanup System

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 PCUS, which is not a safety-related system. The PCUS maintains water chemistry and removes particulates from the UHS. Control of UHS water chemistry is an important-to-safety function. Improper control of water chemistry could result in spent fuel assemblies, the spent fuel rack, or NuScale containment modules becoming susceptible to corrosion. Corrosion could occur if pH is not maintained or if chlorides, fluorides, sulfates, and silica are not removed from the UHS. The PCUS also 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.

As described in 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 PCUS in DCA Part 2, Tier 2, Table 9.1.3-1c, "Equipment Parameters for the Pool Cleanup System."

The cleanup capacity of the PCUS is significantly greater than the minimum requirements needed to meet the water chemistry specified in Table 9.1.3-2, "Water Chemistry Parameters Monitored for the Ultimate Heat Sink Pools." A single PCUS operating at 50-percent capacity is sufficient to meet the design requirements for the UHS. Because the PCUS has three trains but requires only one train in operation, and because the NuScale design can use the chemical and volume control system (CVCS) to clean the UHS, if necessary, there is sufficient margin, redundancy, and independence to ensure that UHS water chemistry can be maintained. The capacity of a single train of the PCUS to process the entirety of the UHS within 4.2 days (or less time if multiple trains are used) provides sufficient assurance that the UHS can be purified within a short timeframe if adverse conditions are found. The design choice of three trains in the PCUS ensures that processing the resin beds or other operations that may make a single train of the PCUS inoperable would not impact normal or refueling operations.

The water chemistry parameters described in DCA Part 2, Tier 2, Table 9.1.3-2, define the requirements for the PCUS. 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 EPRI Guidelines. The applicant provided values for silica, suspended solids, and gamma isotopic activity, which are suggested parameters for monitoring in the EPRI Guidelines.

The staff concludes that the PCUS has sufficient capability and capacity to remove corrosion products, radioactive materials, and impurities from the UHS. The staff also finds that the PCUS has the capability to maintain the UHS water chemistry to prevent corrosion of the spent fuel assemblies, the SFP, and the NuScale modules. Given the limited safety significance of the PCUS, the staff finds that the design of the PCUS provides reasonable assurance that the PCUS will be able to perform the functions described in the DCA that are not safety related (but are important to safety) associated with the removal of impurities and maintaining SFP and RP water chemistry and clarity.

9.1.3.4.5 GDC 63, "Monitoring Fuel and Waste Storage"

Compliance with GDC 63 requires 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.

DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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 DCA 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 SER, 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 operating pressure of the SCWS is higher than the pressure of the pool cooling systems, preventing the loss of UHS water as a result of intersystem leakage.

The staff evaluated the system description in DCA Part 2, Tier 2, 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 PSCS storage tank is located outside the RXB in the plant yard. The catch basin around the tank has sufficient volume to store the pool surge control storage tank volume plus the contents of related piping. The catch basin is constructed of concrete and has a metal liner, which is sealed to prevent leakage to the environment. The pool surge control storage tank has a continuous air monitor with grab sample capabilities to monitor effluent releases from the tank.

The system description provided in DCA Part 2, Tier 2, Section 9.1.3, indicates that the PSCS lines between the RXB and the storage tank have a guard pipe from the catch basin to the RXB. The sump drainline is also within a guard pipe from the catch basin to the radioactive waste building (RWB). Each guard pipe provides collection and permits periodic surveillance for PSCS piping leaks.

The staff evaluated DCA Part 2, Tier 2, 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 SRP Section 9.1.3; therefore, the staff determined 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

Although applicants for design certifications are not required to submit plans for an ITP, RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," acknowledges that design certification applicants have previously submitted these plans to assist a future COL applicant referencing the design certification in meeting the requirements of 10 CFR 52.79(a)(28). DCA Part 2, Tier 2, Table 14.2-4, "Pool Surge Control System Test #4," lists preoperational test requirements for the PSCS. The test ensures the PSCS is capable of removing and adding water to the SFP/UHS by its connections to the dry dock.

The staff's evaluation of the ITP is presented in Section 14.2 of this SER.

9.1.3.6 Technical Specifications

No GTS requirements are directly associated with the SFPCS, RPCS, PCUS, PLDS, or PSCS. GTS 3.5.3 addresses the maximum UHS/SFP bulk temperature and minimum water level. Section 9.2.5 of this SER discusses this TS.

Based on a graded approach commensurate with the safety significance of the SSCs, the staff agrees that GTS are not required for the SFPCS, RPCS, PCUS, PLDS, and PSCS because these systems do not meet the criteria for assigning an LCO based on the criteria 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 item for the SFPCS, RPCS, PCUS, PLDS, and PSCS, and the staff's review of DCA Part 2, Tier 2, Section 9.1.3, did not identify a need for any additional COL item for this system.

9.1.3.8 Conclusion

The staff evaluated the SFPCS, RPCS, PCUS, PLDS, and PSCS for the NuScale design in accordance with the guidance of SRP Section 9.1.3. The staff finds that the design of these systems meets the requirements of GDC 2, 4, 5, 61, and 63 and 10 CFR 20.1101.

9.1.4 Light Load Handling System (Related to Refueling)

9.1.4.1 Introduction

The 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. The safety objective of the system is to avoid criticality accidents, releases of radioactivity as a result of damage to irradiated fuel, and unacceptable personnel radiation exposures.

9.1.4.2 Summary of Application

DCA Part 2, Tier 1: The applicant provided a general description of the FHE system in DCA Part 2, Tier 1, Section 3.4, "Fuel Handling Equipment System."

DCA Part 2, Tier 2: DCA Part 2, Section 9.1.4, "Fuel Handling Equipment," provides the design bases, description, and safety evaluation of the FHE. The major components of the FHE system are the FHM (DCA Part 2, Tier 2, Figure 9.1.4-2, "Fuel Handling Machine"), the NFJC (DCA Part 2, Tier 2, Figure 9.1.4-3, "Jib Crane"), and the new fuel elevator (DCA Part 2, Tier 2, Figures 9.1.4-4a and 9.1.4-4b). 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. DCA Part 2, Tier 2, Table 9.1.4-1, lists design information for the three major components. DCA Part 2, Tier 2, Section 9.1.4.2.2, "Major Component Description," describes each major component.

ITAAC: DCA Part 2, Tier 1, Table 3.4-1, "Fuel Handling Equipment System **I**nspections, Tests, Analyses, and Acceptance Criteria," specifies the ITAAC for the FHE. These ITAAC are evaluated in Section 14.3 of this SER.

Initial Test Program: DCA Part 2, Tier 2, Table 14.2-51, "Fuel Handling Equipment System Test #51," describes the performance testing for the FHE system. The ITP is evaluated in Section 14.2 of this SER.

Technical Specifications: No GTS are applicable to the FHE system.

Technical Reports: No technical reports are associated with the FHE system.

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 DCA Part 2, Tier 2, Section 9.1.4, against the agency's regulatory guidance to ensure that the DCA represents the complete scope of information relating to this review topic. The FHE system is used in the SFP, the RFP, and the new fuel staging area located in the RXB. DCA Part 2, Tier 2, Figure 9.1.4-1, "Refueling Floor Layout," provides the design layout of the FHE system.

The FHM is a traveling bridge and trolley crane that rides on rails set in the concrete on each side of the SFP and an adjacent portion of the RFP. The FHM 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. It is also used to transport new fuel assemblies between the new fuel elevator and their storage locations within the confines of the SFP. In addition, the FHM is used to move spent fuel assemblies from their storage locations to the spent fuel cask located in the RFP when they are ready for long-term dry cask storage on site or shipment off site. The RBC is used to handle the fully loaded spent fuel cask. DCA Part 2, Tier 2, Section 9.1.5, "Overhead Heavy Load Handling Systems," describes the design and operation of the RBC. Section 9.1.5 of this SER documents the associated staff evaluation.

The NFJC 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 new fuel elevator 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. The elevator includes a single track, an integral basket assembly/carrier, a hoist unit, and deflector sheaves. A new fuel assembly is loaded into the elevator by the NFJC at the operating floor level before being lowered to the bottom of the pool.

9.1.4.4.1 GDC 2, "Design Basis 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 RG 1.29, Regulatory Position C.1. This provision provides guidance on determining which SSCs shall be classified as seismic Category I. The guidance also recognizes the uniqueness of passive systems and states that certain passive SSCs may be named differently from SSCs that performed similar functions in previous designs. As noted in DCA Chapter 1, Table 1.9-2, the applicant designates the SSCs in RG 1.29 C.1.a through C.1.h as seismic Category I and SSCs that meet the criteria of C.1.i as seismic Category II. Because the SSCs that meet the criteria of C.1.i of RG 1.29 are not required to be functional after an SSE, the staff finds that this is acceptable.

The NuScale design classifies the FHE as a system that is not safety related or risk significant. The FHM is designed to meet seismic Category I requirements, and the NFJC and the new fuel elevator are 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 NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," issued May 1979, as supplemented with requirements of ASME NOG-1, "Rules for Construction of Overhead and Gantry Cranes (To Running Bridge, Multiple Girder)," for a Type 1 crane.

With regard to the seismic design of single-failure-proof cranes, NUREG-0554, Section 2.5, states, in part, the following:

> [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….

DCA Part 2, Tier 2, Section 9.1.4.2.2, states, in part, that "seismic restraints and restraining bars prevent the FHM bridge from overturning or coming off its rails during a seismic event." DCA Part 2, Tier 2, Figure 9.1.4-2, also shows these seismic restraints and presents an elevation view (east-west) of the FHM and a side view (north-south) of the FHM that, in part, show details of the seismic restraints for the trolley. The staff finds that the DCA figure provides sufficient information on the design features, as specified in NUREG-0554.

Based on the above evaluation, the staff finds that the FHE system design 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"

In DCA Part 2, Tier 2, Section 9.1.4.1, "Design Bases," the applicant stated, in part, the following:

> The FHE is interfaced with up to 12 NuScale Power Modules (NPMs) for initial fueling, fuel shuffling, and removal of spent fuel…. The design of the FHE allows for the performance of refueling activities on one module without affecting the operation of the other modules including potential shutdown and cooldown.

Further, in DCA Part 2, Tier 2, Section 21.2.3, the applicant reached a similar conclusion in its assessment of interactions between the operation of the FHE system during refueling of one NPM and the full-power operation of the remaining NPMs with respect to the fuel handling accident as analyzed in DCA Part 2, Tier 2, Section 15.7.4.

The staff evaluated the applicant's description of the FHE and finds that this equipment operates on a single NPM at a time and is located in a separate section of the UHS. The other NPM continues to operate independently. The FHE does not interact with the other operating NPM. The staff finds 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 for protection against releases of radioactivity to the environment as a result of fuel damage and 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 DCA Part 2, Tier 2, Section 9.1.4.2.3, "System Operation," except for the handling of new fuel assemblies using the NFJC in the new fuel staging area, all other fuel transfer and storage operations using the new fuel elevator 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 DCA Part 2, Tier 2, Section 9.1.4.2.3, radiation shielding is

provided by maintaining a minimum coverage of water (3 m (10 ft) deep) over irradiated fuel. 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. ANSI/ANS 57.1, Section 6.3.4.1.5, 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 assumed lift height for a fuel assembly handled by the FHM is approximately 9.4 m (31 ft) from the bottom of the SFP, and this height is incorporated as the upper limit for the FHM main hoist travel. Further, DCA Part 2, Tier 2, Section 9.1.3, indicates that the common water level for both the reactor pool and the SFP is normally maintained at elevation 28.7 m (94 ft) (21.0 m (69 ft) from the bottom of the SFP) to support key assumptions in the analyses of other DBEs (e.g., LOCA, or the use of the decay heat removal system (DHRS) in non-LOCA events).

The new fuel elevator is used for lowering only new fuel assemblies from the operating floor level to the bottom of the SFP. Further, DCA Part 2, Tier 2, Section 9.1.4.2.2, states that the new fuel elevator has a weigh system that measures load in the basket. If a load is detected, the basket cannot be raised, preventing the inadvertent raising of a spent fuel assembly if it is misplaced on the new fuel elevator. This design feature effectively prevents movement of a spent fuel assembly by the new fuel elevator above the established minimum water level.

Based on the above evaluation, the staff finds that the FHE system design complies with the requirements of GDC 61 with respect to 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.

DCA Part 2, Tier 2, Section 9.1.4.5, "Instrumentation and Control," describes all relevant interlocks associated with the FHE system. The interlocks provided 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 new fuel elevator.

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 NUREG-0554, as supplemented

with the requirements of ASME NOG-1 for a Type 1 crane.

With regard to the NFJC, DCA Part 2, Tier 1, Section 3.4, and DCA Part 2, Tier 2, Section 9.1.4, describe a restrictive traveling range that will prevent the NFJC from moving a new fuel assembly over the stored fuel assemblies in the SFP. The staff finds this description 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 as a result 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 staff reviewed the FHE system for compliance with the requirements of GDC 62, with respect to the prevention of criticality in fuel handling systems. Compliance with the requirements of GDC 62 is based in part on the guidelines of ANSI/ANS 57.1.

As indicated in DCA Part 2, Tier 2, Section 3.1.6.3, "Criterion 62—Prevention of Criticality in Fuel Storage and Handling," the design and controls for operation of the FHE and fuel storage racks prevent an inadvertent criticality for new and spent fuel by using a geometrically safe configuration for the storage racks in the SFP, restraints and interlocks in the design, and programmatic controls for the operation of the FHM.

The applicant analyzed the criticality of the stored new and spent fuel in DCA Part 2, Tier 2, Section 9.1.1, and Section 9.1.1 of this SER provides the associated staff evaluation.

As indicated in DCA Part 2, Tier 2, Section 9.1.4.5, the FHM design incorporates an industrialized computer with a graphical user interface driver for the operator interface and a database that allows complete mapping of alphanumeric core locations into the FHM bridge and trolley positions. During refueling, this design feature, together with the proper sequence of core loading as detailed in refueling operating procedures, will help minimize the misplacement of fuel assemblies that can result in inadvertent criticality of the open core.

Based on the above evaluation, the staff finds that the FHE system design complies with GDC 62 requirements with respect to the prevention of criticality.

9.1.4.5 Initial Test Program

As discussed in Section 14.2 of this SER, the staff reviewed the applicant's ITP in accordance with the review guidance in SRP Section 14.2, Revision 3, "Initial Plant Test Program—Design Certification and New License Applicants," issued March 2007; RG 1.68; and RG 1.206, "Applications for Nuclear Power Plants."

DCA Part 2, Tier 2, Table 14.2-51, describes the system performance testing that will demonstrate proper operation of all FHE system control circuits and associated interlocks, as well as proper transport of a dummy assembly by the NFJC, the new fuel elevator, and the FHM. Testing of the FHM will include movement from the new fuel elevator to the reactor core location in the RFP, and then from the core location to the storage racks in the SFP. The FHM will also be load tested to 125 percent of its rated capacity in accordance with ASME NOG-1 requirements.

Also, DCA Part 2, Tier 2, Table 14.2-51, includes testing for verification of interlocks that

restrict the NFJC from handling a new fuel assembly over storage racks in the SFP.

The staff's evaluation of the ITP is discussed in Section 14.2 of this SER.

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 COL information item numbers and descriptions related to the FHE system, from DCA Part 2, Tier 2, Table 1.8-1.

Table 9.1.4-1 NuScale COL Information Items for Section 9.1.4

COL Item No.	Description	DCA Part 2, Tier 2, Section
9.1-3	A COL applicant that references the NuScale Power Plant design certification will develop procedures related to the transfer of spent fuel to a transfer cask.	9.1.4
9.1-4	A COL applicant that references the NuScale Power Plant design certification will provide the periodic testing plan for fuel handling equipment.	9.1.4

The staff reviewed the proposed COL items and found them acceptable. For COL Item No. 9.1-3, directing the COL applicant to address the development and implementation of these procedures will ensure that an applicant can minimize the likelihood of a load drop event. For COL Item No. 9.1-4, because of the more frequent use of the FHE to support refueling activities of 12 NPMs in every 24-month period, the staff agrees that a unique testing plan developed by a COL applicant using recommendations from the equipment manufacturer is appropriate for the NuScale design of the FHE system. The staff also finds that no additional COL item is needed for the FHE system.

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 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

DCA Part 2, Tier 1: The applicant provided a general description of the OHLHS in DCA Part 2, Tier 1, Section 3.10, "Reactor Building Crane."

DCA Part 2, Tier 2: The applicant provided a detailed description of the OHLHS in DCA Part 2, Tier 2, Section 9.1.5.

For the NuScale 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 RBC. Other OHLHS equipment includes the module lifting adapter (MLA), 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. DCA Part 2, Tier 2, Figure 9.1.5-1, "Safe Load Paths and Heavy Load Exclusion Zones," and Figure 9.1.5-2, "Safe Load Paths," show safe load paths and heavy load exclusion zones. DCA Part 2, Tier 2, Figure 9.1.5-3, shows the design configuration of the RBC, and DCA Part 2, Tier 2, Figure 9.1.5-4, shows design details of the MLA. Finally, DCA Part 2, Tier 2, Table 9.1.5-1, gives the design data for major components of the heavy load handling system

ITAAC: DCA Part 2, Tier 1, Table 3.10-1, "Reactor Building Crane Inspections, Tests, Analyses, and Acceptance Criteria," specifies the ITAAC for the RBC. These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: The applicant provided no specific GTS for the OHLHS. However, GTS 3.5.3, which addresses controls for operation of the UHS, has provisions related to the minimum water level for the reactor pool inventory that affect the use of the RBC during refueling of one out-of-service NPM. Section 9.2.5 and Chapter 16 of this SER document the staff's review of TS 3.5.3 requirements and their associated bases.

Technical Reports: No technical reports are associated with the OHLHS.

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 among multiple operating units at one single site

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 design, in DCA Part 2, Tier 2, Section 9.1.5, the applicant defined heavy loads as "loads whose weight is greater than the combined weight of a single fuel assembly and control rod assembly." A fuel assembly weighs 376 kg (830 lbs), and the control rod assembly weighs 19.5 kg (43 lbs), resulting in a combined weight of 396 kg (873 lbs). The applicant elected to use 410 kg (900 lbs) as the threshold value, as stated in DCA Part 2, Tier 2, Section 9.1.5. The staff finds the proposed definition acceptable because it is consistent with the use of the FHM in the NuScale design to handle this combined load during refueling operation for one NPM.

The OHLHS consists of equipment and components used for critical load handling. DCA Part 2, Tier 2, 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
- damage to equipment essential to achieve or maintain safe shutdown

In DCA Part 2, Tier 2, Section 9.1.5.2, "System Description," the applicant described various OHLHS components, as discussed below. Major components of the OHLHS include the RBC, the MLA, and the wet hoist.

The RBC is a bridge crane that rides on rails anchored to the RXB at elevation 44.35 m (145 feet, 6 inches). The RBC consists of a bridge, a trolley, a main hoist, and two auxiliary hoists. The RBC is designed as a single-failure-proof crane in accordance with the guidelines of NUREG-0554, as supplemented by the requirements of ASME NOG-1 for Type I cranes. The RBC is also designed to meet seismic Category I requirements.

The welded-construction MLA provides the connection method for the RBC to lift and carry the NPM from the operating bay to the refueling bay and dry dock. The MLA is designed as a single-failure-proof special lifting device in accordance with the requirements of ANSI N14.6-1993, "Radioactive Materials—Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More." The MLA is also designed to meet seismic Category I requirements.

The wet hoist is the main below-the-hook structure for lifting equipment underwater. The wet hoist is designed to be single-failure proof, consistent with the requirements of ASME NOG-1. The wet hoist is also designed to meet seismic Category I requirements.

The staff noted that DCA Part 2, Tier 2, Section 9.1.5.2.3, "System Operation," mentions the existence of other cranes used to support various refueling activities (i.e., the traveling wall-mounted jib cranes). In DCA Part 2, Tier 2, Section 9.1.5.2.3 and Table 9.1.5-1 include the

required design information for these traveling jib cranes. The applicant selected a rated lifting capacity of 0.9 metric ton (2,000 lbs or 1 ton), reflecting the anticipated largest load to be handled by these cranes. The staff finds that the DCA contains adequate design information for these cranes.

In DCA Part 2, Tier 2, Section 9.1.5.2.3, the applicant stated, in part, that the wet hoist is used with the RBC to lift the spent fuel cask and transport it to and from the cask loading area in the RFP. The main hoist has a rated capacity of 770 metric tons (850 tons), and the wet hoist a rated capacity of 230 metric tons (250 tons). The applicant's DCA dos not include design information for the spent fuel cask, because the final design of the spent fuel cask is the responsibility of a COL applicant covered under COL Item 9.1-6, as discussed below. The staff finds that the DCA contains adequate design information for these cranes.

Major tools used to support refueling operations include the CFT, the RFT, and the module inspection rack.

The CFT is mounted at the bottom of the RFP in the RXB. It is used to hold the NPM in the upright position and to assemble and disassemble the lower parting flange on the containment vessel (CNV). The CFT is composed of a CNV support stand, guides, and associated tooling for assembly, disassembly, and inspection of the CNV lower parting flange connection and fasteners. The CFT is designed to meet seismic Category II requirements.

The RFT is mounted at the bottom of the RFP adjacent to the CFT. The RFT supports the lower portion of the reactor pressure vessel (RPV), which contains the core, during refueling operation. The RFT comprises an RPV support stand, guides, and remotely operated equipment that performs closure bolt installation and tensioning for assembly and disassembly of the RPV lower parting flange connection and fasteners. The RFT is designed to meet seismic Category I requirements.

The module inspection rack is one of the three components of the module assembly equipment; the others are the module import trolley and the module upender. The function of the module assembly equipment is to facilitate the delivery of the NPMs to the RFP area during the NPM construction period. Only the module inspection rack is used to support refueling activities during commercial operations of the NPMs. The module inspection rack is designed to meet seismic Category II requirements.

The RBC is used to seat the NPM into the CFT, but it remains connected to the upper portion of the module for later transport to the RFT. After being separated from the lower part of the CNV, the RBC is used to move and seat the upper portion of the module, which also includes the RPV, into the RFT to allow disassembly of the RPV. After being separated from the lower RPV, the upper CNV, with the upper RPV still attached, is transported by the RBC to the module inspection rack in the flooded dry dock. During the remaining refueling process, the lower CNV remains in the CFT and the lower RPV, including the core, remains in the RFT. Refueling operations are conducted with the use of the FHM, as described in DCA Part 2, Tier 2, Section 9.1.4. Section 9.1.4 of this SER documents the associated staff evaluation.

To reduce the likelihood of an accidental heavy load drop, SRP Section 9.1.5.III.3 includes guidelines to be implemented for heavy load handling in areas of the facility housing safety-related SSCs. In accordance with guidance in SRP Section 9.1.5, the application should conform to general programmatic guidelines for design, operation, testing, maintenance, and inspection as specified in Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear

Power Plants: Resolution of Generic Technical Activity A-36," issued July 1980. As indicated in DCA Part 2, Tier 2, Section 9.1.5.3, "Safety Evaluation," and COL Item 9.1-7, the COL applicant will develop 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.

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 in part on conformance to the guidelines of NUREG-0554, as supplemented by the requirements of ASME NOG-1 for Type 1 cranes and ANSI N14.6 or ASME B30.9 for lifting devices. The guidelines of NUREG-0554 and the requirements of ASME NOG-1 for Type 1 cranes include provisions for design, installation, inspection, testing, and maintenance of cranes.

As presented above, in the NuScale design, the primary OHLHS equipment that will handle heavy loads in the vicinity of or involving spent fuel or safety-related components are the RBC and its associated load handling attachments (i.e., the MLA and the wet hoist). The RBC and its associated main hoist, two auxiliary hoists, and wet hoist are designed as single-failure-proof components in accordance with the guidelines of NUREG-0554 and ASME NOG-1 requirements for a Type 1 crane. The MLA is also designed as a single-failure-proof component in accordance with ANSI N14.6 requirements for special lifting devices.

Based on the above evaluation, the staff finds that the OHLHS design 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 Section 2.5 of NUREG-0554.

Regulatory Position C.1.i 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. Section 2.5 of NUREG-0554 specifies that single-failure-proof 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 wheels prevented from leaving the track during a seismic event.

As discussed above, the RBC, its hoists, and associated handling tools are designed to meet seismic Category I requirements. Further, in DCA Part 2, Tier 2, Section 9.1.5.3, the applicant stated, in part, that "[u]pon the onset of an earthquake, a seismic switch on the RBC disconnects power. The trolley, bridge, and hoist stop and the brake set. Earthquake restraints keep the trolley on the bridge and the bridge on the runway." The staff finds that these design features conform to the guidance in Section 2.5 of NUREG-0554, more than satisfy Regulatory Position C.1.i of RG 1.29, and therefore are acceptable.

As discussed above, the CFT, the RFT, and the module inspection rack are used to provide

structural supports for holding heavy components (e.g., the lower CNV, the lower RPV containing the reactor in shutdown configuration) in their desired upright positions during refueling of one NPM. From the heavy load handling perspective, the staff finds the assigned seismic Category I for the RFT, and the assigned seismic Category II for the CFT and the module inspection rack, commensurate with the load they would hold and, therefore, acceptable.

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. RG 1.13, Regulatory Position C.5, provides the following guidance for meeting these requirements in spent fuel storage areas:

> Cranes capable of carrying heavy loads should be prevented, preferably by design rather than by interlocks, from moving over the pool. Furthermore, the spent fuel storage facility design should have at least one of the following provisions with respect to the handling of heavy loads, including the spent fuel cask:
>
> a) Cranes should be designed to provide single-failure-proof handling of heavy loads, so that a single failure will not result in the crane handling system losing the capability to perform its safety function.
>
> b) The spent fuel cask-loading area should be designed to withstand, without significant leakage of the adjacent spent fuel storage, the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted.

The NuScale RBC has been designed as a single-failure-proof crane, in conformance with the guidelines of NUREG-0554, as supplemented by the requirements of ASME NOG-1 for Type 1 cranes. Therefore, the applicant is not required to evaluate the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted.

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, on general programmatic controls for the design, operation, testing, maintenance, and inspection of heavy load handling systems, with a particular emphasis on establishing safe load paths for critical load handling.

In DCA Part 2, Tier 2, Section 9.1.5.2.3, the applicant stated the following:

> The RBC is operated to move an NPM between its installed operating position in the reactor pool to the refueling pool and back. Travel paths are determined, and attributes entered into the RBC control system. Each task is specified and scheduled by the crane operator. Figure 9.1.5-1 shows the safe load paths.

> Heavy load exclusion zones and safe load paths are defined in operating procedures and equipment drawings. Heavy load exclusion zones are marked in the plant areas where the load cannot be handled. 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 the alignment of the RBC with the NPM for engagement with the RBC prior to performing lifting operations. Heavy load exclusion zones are dependent on whether or not there is a load on the RBC. 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 DCA Part 2, Tier 2, Figure 9.1.5-1, confirmed the defined safe path for the RBC to move the NPM within the reactor pool and the RFP, as well as the RBC exclusion zone over the SFP. Also, in DCA Part 2, Tier 2, Figure 9.1.5-2, the applicant delineated the coverage for heavy load handling by the RBC auxiliary hoists. In DCA Part 2, Tier 2, Section 9.1.5.5, "Instrumentation and Control," the applicant indicated that limit switches on the runway and bridge determine the edges of the heavy load exclusion zones for the RBC. This, in conjunction with positioning and weighing sensing capability, ensures that the RBC does not travel within heavy load exclusion zones.

Therefore, the staff finds the above description of the interlocks and controls to be consistent with the guidelines of NUREG-0612 with respect to defining safe load paths.

The staff noted that DCA Part 2, Tier 2, Section 9.1.5, defines that the safe load path used to transport the NPM within the RFP area of the common reactor pool/RFP/SFP will also be used for the spent fuel cask. Further, the staff noted that COL Item 9.1-6, as discussed above, addresses, in part, this particular safe load path.

As mentioned above, the RBC and its main hoist, two auxiliary hoists, wet hoist, and attached handling tools are designed as single-failure-proof components to minimize the likelihood of a load drop event. The staff finds that these design features are consistent with the intent of RG 1.13, Regulatory Position C.5 and, therefore, are acceptable.

In DCA Part 2, Tier 2, Section 9.1.5.3, the applicant stated, in part, that "operator training, handling, handling system design, load handling instructions, and equipment inspection provide reasonable assurance of a reliable operation of the handling system." In addition, in DCA Part 2, Tier 2, Section 9.1.5.4, the applicant stated that the RBC is inspected, tested, and maintained in accordance with ASME B30.2, "Overhead and Gantry Cranes."

In COL Item 9.1-7, the applicant stated the following:

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

SRP Section 9.1.5.III.C states that operators should be trained and qualified and conduct themselves in accordance with Chapter 2-3.1 of ASME B30.2-2005. SRP Section 9.1.5.III.d, states that the crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ASME B30.2 prior to use. The staff finds that COL Item 9.1-7 includes key elements that the COL applicant will address in its heavy load handling program, and the staff will review specific conformance to the requirements of ASME B30.2 at that 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 12 NPMs share the OHLHS to support refueling one NPM at a time. The OHLHS design allows for the performance of refueling activities on one module with minimum impact on the operation of the other modules, including potential shutdown and cooldown. 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 Initial Test Program

The staff reviewed the applicant's ITP in accordance with the guidance in SRP Section 14.2.

DCA Part 2, Tier 2, Table 14.2-52, "Reactor Building Crane Test #52," 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. Testing of the RBC also includes the performance of associated handling tools for the disassembly and reassembly of the NPM using the CNV support stand, the RPV support stand, and the module inspection rack. The RBC will also be load-tested to 125 percent of its rated capacity in accordance with ASME NOG-1 requirements.

The staff finds this performance testing of the OHLHS acceptable because it can validate all basic design features of the RBC through its actual transport operation with an NPM. Section 14.2 of this SER provides additional information on the staff's evaluation of the ITP.

9.1.5.6 Technical Specifications

The NuScale design includes a new application of the RBC that is not found in the existing pressurized-water reactor (PWR) plants in the United States. During refueling of one NPM while the remaining 11 NPMs are in their normal operating modes, the RBC is used to transport the out-of-service NPM from its operating bay to the RFP area of the common reactor pool/RFP/SFP. This common water inventory acts as the UHS for the operating NPMs and provides the cooling medium for the spent fuel assemblies stored in the SFP. For the safe handling of the NPM, the rated capacity of the RBC main hoist is determined based on the weight of a partially immersed NPM. At the highest lift point for the NPM during refueling, the credited buoyancy force on the NPM is based on a pool water level of 20.1 m (66 ft) from the bottom of the pool. This required water level for the safe handling of the NPM by the RBC is captured as part of GTS LCO 3.5.3 for controls of UHS operation. LCO 3.5.3 establishes two water-level limits—a 20.7 m (68 ft) limit for safe handling of the NPM and a 16.8 m (55 ft) limit

for the safety functions of the emergency core cooling system (ECCS). In accordance with LCO 3.5.3, if the UHS water level is less than 20.7 m (68 ft) but more than 16.8 m (55 ft), ongoing movement of the NPM, if any, must be suspended immediately, and the water level can be restored within the next 30 days. Further, if the UHS water level is less than 16.8 m (55 ft), the licensee must immediately begin action to restore the water level above 16.8 m (55 ft), and the water level must be restored to above 16.8 m (55 ft) within 24 hours. The staff evaluates these TS requirements and their associated bases in Chapter 16 of this SER.

9.1.5.7 Combined License Information Items

Table 9.1.5-1 lists COL information item numbers and descriptions related to the OHLHS, from DCA Part 2, Tier 2, Table 1.8-1.

Table 9.1.5-1 NuScale COL Information Items for Section 9.1.5

COL Item No.	Description	DCA Part 2, Tier 2, Section
9.1-5	The COL applicant that references the NuScale Power Plant design certification will describe the process for handling and receipt of critical loads including NuScale Power Modules.	9.1.5
9.1-6	The COL applicant that references the NuScale Power Plant design certification will provide a design for a spent fuel cask and handling equipment including procedures and programs for safe handling.	9.1.5
9.1-7	The COL applicant that references the NuScale Power Plant design certification will provide a description of the program governing heavy loads handling. The program should address: • operating and maintenance procedures; • inspection and test plans; • personnel qualifications and operator training; and • detailed description of the safe load paths for movement of heavy loads.	9.1.5

The NuScale design includes a new application of the RBC that was not found in existing U.S. PWR plants. In particular, the use of the RBC to transport an NPM that contains a full core, albeit in shutdown condition, imposes additional concerns about reactor core safety given the potential for a load drop event as a result of operator errors or crane failure. For COL Item 9.1-5, development of procedures for this unique handling of the NPM by a COL applicant is preferably listed separately in DCA Part 2, Tier 2, Section 9.1.5, rather than being broadly covered by a COL item in Chapter 13 related to the development, in general, of system operating procedures. Section 9.1.5.4 of this SER documents the staff's evaluation of COL Items 9.1-6 and 9.1-7.

Based on the above discussion, no other COL item is needed for the OHLHS.

9.1.5.8 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 Water Systems

9.2.1 Station Service Water System

DCA Part 2, Tier 2, 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 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 submerged 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 the above 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
- CVCS nonregenerative heat exchangers
- containment evacuation system (CES) condensers and vacuum pumps
- process sampling system (PSS) coolers and analyzer cooler temperature control units

The RCCWS comprises two identical subsystems each supporting up to six reactor modules. The two subsystems are independent and do not have the ability to cross-connect. The RCCWS transfers the heat from these systems to the SCWS and then to the environment through the SCWS cooling tower.

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.

9.2.2.2 Summary of Application

DCA Part 2, Tier 1: No Tier 1 information is associated with this section.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 9.2.2, "Reactor Component Cooling Water System," provides a complete description of the RCCWS, including the RCCWS design bases, system descriptions, component descriptions, system operation, safety evaluation, and information on inspection and testing of the RCCWS.

In DCA Part 2, Tier 2, Section 9.2.2.3, "Safety Evaluation," the applicant stated that during normal and off-normal conditions, sufficient redundancy and cross-connectivity within each subsystem exist to remove heat from the serviced systems and that the RCCWS does not perform any safety-related or risk-significant function. It also stated that RCCWS cooling is not required for any safety-related or risk-significant components to perform their function. Also, during and after an accident, the RCCWS is not relied upon to remove heat from the affected NPM.

ITAAC: The applicant has not proposed any ITAAC related to the RCCWS.

Technical Specifications: No GTS requirements are associated with the RCCWS.

Technical Reports: No technical reports are associated with 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, and only the following regulatory requirements noted in SRP Section 9.2.2 are relevant to this particular 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 turbine gland sealing system design for the control of releases of radioactive materials to the environment

- GDC 64, "Monitoring Radioactivity Releases," as it relates to the turbine gland sealing system 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 results of the staff's review are provided below.

9.2.2.4.1 System Design Considerations

Design Basis

The design basis for the RCCWS states that 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. The applicant 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 in designing the RCCWS.

The staff reviewed the information on the RCCWS design bases provided in DCA Part 2, Tier 2, Section 9.2.2.1, and found that the criteria addressed those items identified in Regulatory Position C.I.9.2.2.1 of RG 1.206. Based on its review, the staff finds the design basis appropriate relative to the important-to-safety, but not safety-related, functions that the RCCWS is designed to perform.

System Design and Operation

In DCA Part 2, Tier 2, Section 9.2.2.2, "System Description," and Section 9.2.2.2.3, "System Operation," include information on the design and operation of the RCCWS. DCA Part 2, Tier 2, Figure 9.2.2-1, "Reactor Component Cooling Water System Diagram," shows the simplified diagram for the RCCWS.

The RCCWS is located in the RXB and comprises two identical subsystems, each designed to support up to six NPMs at a time. 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 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.

As shown in DCA Part 2, Tier 2, Figure 9.2.2-1, each RCCWS subsystem also incorporates an expansion tank and duplex strainer connected to the suction side of the three RCCWS pumps that deliver water to the two RCCWS heat exchangers. A single NPM DBE will isolate only the containment for the affected NPM. Thus, its effect will be limited to a single module, and the RCCWS will continue to operate, but CRDM, CVCS, CES, and PSS heat loads for the affected reactor module are eliminated when the containment isolation valves are shut.

In its supplemental response dated December 12, 2017 (ADAMS Accession No. ML17346B303), to Request for Additional Information (RAI) 9101, Question 09.02-02-4, the applicant provided RCCWS flow rates and heat loads, along with the basis for the sizing of the system. Based on the design information provided by the applicant in its RAI response, the staff confirmed that the 2.2×10^7 kJ/h (21 MMBtu/h) capacity for the RCCWS heat exchangers specified in DCA Part 2, Tier 2, Table 9.2.2-1, "Reactor Component Cooling Water System Design Data," is sufficient to accommodate the cooling requirements of the RCCWS components that it supports.

The applicant also stated in its response that RCCWS cooling is not required for any safety-related or risk-significant components to perform their functions. The staff found that the CRDMs are safety related because of their function of safe shutdown of the reactor and that the electromagnetic drive coils and rod position indication that are cooled by the RCCWS are part of the control rod drive system. However, 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.

Based on its review of the information provided about the design and operation of the RCCWS, the staff has confirmed that, 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, and RCCWS operation is not required to support NPM cooling during shutdown or postaccident conditions. 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 a 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.e for the safety-related portions of the system and Regulatory Position C.1.i for the portions of the system that are not safety related.

The RCCWS is located in the RXB, which is a seismic Category I structure designed to protect SSCs from extreme winds and missiles that may result from natural phenomena such as earthquakes, tornadoes, and hurricanes. DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 3.4.2, “Protection of Structures Against Flood from External Sources.”

DCA Part 2, Tier 2, Section 3.2, “Classification of Structures, Systems and Components,” categorizes SSCs based on safety importance and other considerations. DCA Part 2, Tier 2, Table 3.2-1, “Classification of Structure, Systems, and Components,” gives the location, safety classification, and seismic category for the RCCWS. The staff noticed 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 in DCA Part 2, Tier 2, Table 3.2-1. The applicant included RCCWS supply and return lines in the vicinity of the containment as part of the containment system in DCA Part 2, Tier 2, Table 3.2-1.

The staff reviewed the information on the RCCWS in DCA Part 2, Tier 2, Figure 9.2.2-1 and Table 3.2-1, to verify that RCCWS components that perform important-to-safety functions, and RCCWS SSCs whose failure could affect SSCs important to safety, were appropriately designed in accordance with the guidance in RG 1.29. According to DCA Part 2, Tier 2, Table 3.2-1, 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. DCA Part 2, Tier 2, Table 3.2-1, indicates that for 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 to conform to RG 1.29, Regulatory Position C.1. The table also indicates that control rod drive system cooling water piping and pressure relief valves are designated as Seismic Classification II, and DCA Part 2, Tier 2, Section 9.2.2.2.1, states that the RCCWS piping from the NPM disconnect flange to the pipe gallery wall is seismic Category II because the piping does not provide a safety-related function, but it must be properly restrained to prevent effects on other safety-related seismic Category I SSCs such as the CNV, other containment isolation valves, and equipment on the top head of the module. The staff finds that Seismic Classification II is appropriate and conforms to RG 1.29, Regulatory Position C.1.i.

Based on the above discussion, the staff concludes that the RCCWS as described in the DCA 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 SER. 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." DCA Part 2, Tier 2, 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 G 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. Compliance with 10 CFR 50.49 for the qualification of equipment located in a harsh environment is addressed in Section 3.11 of this SER. 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 unit design includes 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.

Each RCCWS supports up to six NPMs at a time. The RCCWS supplies cooling water that directly supports heat removal from the CRDM, CVCS, CES, and PSS. 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. This event is evaluated by the applicant in DCA Part 2, Tier 2, Section 15.1.6, and by the staff in Section 15.1.6 of this SER. A total failure of an RCCWS subsystem would eventually result in the manual shutdown of up to six NPMs because of rising CRDM temperatures, which are indicated in the control room, but such failure does not prevent safety-related NPM functions and does not result in a DBE. Therefore, the staff finds that the failure of components in the RCCWS does not significantly impair the ability of other NPMs to perform their safety functions, and 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 unit design to monitor and suitably control the release of radioactive materials during normal operation, including anticipated operational occurrences.

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 DCA Part 2, Tier 2, 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 discussed in DCA Part 2, Tier 2, Section 11.5.2.2.12, "Reactor Component Cooling Water System." In addition, a single adjacent-to-line radiation monitor is provided on the normally noncontaminated RCCWS drain tank to ensure the prompt identification of reactor component cooling water radiological contamination. 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 DCA 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 design certification 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 RCCWS reduces the possibility of radioactive leakage to the environment by providing an intermediate barrier between radioactive or potentially radioactive systems and the SCWS. DCA Part 2, Tier 2, Section 9.2.2.2.3, states that the RCCWS design minimizes the contamination of the facility and the environment in accordance with RG 4.21, Revision 0, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," issued June 2008, and 10 CFR 20.1406.

The staff reviewed the design of the RCCWS for compliance with the requirements of 10 CFR 20.1406. In DCA Part 2, Tier 2, Section 9.2.2.3, the applicant stated that 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 in DCA Part 2, Tier 2, Table 12.3-32, "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 SER 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 DCA 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 Combined License Information Items

Table 9.2.2-1 lists COL relevant information items and descriptions from DCA Part 2, Tier 2.

Table 9.2.2-1 NuScale COL Information Items

COL Item No.	Description	DCA Part 2, Tier 2, Section
9.2-1	A COL applicant that references the NuScale Power Plant design certification will select the appropriate chemicals for the reactor component cooling water system based on site-specific water quality and materials requirements.	9.2.2.2.2

9.2.2.8 Conclusion

Based on the review of the information provided as described above, the staff finds the RCCWS design acceptable because it meets the applicable acceptance criteria of SRP 9.2.2 and the applicable regulatory requirements of GDC 2, 4, 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 three DWS pumps. Up to 12 NPMs share the DWT skid, DWST, and DWS. The DWST is sized to support full 12-module demineralized water demand.

The DWS provides plant support during abnormal conditions by providing additional makeup water to the SFPCS to compensate for inventory loss and to the condenser for emergency fill. 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

DCA Part 2, Tier 1: The NuScale DCA has no Tier 1 entries for this area of review.

DCA Part 2, Tier 2: Tier 2 of DCA Part 2 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.

ITAAC: The applicant has not proposed any ITAAC related to the DWS.

Technical Specifications: No GTS requirements are associated with the DWS.

Technical Reports: No technical reports are associated with 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, to facilitate evaluation of DCA Part 2, Tier 2, 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

In addition, the following regulatory requirement also applies to the DWS:

- 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 results of the staff's review are provided below.

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 nonseismic portions of the DWS.

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

DCA Part 2, Tier 2, Section 9.2.3.1, "Design Basis," states that the DWS does not perform safety-related functions, is not credited for the mitigation of DBAs, and has no safe-shutdown functions. DCA Part 2, Tier 2, Section 9.2.3.3, states that, 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 Section 3.2.

The staff reviewed the DCA information on 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 as 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.1.i, for all areas except within the control building (CRB) and RXB.

Within the CRB, DCA Part 2, Tier 2, Section 9.2.3.1, states that 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 DCA Part 2, Tier 2, Table 3.2-1, 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 Position C.1.i 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.1.i.

Within the RXB, the DWS provides water to a variety of systems that are not safety related. However, DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, 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 SER. 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.1.i. 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 design, up to 12 NPMs share the DWT skid, DWST, and DWS pumps. However, as indicated in DCA Part 2, Tier 2, Section 9.2.3.2, “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 DCA 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 design certification 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.

DCA Part 2, Tier 2, Section 9.2.3.1, states that 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 includes backflow preventers in lines that interface with radioactive waste processing systems and lines that contain radioactive liquids. In addition, the design includes isolation valves to secure flow in the event of a line break or other abnormal condition. As described in DCA Part 2, Tier 2, Section 11.5.2.2.16, "Demineralized Water System," DWS radiation monitors provide the control room with an early indication of leakage from radiologically contaminated systems to the clean DWS.

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 DCA Part 2, Tier 2, Section 12.3, and Section 12.3 of this SER includes the staff's review of those features.

Based on the above discussion, the staff concludes that the DWS design as described in the DCA 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

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

9.2.3.6 Technical Specifications

No GTS requirements are associated with the DWS.

9.2.3.7 Combined License Information Items

In accordance with DCA Part 2, Tier 2, Table 1.8-2 and Section 9.2.3, the applicant has not identified any COL information items that are directly applicable to the DWS. The staff did not identify any additional COL items that should be in DCA Part 2, Tier 2, Table 1.8-2.

9.2.3.8 Conclusion

Based on the review of the information provided as 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 and sanitary water system (PSWS) is a nonsafety-related system that provides potable water for human use and sanitary water collection throughout the plant for treatment and discharge.

9.2.4.2 Summary of Application

DCA Part 2, Tier 1: There are no Tier 1 entries for this area of review in the DCA.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 9.2.4, "Potable and Sanitary Water Systems," provides information on the PSWS. The applicant indicated that the PSWS serves no safety-related functions, is not credited for mitigation of DBAs, and has no safe-shutdown functions. The design basis states that the PSWS is not required to function during or after a natural phenomenon event or other events that result in the generation of missiles, pipe whipping, or discharging fluids and that there are no safety-related, risk-significant or safe-shutdown functions in the potable water system (PWS) that are shared between NPMs.

DCA Part 2, Tier 2, Section 9.2.4.2, "System Description," states that the PWS provides piping, valves, and other control components to distribute potable water to final use locations, and indicates potable water usage includes drinking fountains, kitchen/breakroom facilities, sinks, showers, water closets, and emergency eyewash/shower stations. It also states that any connection to or from other water systems is accomplished using isolation devices, such as backflow preventers or air gaps, between the PWS and interfacing systems to prevent any system cross-contamination. The overall potable water capacity will be based on the anticipated number of onsite personnel for a 24-hour period during normal operations.

ITAAC: The applicant has not proposed any ITAAC related to the PWS.

Technical Specifications: There are no GTS associated with the PWS.

Technical Reports: There are no technical reports associated with the PWS.

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 given in SRP Section 9.2.4, Revision 3, "Potable and Sanitary Water Systems," issued March 2007, and are summarized below:

- 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 PSWS

In addition, the following regulatory requirements also apply to the PWS:

- 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

- 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 design in accordance with the review procedures in SRP Section 9.2.4, Revision 3. The results of the staff's review are provided below.

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 Position C.1e of RG 1.29 for the safety-related portion of the system and Position C.1.i for the nonsafety-related portions of the system.

In DCA Part 2, Tier 2, Section 9.2.4, the applicant stated that the PWS has no safety-related functions, is not credited for mitigation of DBAs, and has no safe-shutdown functions. DCA Part 2, Tier 2, Table 3.2-1, indicates that all components of the PWS are nonsafety-related and non-risk-significant and Seismic Classification III. However, DCA Part 2, Tier 2, Section 9.2.4, "Safety Evaluation," states that portions of the system that are in proximity to seismic Category I SSCs are designed to seismic Category II. The same section of the DCA states that the PWS does not provide service directly to the RXB or the radwaste building, but potable water is supplied to the CRB.

The PWS piping in the CRB penetrating the control room envelope (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. In DCA Part 2, Tier 2, Section 9.2.4.2, the applicant stated 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 applicant also stated that each applicable line is designated seismic Category II from the outer wall of the CRE to and including the loop seal.

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 will not fail in a way that will result in incapacitating injury to occupants of the control room or cause the failure of seismic Category I SSCs that are required to withstand 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.

The staff noticed that in DCA Part 2, Tier 2, Section 9.4.1.1, the applicant stated that under certain postulated conditions the CRE is isolated and air is provided by the control room habitability system (CRHS). The applicant discussed the control room habitability and the CRE in DCA Part 2, Tier 2, Section 6.4. In that section, the applicant described the CRHS as a nonsafety-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 DCA Part 2, Tier 2, Section 6.4.4, the applicant stated that the CRE and the supporting habitability systems and components are not safety related. In addition, DCA Part 2, Tier 2, 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"

DCA Part 2, Tier 2, Section 9.2.4, states that 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 as described in the DCA does not incorporate sharing of any important-to-safety SSCs among the nuclear power modules. Therefore, the PWS complies 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 PSWS for compliance with the requirements of GDC 60 for the control of releases of radioactive materials. DCA Part 2, Tier 2, Section 9.2.4, states that the PWS does not provide service directly to the RBX or the radwaste building and that the PWS piping is not interconnected with other system piping that conveys radioactive materials. It also states that the PWS is separated from other nonradioactive plant water systems by the installation of backflow prevention measures, such as reduced pressure backflow prevention devices or air gaps, where appropriate. 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 above, PWS piping is not interconnected with other system piping that conveys radioactive materials. Additionally, there is no PWS piping in the radwaste or RBX, and therefore, 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, as described in the DCA, complies with the requirement in 10 CFR 20.1406.

9.2.4.5 Initial Test Program

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

9.2.4.6 Technical Specifications

There are no GTS requirements associated with the PWS.

9.2.4.7 Combined License Information Items

Table 9.2.4-1 lists the COL information items and descriptions for the PWS from Table 1.8-2 of DCA Part 2, Tier 2.

Table 9.2.4-1 NuScale COL Information Items for Section 9.2.4

COL Item No.	Description	DCA Part 2, Tier 2, Section
9.2-2	A COL applicant that references the NuScale Power Plant design certification will describe the source and pre-treatment methods of potable water for the site, including the use of associated pumps and storage tanks.	9.2.4.2
9.2-3	A COL applicant that references the NuScale Power Plant design certification will describe the method for sanitary waste storage and disposal, including associated treatment facilities.	9.2.4.2

9.2.4.8 Conclusion

Based on the review of the information provided as described above, the staff finds the PSWS design acceptable because it meets applicable SRP acceptance criteria and 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 regarding the minimization of contamination.

9.2.5 Ultimate Heat Sink

9.2.5.1 Introduction

The NuScale 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 12 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 DHRS, the CNVs, and the spent fuel assemblies stored in the storage racks, (2) providing borated water for reactivity control during refueling, and (3) radiation shielding for the spent fuel assemblies and NPMs. During accident scenarios, the NuScale design credits the safety-related water inventory stored in the UHS to passively remove the decay heat. The applicant 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.

DCA Part 2, Tier 2, Section 9.1.2, discusses the design of provisions of the SFP and support components to ensure adequate safe storage of the spent fuel. DCA Part 2, Tier 2, Section 9.1.3, discusses the design and performance of the pool support systems, which include the active SFPCS, the active RPCS, the PCUS, and the PSCS. The staff evaluates these pool-related systems in Sections 9.1.2 and 9.1.3 of this SER. 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

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 3.6, describes the UHS and design commitments relating to the UHS.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, 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, DCA Part 2, Tier 2, Figure 9.2.5-1, provides the basic layout of UHS pools. DCA Part 2, Tier 2, Figure 9.2.5-2, provides the UHS makeup line and instrumentation diagram. DCA Part 2, Tier 2, Table 9.2.5-1, lists relevant UHS parameters, and DCA Part 2, Tier 2, Table 9.2.5-2, "Ultimate Heat Sink Heat Loads: Boil-off Event," lists the analysis results of a boiloff event.

ITAAC: DCA Part 2, Tier 1, Table 3.6-2, provides ITAAC for UHS piping and connections. These ITAAC are evaluated in Section 14.3 of this SER.

Initial Test Program: There are no inspections or testing of the UHS before plant operation.

Technical Specifications: The NuScale DCA Part 4 provides plant GTS in LCO 3.5.3 to specify the level, temperature, and boron concentration in the UHS for all operating modes.

Technical Reports: No technical reports are associated with the UWS.

9.2.5.3 Regulatory Basis

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

- 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 requested an exemption from certain electrical power provisions of GDC 44 and, as described in DCA Part 2, Tier 2, Section 3.1.4.15, identified PDC 44, which eliminates consideration of onsite and offsite electrical power. The staff's

evaluation of the exemption that supports PDC 44 is documented 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 provides 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 DCA in accordance with the applicable sections of SRP Sections 9.1.3 and 9.2.5. Section 9.1.3 of this SER contains the staff's evaluation of the pool support systems, as described in DCA Part 2, Tier 2, Section 9.1.3 (i.e., SFPCS, RPCS, PCUS, and PSCS).

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 SER contains the staff's review of the heat transfer to the UHS under accident conditions, and Section 9.1.3 of this SER includes the review under normal operating conditions.

In Chapter 8 of this SER, the staff has evaluated and concluded that an exemption from the requirements of electrical power requirements in GDC 17 and 18 and the electric power provisions of GDC 33, 34, 35, 38, 41, and 44 is justified and approved. In DCA Part 2, Tier 2, Section 3.1.4.15, the applicant proposed PDC 44 as an alternative to 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 assure that the system safety function can be accomplished, assuming a single failure.

The staff evaluated PDC 44 and found that it proposed no changes to the requirements of the cooling capability identified in GDC 44. Therefore, the staff determined that PDC 44 adequately addresses 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 water between 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 the requirements of PDC 44 and the applicable portions of GDC 61.

DCA Part 2, Tier 2, Section 9.2.5.2.1, states that the UHS will (1) remove the decay heat from each NPM, maintaining the core temperature at low levels after a LOCA resulting in the initiation of the ECCS, (2) provide sufficient cooling to the stored spent fuel assemblies in the SFP, and (3) maintain the spent fuel assemblies covered in water under operational scenarios.

In addition, DCA Part 2, Tier 2, Section 9.2.5.2.3, states the following:

> During an event where loss of electric power occurs, the volume of water already in the pool provides the inventory for the necessary heat removal. Upon loss of power, the reactor pool cooling and SFP cooling systems shut down. The UHS water expands as it heats and eventually begins to boil. Heat continues to be removed 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 design is such that UHS water boil-off will continue to remove heat from the power modules and spent fuel for greater than 30 days without the need for operator action, makeup water, or electric power.

The staff reviewed DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Sections 9.2.5.2.2, 9.2.5.2.3, and 9.2.5.4 and Table 9.2.5-1, provide the UHS thermal analysis, including 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. DCA Part 2, Tier 2, Section 9.2.5.4, states that the UHS is a passive system and does not require electric power (alternating current (ac) or direct current (dc)) to remove heat. An accident in one NPM concurrent with a loss of ac power is assumed to result in a shutdown of up to 11 NPMs. The heat load of 12 NPMs is assumed. No credit is needed for pool water addition for more than 30 days as boiloff of the pool water occurs to remove the heat. The applicant further states that

the initial conditions use the UHS pool water starting temperature and level based on the LCOs in Section 3.5.3 of Part 4, “Generic Technical Specifications,” of the NuScale DCA. GTS LCO 3.5.3 identifies two monitored water levels: 20.7 m (68 ft) from the pool floor provides buoyancy assumed in the RBC analysis, and 16.8 m (55 ft) from the pool floor provides margin above the minimum level required to support DHRS and ECCS operation in response to LOCA and non-LOCA DBEs.

The staff finds that in DCA Part 2, Tier 2, Table 9.2.5-1, “Relevant Ultimate Heat Sink Parameters,” the applicant indicated that the spent fuel pool cooling system (SFPCS) and the reactor pool cooling system (RPCS) penetrate the UHS at building elevation 25.9 m (85 ft) (pool level 18.3 m (60 ft)). DCA Part 2, Tier 2, Section 9.1.3, states that the pool support systems are 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. The staff finds that an SSE could cause failure of components in these systems and drain the UHS level to building elevation 25.9 m (85 ft) (pool level 18.3 m (60 ft)). Based on the seismic design of the SFPCS and the RPCS and the relative elevation of the pipe penetration, the staff determined that the UHS minimum safety water level of 16.8 m (55 ft) from the pool floor is adequately protected from siphoning.

DCA Part 2, Tier 2, Section 9.2.5.4, further indicates that the most limiting scenario evaluated assumes that the SFP contains 1,404 fuel assemblies representing a nominal 18 years of power operation and an additional 13 freshly offloaded assemblies from a recent refuel. The applicant assumed a 24-month refueling cycle; therefore, for a 12-NPM plant, it assumed a staggered refueling schedule of every 2 months. 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 and assumed no heat transfer between the pool liner and walls.

In DCA Part 2, Tier 2, Section 9.1.2.2.2, the applicant indicated that the SFP rack contains 1,694 fuel locations but, as a result of the FHM travel limitations and difficulty in reaching certain storage locations near the weir wall, credited only 1,393 fuel storage locations as available. However, as described in DCA Part 2, Tier 2, Section 9.2.5.4, the pool boiloff thermal calculation conservatively assumes that 1,404 storage locations are full of spent fuel plus an additional 13 freshly offloaded fuel assemblies. The staff finds that assuming a larger number of fuel assemblies in the SFP thermal calculation would result in a higher decay heat load and, therefore, a more conservative assumption. 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 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 11 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 12 units in operation. The staff discusses its evaluation of the different DBAs in Chapter 15 of this SER. The staff finds that using the highest heat load from the DBAs discussed in DCA Part 2, Tier 2, 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 unit (or units) for a minimum of 30 days without makeup unless acceptable makeup capabilities can be demonstrated.

The DCA indicates 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 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.

DCA Part 2, Tier 2, Section 9.2.5.1, states that, "Consistent with GDC 2 and 4, the UHS is designed to remain functional by withstanding the effects of natural phenomena and the dynamic effects of missiles resulting from equipment failures." In addition, DCA Part 2, Tier 2, Section 9.2.5.2.1, states that "the UHS has a makeup line that is designed to meet Regulatory Guide 1.26, Quality Group C; Regulatory Guide 1.29 Seismic Category I; and American Society of Mechanical Engineers BPVC Section III requirements and is protected from external natural phenomena."

DCA Part 2, Tier 2, Table 3.2-1, identifies the SSC classification of the UHS pool to be "A1," which means it is safety related and risk significant. In DCA Part 2, Tier 2, Table 3.2-1 and Section 9.2.5.2.1 state that the RXB concrete forming the UHS and UHS pool liner meets seismic Category I requirements.

Additional information on GDC 2 compliance for the SFP function is in DCA Part 2, Tier 2, Sections 9.1.2 and 9.1.3, and its evaluation is in SER Sections 9.1.2 and 9.1.3.

Based on the seismic design of the RB and the UHS as discussed above, the staff finds that the UHS is adequately designed and protected against the effects of natural phenomena like 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.

DCA Part 2, Tier 2, Section 9.2.5.4, states the following:

> The UHS is protected from the effects of turbine missiles, as described in Regulatory Guide 1.13, Regulatory Position C.3, without loss of the UHS safety functions specified in Section 9.2.5.1. Section 3.5.1 provides additional detail on protection from turbine missiles. The UHS is 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. Additionally, the physical location of the UHS within the RXB ensures

that the effects of equipment failures and events, and conditions that may occur outside the NPM have no reasonable likelihood of adversely impacting UHS safety functions.

The UHS is below grade and contains two heat sources (i.e., stored spent fuel and power modules). The RXB is serviced with a non-safety related heating ventilation and air conditioning system (see Section 9.4.2) that controls the environment. The resident heat sources in the UHS, the fact that it is below grade, and the controlled environment within the RXB prevents the UHS from reaching freezing temperatures.

A Safe Shutdown Earthquake (SSE) event can generate waves in the UHS. An analysis of sloshing determined that an SSE generates a maximum wave height of less than three feet [0.9 m]. The top of the normal SFP water level range is at 94′ [28.7 m] building elevation or six feet [1.8 m] lower than the operating floor at the 100′ [30.5 m] building elevation. The UHS pool level provides approximately six feet [1.8 m] of freeboard space from the normal pool level operating level for accommodation of sloshing waves or overfill conditions. Normal pool level control is monitored to initiate draining excess volume to the PSCS storage tank.

Based on the above information, the staff 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 unit and the effects of environmental pool sloshing dynamics.

The staff evaluation of the protection of essential SSCs against pipe failures is discussed in Section 3.6.1 of this SE. Additional information regarding GDC 4 compliance for the SFP function is in DCA Part 2, Tier 2, Sections 9.1.2 and 9.1.3, and its evaluation is in SER Sections 9.1.2 and 9.1.3.

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.

DCA Part 2, Tier 2, Section 9.2.5.1, states that "consistent with GDC 5, the UHS is a shared system that is capable of providing sufficient cooling to dissipate the heat from an accident in one unit and permitting the simultaneous and safe shutdown of the remaining units and maintaining them in a safe shutdown condition."

The staff verified that the NuScale UHS design capacity for abnormal and accident conditions, as described in DCA Part 2, Tier 2, Section 9.2.5, includes the combined heat loads from all NPMs. 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.

DCA Part 2, Tier 2, Section 9.2.5.1, states that the NuScale 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. Functional testing to ensure structural leaktight integrity is accomplished by maintaining the pool 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.

In addition, DCA Part 2, Tier 2, Section 9.2.5.5, "Inspection and Testing Requirements," states the following:

> To assure the integrity and capability of the UHS heat removal and shielding functions, the UHS design permits the inspection of important components, such as the pool water level instrumentation, the pool liner, and the outside surfaces of the containment vessels. Section 6.6 provides additional information related to the inspection of the containment vessel exterior. Verification of the pool water level ensures adequate water inventory to provide sufficient cooling for the necessary loads.
>
> Table 9.2.5-1 lists the minimum water levels. The integrity of the UHS is monitored by the pool leak detection system for evidence of liner leaks. The liner welds are inspected during power operation or shutdown for leak tightness.

Based on the above, the staff verified that the NuScale 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 Fuel Storage and Handling and Radioactivity Control

The NuScale 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 SER. The staff evaluates the SFP for compliance with GDC 61 in Section 9.1.3.4.4 of this SER.

9.2.5.4.7 Leakage and Makeup

DCA Part 2, Tier 2, Section 9.2.5.2.1, discusses potential UHS pool leakage and water makeup. Section 9.1.3 of this SER 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 fuels for abnormal and accident conditions.

DCA Part 2, Tier 2, Section 9.2.5.6, "Instrumentation Requirements," provides NuScale instrumentation design information, which includes temperature instrumentation and level instrumentation:

> The reactor pool cooling system and SFP cooling system temperature instrumentation is used to monitor the UHS. Temperature instrumentation located in the pool is seismic Category I. The UHS water level instrumentation is seismic Category 1.
>
> Water level instrumentation in the UHS is powered under normal and off-normal operational scenarios by the plant electrical distribution system and is battery backed. Remote power connections for the electrical distribution system are provided to enable repowering the equipment from outside the plant. UHS primary and backup level instrument channels are qualified for temperature, humidity, and radiation levels consistent with the pool water at saturation conditions for an extended period.
>
> The UHS pool level instrumentation mounting protects it from natural phenomena. To ensure redundancy, instruments are physically separated and mounted at opposite ends of the pools. Since the UHS communicates with pool areas while the water is above the weir, this provides multiple areas to monitor pool level. The location for each of the instruments provides assurance that a single event will not cause damage to all of the level instruments.
>
> The UHS level information is displayed in the main control room and remote shutdown station. Alarms alert the operator of these parameters during both normal and post-accident conditions. UHS Level instrumentation provides level information for post-accident monitoring. DCA Part 2, Tier 2, Figure 9.2.5-2 shows the relative location of the level instrumentation.

In addition, DCA Part 2, Tier 2, Section 20.1.4, has more details about UHS level instrumentation, which the staff evaluates in Section 20.1.4 of this SER.

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 design includes for initiating appropriate safety actions. Based on the above, the staff finds that the DCA 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 staff evaluates the ITP in Section 14.2 of this SER.

9.2.5.6 Technical Specifications

NuScale DCA Part 4 provides plant TS 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.

Chapter 16 of this SER provides additional review of the TS.

9.2.5.7 Combined License Information Items

There are no COL information items for the UHS.

9.2.5.8 Conclusion

Based on the above, the staff has determined that the standard design criteria and guidance as described in the DCA 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 condenser hotwell based on hotwell 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

DCA Part 2, Tier 1: The NuScale DCA has no Tier 1 entries for this area of review.

DCA Part 2, Tier 2: The CST supplies makeup water to the steam cycle. DCA Part 2, Tier 2, Section 9.2.6, "Condensate Storage Facilities," provides very limited information on the system. It states that the CST includes the tank, piping, valves, tank level, instrumentation, vents, drains, and piping connections to the hotwell. It also states that the CST does not provide makeup water to systems that remove heat from the reactor if normal heat removal methods fail or are unavailable.

DCA Part 2, Tier 2, Section 10.4.7.2.2, "Component Description," provides additional information about the CST, including component design data in DCA Part 2, Tier 2, Table 10.4-17, "Condensate and Feedwater System Design Data," and information on CST instrumentation in DCA Part 2, Tier 2, Table 10.4-19, "Condensate and Feedwater System Instrumentation."

ITAAC: The applicant has not proposed any ITAAC related to the CSF.

Technical Specifications: No GTS requirements are associated with the CSF.

Technical Reports: No technical reports are associated with the CSF.

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 particular 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 results of the staff's review are provided below.

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 RG 1.29, Regulatory Position C.1, for the safety-related portion of the system and Regulatory Position C.2 for the nonsafety-related portions of the system.

In DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 10.4.7.2.2, states that the CST, located external to the turbine generator building (TGB), includes tanks, piping (including the piping connections to the hotwell), valves, and tank-level instrumentation.

The staff reviewed the information on the CSF in the DCA and found that the DCA did not contain a description of the CSF system or a piping and instrumentation drawing. The DCA addressed only the CST, which it identified as being located outside the TGB and connected to the condenser hotwell. Because the CSF system has no safety-related portions, only Regulatory Position C.1.i of RG 1.29 is applicable. Based on the location of the CST and other CSF system components (TGB and in the yard), the staff found that the failure of portions of the CSF in the TGB would not have an adverse impact on SSCs important to safety. However, DCA Part 2, Tier 2, Table 3.2-1, 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.

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. DCA Part 2, Tier 2,

Section 3.4.1.4, “Flooding Outside of the Reactor 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 DCA Part 2, Tier 2, Section 9.2.6; the plant layout as described in Chapter 1 of DCA Part 2, Tier 2; and the information on flooding outside the RXB and CRB in DCA Part 2, Tier 2, Section 3.4.1.4. Because the CSTs are nonseismic, a seismic event could cause the failure of as many as 12 CSTs, which would have a combined volume of approximately 120,000 gallons of water. The staff was unable to verify the proximity of the CSTs to the CRB and RXB based on information in the DCA; however, the staff did find that COL Item 3.4-4 directs COL applicants that reference the NuScale 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 DCA Part 2, Tier 2, 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 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 DCA Part 2, Tier 2, 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. 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, acceptance for meeting the relevant aspects of GDC 60 is based on meeting the guidance in RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.”

DCA Part 2, Tier 2, Section 9.2.6, states that, in accordance with RG 1.143, CST instrumentation includes a high-level alarm that provides visual or audible indication locally and in the main control room (MCR). Also, a low-level alarm is provided for the CSF. The alarms give an early indication of a potential tank overfill or significant tank leak.

The staff reviewed DCA Part 1, Tier 2, 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 hotwell 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 as 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. DCA Part 2, Tier 2, Table 10.4.19, indicates that both the CST

level-indicating transmitter and the CST inlet flow transmitter are displayed both locally and in the MCR. 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 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 found 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 that provide visual or audible indication locally and in the control room, giving early indication of the potential for a tank overfill or a substantial leak from the tank. Furthermore, the applicant pointed out that COL Item 12.3-7 directs the COL applicant to develop processes and programs to demonstrate compliance with 10 CFR 20.1406 and the guidance in RG 4.21, and that the processes and programs will include inspection of buried piping in accordance with RG 4.21, Appendix A, paragraph 1.c.

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 DCA to be in compliance with 10 CFR 20.1406.

9.2.6.5 Initial Test Program

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

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 DCA Part 2, Tier 2, 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 items that should be in DCA Part 2, Tier 2, Table 1.8-2.

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 or after

a DBE. Serviced loads for the SCWS include equipment in the RXB, central utility building, north and south TGB, and auxiliary boiler building.

9.2.7.2 Summary of Application

DCA Part 2, Tier 1: The applicant provided no DCA Part 2, Tier 1, information for the SCWS.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 9.2.7, "Site Cooling Water System," describes the SCWS in detail. DCA Part 2, Tier 2, Table 9.2.7-1, "Site Cooling Water System Equipment Design Data," gives the system design parameters, and DCA Part 2, Tier 2, Figure 9.2.7-1, "Site Cooling Water System Diagram," shows the system diagram.

ITAAC: DCA Part 2, Tier 1, does not provide any ITAAC for the SCWS.

Technical Specifications: No GTS are applicable to the SCWS.

Technical Reports: No technical reports are associated with the SCWS.

9.2.7.3 Regulatory Basis

Although DCA Part 2, Tier 2, 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 relevant regulatory requirements for this area of review, as summarized below, 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 unit 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 design certifications 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

- 10 CFR 52.47(a)(24), which requires that a DCA contain a representative conceptual design for those portions of the plant for which the application does not seek certification, to aid the NRC in its review of the FSAR, and to permit assessment of the adequacy of the interface requirements in 10 CFR 52.47(a)(25)

- 10 CFR 52.47(a)(25), which requires that a DCA contain interface requirements to be met by those portions of the plant for which the application does not seek certification; these requirements must be sufficiently detailed to allow completion of the FSAR

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, north and south TGB, and auxiliary boiler building. Specifically, the SCWS provides cooling water to the condenser air removal system, chilled water system (CHWS), RCCWS, RPCS, SFPCS, auxiliary boiler blowdown cooler, PSS chillers, condensate and feedwater sample coolers, main steam sample coolers, turbine generator heat exchangers, lube oil and governor, and instrument air compressors and coolers.

The major components of the SCWS include three 50-percent-capacity SCWS pumps and associated piping, three cells of mechanical draft cooling towers, and associated basins and traveling screens. DCA Part 2, Tier 2, Section 9.2.7, describes the SCWS, and DCA Part 2, Tier 2, Figure 9.2.7-1, shows the system diagram.

DCA Part 2, Tier 2, Section 1.2.1, states that details associated with the location and orientation of the cooling towers, as well as equipment design and operation, are site specific. The cooling towers are designated as conceptual design information (CDI) in DCA Part 2, Tier 2, Figure 1.2-2. The applicant stated that the detailed design of the cooling tower basin, including its capacity and its ability to operate for periods without makeup water, is site specific and therefore is not included in the DCA and not intended to be part of the certified design. Moreover, the applicant stated that only a portion of the SCWS (i.e., the design and operation of the associated cooling towers) is CDI.

Based on the above, the staff will review in COL applications the portions of the SCWS that are designated as CDI.

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

DCA Part 2, Tier 2, 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.1.i. Based on its review of the NuScale DCA, the staff understands that the SCWS is a nonsafety-related, nonseismically designed system. DCA Part 2, Tier 2, Section 9.2.7.3, states that the SCWS is located sufficiently far from any seismic Category I or II structures or safety-related components. DCA Part 2, Tier 2, Section 9.2.7.1, states that the portions of the SCWS whose structural failure could adversely affect the function

of seismic Category I SSCs are seismic Category II. DCA Part 2, Tier 2, Table 3.2-1, Note 5, states the following:

> [W]here SSC (or portions thereof) as determined in the as-built plant which 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.

Note 5 applies to all seismic Category III piping installed within a seismic Category I structure.

Based on the above DCA statement and Note 5 to clarify seismic Category II SSCs to be consistent with RG 1.29, Position C.1.i, the staff finds that the design of the SCWS 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). DCA Part 2, Tier 2, Section 9.2.7, states that GDC 4 was considered in the design of the SCWS and monitoring for low system pressure and the presence of isolation valves at various points throughout the system are design features that allow any sudden large leak to be identified and isolated promptly. Additionally, the SCWS design includes an interlock for a slow opening of the pump discharge valve at pump start to minimize the effect of water hammer condition.

Based on the monitoring and isolation features and water hammer minimization design 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 finds that the design of the SCWS as described in the DCA 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. Further, the staff noted that DCA Part 2, Tier 2, Chapter 21, discusses the SCWS with respect to interactions between shared systems and operation of the 12 NPMs in the NuScale design. The staff finds the applicant's assessment for the SCWS in this regard complete and acceptable, and therefore, it 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. Radioactive effluent release, via the SCWS, is minimized by maintaining the process fluid at a higher pressure than potentially contaminated interfacing systems. The SCWS outlet of the RPCS, the SFPCS heat exchangers, and the RCCWS heat exchangers has radiation detectors, including sampling capability, 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. In addition, SCWS drains within the isolated boundary are directed to the liquid radwaste system.

Based on its review, the staff finds reasonable assurance of 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 at 10 CFR 20.1406 require, in part, that each design certification 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 SER contains additional information on the evaluation of the NuScale design with regard to the minimization of contamination.

Based on the SCWS design features described in DCA Part 2, Tier 2, 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 Combined License Information Items

Table 9.2.7-1 lists COL information item numbers and descriptions related to the SCWS, from DCA Part 2, Tier 2, Table 1.8-2.

Table 9.2.7-1 NuScale COL Information Item for Section 9.2.7

COL Item No.	Description	DCA Part 2, Tier 2, Section
9.2-4	A COL applicant that references the NuScale Power Plant design certification will provide details on the prevention of long-term corrosion and organic fouling in the site cooling water system.	9.2.7

The staff finds the above listing to be complete. Also, the list adequately describes programmatic actions necessary for the COL applicant to address. DCA Part 2, Tier 2, Table 1.8-2, does not need to include any additional COL information items for the SCWS.

9.2.7.8 Conclusion

The staff evaluated the SCWS for the NuScale 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 RG 1.29, Position C.1.i.

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 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

DCA Part 2, Tier 1: The applicant provided no DCA Part 2, Tier 1, information for the CHWS.

DCA Part 2, Tier 2: The applicant provided the design bases for and the detailed description of the CHWS in DCA Part 2, Tier 2, Section 9.2.8. DCA Part 2, Tier 2, Table 9.2.8-1, "Chilled Water System Equipment Design Data," provides CHWS design parameters, and DCA Part 2, Tier 2, Figure 9.2.8-1, "Chilled Water System Diagram," shows a basic flow diagram for the CHWS.

ITAAC: The applicant has not proposed any ITAAC related to the CHWS.

Technical Specifications: No TS are applicable to the CHWS.

Technical Reports: No technical reports are associated with the CHWS.

9.2.8.3 Regulatory Basis

SRP Section 9.2.7, Revision 0, "Chilled Water System," issued September 2015, 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. Note that as discussed in Section 9.2.5 of this report, the applicant has proposed PDC 44 as an alternative to GDC 44 to eliminate certain electrical power provisions from this requirement.

- 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(a), as it relates to how facility design and procedures for operation 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. DCA Part 2, Tier 2, Section 9.2.8, describes the CHWS, and DCA Part 2, Tier 2, Figure 9.2.7-1, shows the system diagram.

The primary CHWS consists of three 50-percent chillers (two operating, one standby) and three 50-percent pumps (two operating, one standby), 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, compressors, 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 SER.

The CRVS standby CHWS consists of one 100-percent air-cooled chiller and one 100-percent variable-speed pump. These standby components operate only when the backup power supply system (BPSS) is activated and the primary CHWS is unavailable to support the CRVS.

The major CHWS components are located in the central utility building and the top level of the CRB. The CRVS, RWBVS, and RBVS chilled water-cooling coils are located 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 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(a) 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. 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.1.i, for any nonsafety-related component whose failure during a seismic event could potentially affect the performance of safety-related SSCs.

DCA Part 2, Tier 2, Section 3.2, categorizes SSCs based on safety importance and other considerations. DCA Part 2, Tier 2, Table 3.2-1, provides the component safety classifications, seismic category, applicable codes and standards, and locations of the SSCs. All CHWS components are designated as 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 Section 3.2.1.2 and analyzed as described in 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 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, Position C.1.i), 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 DCA Part 2, Tier 2, Section 9.2.8.1, the CHWS is a shared system, but it does not interfere with the ability to operate or shut down a unit.

The CHWS provides cooling to HVAC systems and to radioactive waste systems but does not provide cooling to individual nuclear power modules. Based on the above DCA description, the staff finds that GDC 5 is satisfied.

9.2.8.4.5 PDC 44, "Cooling Water"

SRP Section 9.2.7 provides guidance for addressing GDC 44 requirements that are applicable to safety-related SSCs in the CHWS of an active PWR plant. As discussed in Section 9.2.5 of this report, the applicant has requested an exemption for certain electrical power provisions of GDC 44 and proposed an alternate PDC 44 to address cooling water requirements. In the NuScale 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 CHWS performs no important-to-safety cooling water functions, the provisions of PDC 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 the CHWS of an active PWR plant. The NuScale passive design classifies the entire CHWS as not safety related. Therefore, in DCA Part 2, Tier 2, Section 9.2.8.1, the applicant stated, in part, the following:

> Consistent with GDC 45, the CHWS is not safety-related and does not perform any safety-related functions during normal operations, anticipated operational occurrences, and accident conditions; therefore, no specific provisions are included in the design of the CHWS for periodic inspection. Consistent with GDC 46, the CHWS is not safety-related and therefore periodic pressure and functional testing of the system is not required…

Because the CHWS is not safety related and has no cooling function important to safety, the staff finds that the requirements of 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 design certifications describe how the facility design and procedures for operation will minimize contamination of the facility and the environment, as well as 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(a) as discussed in Section 12.3.

As described in DCA Part 2, Tier 2, Section 9.2.8.3, contamination is minimized by having CHWS pressure higher than the LRWS and GRWS pressures at the system interfaces to eliminate any driving force into the CHWS. The LRWS and GRWS are the only radioactive systems that interface with the CHWS.

In DCA Part 2, Tier 2, Table 1.9-3, the applicant stated, “The CHWS is at a higher pressure than the LRWS and GRWS where the systems interface, precluding introduction of radioactive contaminants into the CHWS.”

In DCA Part 2, Tier 2, Section 9.2.8.1, the applicant stated, in part, “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(a) as discussed in Section 12.3.”

Because the CHWS has an effective means to protect against contamination entering the system, the staff finds that the CHWS complies with the requirements of 10 CFR 20.1406(a).

9.2.8.5 Initial Test Program

In DCA Part 2, Tier 2, Table 14.2-8, “Chilled Water System Test #8,” the applicant described the system performance testing that will demonstrate proper operation of all CHWS components, including their associated instrumentation and control (I&C) circuitry. The DCA also lists a completed CHWS flow balance as a prerequisite for this system performance test.

The staff finds this performance testing of the CHWS acceptable because it can validate the functionality of the system, including verification of all required cooling water flows to meet the respective HVAC systems’ heat loads, as well as cooling loads for plant components in the GRWS and LRWS. The ITP is discussed further in Section 14.2 of this SER.

9.2.8.6 Technical Specifications

No GTS requirements are associated with the CHWS.

9.2.8.7 Combined License Information Items

The applicant did not propose any COL items for the CHWS, and the staff’s review of DCA Part 2, Tier 2, Section 9.2.8, did not identify a need for any additional COL item for this system.

9.2.8.8 Conclusion

The staff evaluated the CHWS for the NuScale 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.1.i.

9.2.9 Utility Water Systems

9.2.9.1 Introduction

This section describes the staff’s review of the NuScale utility water system (UWS). The UWS provides clarified water to the fire water tank, DWS, PWS, RBX, CRB, annex building, RWB, turbine building, central utility building, and other plant users. The UWS also supplies raw water that has not been clarified to the two circulating water system cooling tower basins and SCWS cooling tower basin for makeup water purposes. Raw water is the source of water for the UWS. The water supplied by the UWS does not provide cooling functions.

9.2.9.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, contains no entries for the UWS.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 9.2.9, provides information on the UWS. DCA Part 2, Tier 2, Figure 9.2.9-1, "Utility Water System Diagram," provides a functional arrangement of the UWS.

ITAAC: The applicant has not proposed any ITAAC related to the UWS.

Technical Specifications: There are no proposed GTS associated with the UWS.

Technical Reports: There are no technical reports associated with the UWS.

9.2.9.3 Regulatory Basis

The relevant regulatory requirements for this area of review and the associated acceptance criteria are given in SRP Section 9.2.4 and are summarized below. Review interfaces with other SRP sections are also indicated in SRP Section 9.2.4, Item I.

- 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, including 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 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

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. DCA Part 2, Tier 2, Table 3.2-1, classifies all UWS components as Seismic Classification III. The applicant provided a note at the end of Table 3.2-1, which states the following:

> Where SSC (or portions thereof) as determined in the as-built plant which 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.

DCA Part 2, Tier 2, Table 3.2-1, also provides the quality group classification of UWS components and equipment. The applicant stated that the UWS is in 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. Up to 12 NPMs share the UWS pumps and storage tank. The design and layout of the UWS include provisions to ensure that a failure of the UWS will not adversely affect the functional performance of safety-related systems. The UWS has no safe-shutdown functions that are shared between NPMs. 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. A monitor with sampling capability continuously monitors the UWS site discharge path for radiation. If predetermined system thresholds are exceeded, an alarm activates in the MCR and the waste management control room. 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 thresholds are exceeded, 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 design certification 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. DCA Part 2, Tier 2, Table 12.3-43, 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 for reliable service for the life of the plant.

The staff reviewed DCA Part 2, Tier 2, Sections 9.2.9 and 12.3, as related to the prevention and minimization of contamination. Because the NuScale DCA describes adequate measures for early 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 DCA complies with 10 CFR 20.1406.

9.2.9.5 Technical Specifications

There are no GTS requirements associated with the UWS. The system is not safety related and is not required for safe shutdown, and it does not meet a criterion in 10 CFR 50.36, "Technical Specifications," that would require a TS. Therefore, the staff finds this system acceptable.

9.2.9.6 Initial Test Program

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

9.2.9.7 Combined License Information Items

Table 9.2.9-1 lists COL information item numbers and descriptions related to the UWS.

Table 9.2.9-1 NuScale COL Information Item for Section 9.2.9

COL Item No.	Description	DCA Part 2, Tier 2, Section
9.2-5	A COL applicant that references the NuScale Power Plant design certification will identify the site-specific water source and provide a water treatment system that is capable of producing water that meets the plant water chemistry requirements.	9.2.9

9.2.9.8 Conclusion

Based on the review above, the staff concludes that the UWS for the NuScale design satisfies the relevant requirements for the UWS as described in the "Regulatory Basis" of this section.

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

9.3.1.1 Introduction

The compressed air system (CAS) consists of the instrument 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

DCA Part 2, Tier 1: The NuScale DCA has no Tier 1 information for the CAS.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 9.3.1, "Compressed Air System," provides the CAS description and operation, including design bases, instrumentation, and the inspection and testing program.

Technical Specifications: No proposed TS requirements are associated with the CAS.

Technical Reports: No technical reports are related to the CAS.

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 DCA Part 2, Tier 2, 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 SER. The applicant stated that the CAS is composed of the IAS, SAS, and 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 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 DCA Part 2, Tier 2, 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 unit 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 unit 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 determined that the licensee met the design requirements of GDC 1, 2, and 5 and provided the appropriate design features and preoperational tests to give

reasonable assurance that the design commitments are met and the as-built plant conditions will operate in accordance with the design certification.

9.3.1.5 Initial Test Program

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

9.3.1.6 Technical Specifications

No GTS requirements are associated with the CAS. Based on a graded approach commensurate with the safety significance of the SSCs, the staff agrees that GTS are not required for the CAS because this system does not meet the criteria for assigning an LCO based on the criteria in 10 CFR 50.36(c)(2). Therefore, the staff finds this acceptable.

9.3.1.7 Combined License Information Items

No COL items are associated with the CAS.

9.3.1.8 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 applicant has provided reasonable assurance that the CAS will operate as designed and its postaccident failures will not impact the performance of any safety-related equipment in the plant.

9.3.2 Process and Postaccident Sampling Systems

9.3.2.1 Introduction

The PSS allows the plant staff to obtain liquid and gaseous samples and determine their physical and chemical characteristics by measurement and analysis. Centralized and local facilities permit samples of primary and secondary coolant containment atmosphere to be taken. The system consists of the normal primary sampling system and the secondary sampling system.

9.3.2.2 Summary of Application

DCA Part 2, Tier 1: The applicant provided no Tier 1 information for this system.

DCA Part 2, Tier 2: The applicant described the PSS in DCA Part 2, Section 9.3.2, “Process Sampling System,” as a nonsafety system that consists of the primary sampling system, containment sampling system, secondary sampling system, and provisions for local grab sampling. The PSS has the ability to obtain a representative sample either through continuously analyzed or grab samples at various locations with normal system operating temperature and pressures from various locations. In DCA Part 2, Tables 9.3.2-1 through 9.3.2-4 contain detailed descriptions of all the sample points and the type of sampling done at each point.

ITAAC: No ITAAC are provided for this area of review.

Technical Specifications: No GTS requirements are associated with this area of review.

9.3.2.3 Regulatory Basis

SRP Section 9.3.2, Revision 3, "Process and Post-Accident Sampling Systems," issued March 2007, gives the following acceptance criteria for chemistry and chemical engineering issues in sampling systems:

- 10 CFR 20.1101(b), as it relates to using engineering controls to keep doses to workers and the public ALARA
- 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 maintaining the reliability of the RCPB by sampling for chemical species that affect the RCPB
- GDC 26, "Reactivity Control System Redundancy and Capability," as it relates to controlling reactivity 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 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

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 adhere to 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." The use and acceptability of these EPRI guidelines in the NuScale design are discussed in more detail in Section 9.3.4 and Chapter 10 of this SER.

9.3.2.4 Technical Evaluation

The staff has determined that the proposed PSS meets (1) the requirements of GDC 13 to monitor variables that can affect the fission process for normal operation and anticipated operational occurrences by sampling the reactor coolant, the refueling water, and the boric acid storage tank of the boron addition system (BAS) tank for boron concentrations, (2) the requirements of GDC 13 and 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 sampling the reactor coolant and the secondary coolant for chemical impurities that can affect the RCPB, (3) the requirements of GDC 26 to control the rate of reactivity changes by sampling the reactor coolant, the refueling water, and the boric acid storage tank for boron concentration, and (4) 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.

The staff has further determined that the proposed PSS meets (1) the requirements of 10 CFR 20.1101(b) to keep radiation exposures ALARA as 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) and of GDC 60 to control the release of radioactive materials to the environment by purging and draining sample streams back to the system of origin or to an appropriate LRWS and by providing either redundant isolation valves that fail in the closed position or passive flow restrictions in the sampling lines, and (2) the requirements of GDC 63 to detect conditions that may result in excessive radiation levels in fuel storage and the LRWS by sampling the SFP water and the GRWS and LRWS for radioactivity.

The staff also has determined that the proposed PSS meets the quality standards requirements of GDC 1 and the seismic requirements of GDC 2 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, and RG 1.29, Regulatory Position C.1.

However, the staff has found that the leakage control and detection requirements of 10 CFR 50.34(f)(2)(xxvi) and related clarifications of Item III.D.1.1 in NUREG-0737 are not met. 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.

Postaccident Sampling 10 CFR 50.34(f)(2)(viii) Exemption Request

The NRC staff notes that by letter dated January 31, 2019 (ADAMS Accession No. ML19031C975), as amended by the RAI 9682 response dated July 26, 2019 (ADAMS Accession No. ML19207A534), NuScale submitted a request for an exemption from 10 CFR 50.34(f)(2)(viii). 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 52.47(a) requires, in part, the following:

  The [design certification] 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 as a whole, and must include the following information:
  …

  (3) The design of the facility including:

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

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

- 10 CFR 52.7, "Specific Exemptions," 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 in order for the NRC to consider granting an exemption request.

Authorized by Law

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

No Undue Risk to Public Health and Safety

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

Consistent with Common Defense and Security

The requested exemption will not alter the design, function, or operation of any structures or plant equipment necessary to maintain a safe and secure plant status. In addition, the changes have no impact on plant security or safeguards. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the common defense and security are not impacted by this exemption.

Special Circumstances

Underlying Purpose of the Rule

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. In 10 CFR 50.34(f)(2)(viii), the NRC states that an applicant must provide a capability to promptly obtain and analyze samples from the reactor coolant system (RCS) and containment that may contain accident source term radioactive materials without radiation exposures to any

individual exceeding 50 millisievert (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 finding on the different aspects of 10 CFR 50.34(f)(2)(viii) for which NuScale has requested an exemption.

Radionuclides: To determine the potential need to sample for radionuclides, the staff evaluated NuScale'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 postaccident 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 postaccident sampling capability as a means of meeting the requirements of 10 CFR 50.47(b)(9). The staff is satisfied that a postaccident sampling capability is not necessary for sampling radionuclides for the NuScale design based on the use of radiation monitors to assess core damage.

Hydrogen in the Containment Atmosphere: The purpose of sampling hydrogen in the atmosphere is to ensure that hydrogen and oxygen concentrations do not support combustion that could challenge the containment. NuScale does not need to sample, as it has the capability to monitor hydrogen and oxygen concentrations as required by 10 CFR 50.44(c)(4) as discussed in Section 6.2.5 of this SER.

Dissolved Gases: Sampling for dissolved gases has generally been required to ensure that natural recirculation is not inhibited. This is not necessary for NuScale because of the design-specific considerations of the NuScale ECCS, as documented in the staff's evaluation of the exemption for high-point vents in Section 5.4.5 of this SER.

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 design limits chloride sources by design and operation, such as containment cleanliness requirements and minimal use of chlorinated cable insulation. NuScale also limits chloride by monitoring and control of reactor water chemistry based on industry guidelines contained in EPRI's PWR Primary Water Chemistry Guidelines. The staff finds it acceptable for NuScale to not perform postaccident chloride sampling because of the minimal use of chlorinated cable insulation and the monitoring of chlorine concentration using the EPRI PWR Primary Water Chemistry Guidelines during normal operation.

Boron Concentrations: The purpose of sampling 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 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 temperature. The transient and accident analyses described in DCD Part 2,

Tier 2, Chapter 15, does not rely on the measurement of RCS boron concentration and is 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 agrees with the applicant that a postaccident boron sample is not necessary for the NuScale design. The staff notes that the applicant requested an exemption to GDC 27, "Combined Reactivity Control Systems Capability," and proposed PDC 27. The staff utilized the review criteria in SECY-18-0099, "NuScale Power Exemption Request from 10 CFR Part 50, Appendix A, General Design Criterion 27, 'Combined Reactivity Control Systems Capability,'" dated October 9, 2018, and documents its evaluation of the exemption request and proposed PDC in Section 15.0.6 of this SER.

The staff notes that the exemption request 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 DCA Part 2, Tier 2, 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 design certification review but are important to capture in the development of operating procedures. The applicant included COL item 13.5-2 in DCA Part 2, Tier 2, Section 13.5.2, "Operating and Maintenance Procedures," for development of operating procedures at a future licensing stage.

Based on the above, the staff finds that application of the regulation in the 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) are met.

The applicant stated in its DCA Part 7 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) is not necessary for the exemption to be granted. Because the staff finds that special circumstances are present in accordance with 10 CFR 50.12(a)(2)(ii), the staff makes no finding regarding the presence of special circumstances described in 10 CFR 50.12(a)(2)(iv).

Conclusion

Based on the evaluation above, the staff finds that the applicant's design meets the requirements for an exemption under 10 CFR 50.12(a). The exemption is authorized by law, will not present an undue risk to the public health and safety, is consistent with the common defense and security, and meets the special circumstances requirement of 10 CFR 50.12(a)(2)(ii). Therefore, the staff approves granting NuScale's proposed exemption from the requirements of 10 CFR 50.34(f)(2)(viii).

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 will be exempt from 10 CFR 50.34(f)(2)(viii) and the applicant has removed all information from the FSAR indicating that postaccident 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.

Radiation Protection Evaluation

The radiation protection staff reviewed DCA Part 2, Tier 2, Section 9.3.2, and supporting DCA Part 2, Tier 2, 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 DCA Part 2, Tier 2, Section 9.3.2, 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 the vent hood 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 located in the RXB.

DCA Part 2, Tier 2, 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, which includes grab samples. 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. The monitoring of exposure to hazardous chemicals will be performed as directed by the plant's Occupational Safety and Health Administration required Chemical Hygiene Plan.

In addition, DCA Part 2, Tier 2, 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 steamlines and the radiation monitors on the condenser air removal 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 as a means 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 Table 9.3.2-4 of DCA Part 2, Tier 2, Section 9.3.2. These include local sample points for the LRMS, 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.

DCA Part 2, Tier 2, Section 9.3.2, provides several examples of features that are included in the NuScale 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, which are drained to the RWDS or the balance-of-plant drain system (BPDS), depending on the location of the sample sink

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, 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 provides 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 DCA Part 2, Tier 2, Tables 11.5-2 and 11.5-3.

Postaccident Conditions

As discussed above, NuScale requested an exemption from the postaccident sampling requirements of 10 CFR 50.34(f)(2)(viii). Since postaccident sampling will not occur in the NuScale design, the staff did not review the radiation dose consequences of collecting or analyzing postaccident samples. However, the NuScale design includes postaccident hydrogen and oxygen monitoring, which includes use of the PSS, the CES, and the containment flood and drain system. These systems are designed to conduct hydrogen and oxygen monitoring in accordance with 10 CFR 50.44(c)(4). The postaccident hydrogen and oxygen monitoring function are discussed in Chapter 6 of this SER. Consistent with COL Item 9.3-1, and as required by 10 CFR 50.34(f)(2)(xxvi), the COL applicant will submit a leakage control program, including an ITP, associated with controlling leakage from these hydrogen and oxygen monitoring systems. As discussed in Chapter 12 of this SER, the COL applicant is to provide assurance that postaccident leakage from these systems does not result in the total MCR dose exceeding the dose criteria (i.e., 50 mSv (5 rem)) for the surrogate event with significant core damage or the application must include design features in accordance with 10 CFR 50.34(f)(2)(xxvi) and 10 CFR 50.34(f)(2)(xxviii). In addition, the COL applicant will also provide information to verify, as appropriate, that postaccident leakage from these systems does not result in the total dose for the surrogate event with significant core damage exceeding the offsite dose criteria, as required by 10 CFR 52.47(a)(2)(iv). The radiation shielding design and compliance with 10 CFR 50.34(f)(2)(vii) for actions necessary to establish hydrogen and oxygen monitoring are discussed in Chapter 12 of this SER. Consideration of the offsite and MCR dose consequences due to leakage from the systems used for hydrogen and oxygen monitoring are also discussed in Chapter 15 of this SER. Survivability of the equipment associated with

hydrogen and oxygen monitoring and any other related equipment survivability functions are discussed in Chapter 19 of this SER.

9.3.2.5 Initial Test Program

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

9.3.2.6 Inspections, Tests, Analyses, and Acceptance Criteria

The staff evaluates ITAAC in Section 14.3 of this SER.

9.3.2.7 Combined License Information Items

Table 9.3.2-1 lists relevant COL information item numbers and descriptions from DCA Part 2, Tier 2.

Table 9.3.2-1 NuScale COL Information Items

COL Item No.	Description	DCA Part 2, Tier 2, Section
9.3-1	A COL applicant that references the NuScale Power Plant design certification will submit a leakage control program for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident (including systems and components used in post-accident hydrogen and oxygen monitoring of the containment atmosphere). 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.	9.3.2

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 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). The staff finds COL Item 9.3-1 acceptable because it addresses as-built plant information that the design certification cannot provide.

9.3.2.8 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 concludes that the sampling locations are consistent with SRP Section 9.3.2, RG 1.21, and the EPRI Guidelines; hence, they satisfy the requirements of 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 plant sampling system during normal operation is acceptable. As discussed above, the applicant has been granted an exemption from postaccident sampling under 10 CFR 50.34(f)(2)(viii).

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 EFDS 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 liquid radioactive waste system (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 reactor building (RXB) or radwaste building (RWB.)

9.3.3.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Table 3.0-1, "Shared Systems Subject to Inspections, Tests, Analyses, and Acceptance Criteria," provides the maximum number of NPMs shared by each BPDS.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 9.3.3, "Equipment and Floor Drain Systems," provides a complete description of the EFDS. Information provided includes the EFDS design bases, system and component descriptions, monitoring instrumentation, and details about the EFDS operation.

ITAAC: The applicant gave the ITAAC associated with DCA Part 2, Section 9.3.3, in DCA Part 2, Tier 1, Section 3.17, "Radiation Monitoring—NuScale Power Modules 1–6," and DCA Part 2, Tier 1, Section 3.18, "Radiation Monitoring—NuScale Power Modules 7–12." These ITAAC are evaluated in Section 14.3 of this SER.

Initial Test Program: DCA Part 2, Tier 2, Table 14.2-24, "Balance of Plant Drain System Test #24," describes the preoperational tests related to the BPDS that are being evaluated as part of the design certification review. The ITP is evaluated in Section 14.2 of this SER.

Technical Specifications: There are no proposed GTS associated with the EFDS.

Technical Reports: There are no technical reports associated with the EFDS.

9.3.3.3 Regulatory Basis

The relevant regulatory requirements for this area of review and the associated acceptance criteria are given in SRP Section 9.3.3, "Equipment and Floor Drainage System," and are summarized below. Review interfaces with other SRP sections are also indicated in SRP Section 9.3.3, item I.

- 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 unit 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 EFDS:

- 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 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 EFDS design for compliance with the regulatory basis provided above in Section 9.3.3.3 of this SER. The results of the staff's review are provided below.

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

DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, 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 0.3 m (1 ft) below grade. Therefore, the staff's evaluation of GDC 2 in this case is based on the guidance in Regulatory Position C.1.i of RG 1.29, which specifies that failure of systems that are not safety related should not adversely affect safety-related systems. DCA Part 2, Tier 2, Section 9.3.3.1, states that portions of the RWDS and BPDS system that are in proximity to seismic Category I SSCs are designed to seismic Category II standards. DCA Part 2, Table 3.2-1, classifies all RWDS and BPDS components as seismic Category III. The applicant provided a note at the end of Table 3.2-1, which states the following:

> Where SSC (or portions thereof) as determined in the as-built plant which 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"

DCA Part 2, Tier 2, Section 9.3.3.3, states that the design of the RWDS and BPDS ensures that safety-related equipment functions are not impacted by undue water accumulations within the plant. The internal flood analysis provided in 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, the staff finds that the design of the RWDS and BPDS meets GDC 4.

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

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

The applicant stated that the NPMs share the 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, 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"

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. DCA Part 2, Tier 2, Section 9.3.3.2.1, indicates that the RWDS and BPDS are designed to include surge capacity to support other activities, such as runoff from firefighting activities. Both systems are designed with radiation monitoring, which includes functions to automatically terminate tank discharges. 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 main control room (MCR) and waste management control room. If a monitor detects a high-radiation condition, an alarm sounds in the MCR and waste management control room. The RWDS is designed to receive radiologically contaminated liquids and normally noncontaminated liquids, including from the reactor component cooling water system (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"

In part, 10 CFR 20.1406 requires that each design certification 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. In DCA Part 2, Tier 2, Tables 12.3-15 and 12.3-37 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 double-walled pipes with leak detection in underground piping, use of components designed for service for the life of the plant, and the minimal use of embedded piping by the RWDS and BPDS. Because the NuScale design provides adequate measures for early 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

The preoperational test related to the BPDS for design certification is BPDS Test #24. This test is performed in accordance with DCA Part 2, Tier 2, Table 14.2-24. Section 14.2 of this SER provides the staff's evaluation of the ITP.

9.3.3.6 Technical Specifications

There are no GTS requirements associated with the EFDS. The system is not safety related, not required for safe shutdown, and does not meet a criterion in 10 CFR 50.36 that would require a TS; therefore, the staff finds this acceptable.

9.3.3.7 Combined License Information Items

There are no COL information items associated with the EFDS.

9.3.3.8 Conclusion

Based on the review above, the staff finds that the EFDS for the NuScale design satisfies the relevant requirements for the EFDS as described above in the Regulatory Basis section.

9.3.4 Chemical and Volume Control System

9.3.4.1 Introduction

The NuScale 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.

9.3.4.2 Summary of Application

DCA Part 2, Tier 1: The Tier 1 information associated with this section appears in DCA Part 2, Tier 1, Section 2.2, "Chemical and Volume Control System."

DCA Part 2, Tier 2: The applicant provided a Tier 2 system description in DCA Part 2, Tier 2, Section 9.3.4, "Chemical and Volume Control System," summarized in Section 9.3.4.1 of this SER.

ITAAC: The applicant provided the ITAAC associated with DCA Part 2, Tier 2, Section 9.3.4, in DCA Part 2, Tier 1, Section 2.2.2, "Inspections, Tests, Analyses, and Acceptance Criteria." These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: NuScale DCA Part 4, Volume 1, "Generic Technical Specifications," gives the TS associated with DCA Part 2, Tier 2, Section 9.3.4. The GTS include Section 3.1.9, "Boron Dilution Control," and Section 3.4.6, "Chemical and Volume Control System Isolation Valves."

Technical Reports: No technical reports are associated with this section of the applicant's DCA Part 2.

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, "Design Bases for Protection against Natural Phenomena," as it relates to structures housing the facility and the system itself being capable of withstanding the effects of earthquakes

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

- GDC 14, "Reactor Coolant Pressure Boundary," 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, "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

- 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, as described in Section 1.0 of this SER, to evaluate the CVCS. DCA Part 2, Tier 2, Table 3.2-1, details the safety-significance categorization for the CVCS. Sections 3.2.2 and 17.4 of this SER provide the basis for acceptability of the CVCS safety-significance categorization.

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 RG 1.26.

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, which the applicant designated as safety related because these lines and components maintain the RCPB, are all designed to Quality Group A. The staff confirmed that, according to DCA Part 2, Tier 1, Section 2.1, 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 RCPB meet the requirements for Class 1 components in the ASME Code, Section III.

The staff also 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 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, which is defined in RG 1.26. The staff also confirmed that the CVCS injection check valve, the CVCS discharge isolation valve, the CVCS pressurizer spray check valve, and the RPV high-point degasification isolation valve, along with the associated piping and reactor module removable spool pieces and their associated drain valves, are all categorized as Quality Group C. The staff further confirmed that, according to DCA Part 2, Tier 1, Section 2.2, "Chemical and Volume Control System," these Quality Group C SSCs are appropriately designed to the applicable ASME Code, Section III class, which is Class 3. 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 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 DCA Part 2, Tier 2, Sections 9.3.4 and 3.2. 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 isolation valve, the CVCS pressurizer spray check valve, and the RPV high-point degasification isolation valve, along with the associated reactor module removable spool pieces and their associated drain valves, are all designed to seismic Category I standards. The staff also confirmed that the piping from the containment isolation valves to the reactor module removable spool pieces is 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 and distribution assembly, including the excess flow valve, and the pressure instrumentation on the hydrogen bottle, are designed to seismic Category II standards, which the staff finds acceptable because having such a seismic design for the hydrogen injection equipment will keep the equipment 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 RXB is designed to seismic Category I, which protects the CVCS from external phenomena.

The staff finds that the NuScale 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 5, "Sharing of Structures, Systems, and Components"

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

9.3.4.4.4 GDC 14, "Reactor Coolant Pressure Boundary"

The staff reviewed the NuScale 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 their 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 CVCS does not have 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 module bypass 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 module bypass 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 CVCS because the CVCS has no tank that could fail via wall-inward buckling during normal operation.

The staff reviewed DCA Part 2, Tier 2, Section 9.3.4, to determine whether the CVCS's purification components have adequate overtemperature and overpressure protection. The staff noted that the NuScale 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, in DCA Part 2, Tier 2, Section 9.3.4.2 and Section 9.3.4.5 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 and could indicate reduced performance of these components. This instrumentation is also shown in DCA Part 2, Tier 2, 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. DCA Part 2, Tier 2, Section 9.3.4.2.1, states that all BAS components are located in the RXB, which is maintained above 10 °C (50 °F). Therefore, the staff finds that the CVCS and BAS are adequately protected against boron precipitation because both systems are maintained above the solubility limit of 7.2 °C (45 °F) corresponding to the specified maximum boron concentration of 5600 mg/ml (5600 ppm).

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 DCA Part 2, Tier 2, 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.

Because the CVCS does not continually maintain the water chemistry of the stagnant water within this isolated ECCS line the staff further noted that contaminants could accumulate in this isolated line and ultimately affect the operation of the ECCS valve. In a letter dated February 12, 2018 (ADAMS Accession No. ML18043A162), the applicant confirmed that the CVCS could adequately maintain the appropriate water chemistry in the ECCS lines over the course of plant operation. The staff determined that the justification provided by the applicant was acceptable because the applicant confirmed that the water chemistry in the stagnated water lines will be appropriately maintained in accordance with a site-specific water chemistry program. Specifically, the applicant stated that 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 the applicant's DCA also contains COL Item 5.2-4, 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 CVCS filtration and purification components ensures that particulates do not accumulate in the stagnated water lines. The staff finds that the COL 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 DCA Part 2, Tier 2, Section 9.3.4, using the guidance in DSRS Section 9.3.4 on the materials and chemistry aspect of the CVCS. The DSRS 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.

SRP Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," also covers reactor coolant chemistry, where the primary concern is the compatibility of RCS materials with the coolant. Section 5.2.3 of this SER discusses this subject in greater detail, establishing that materials are compatible with primary system coolant, and the RCS water does not negatively affect the performance of these materials, provided that adequate chemistry control occurs. For that reason, this section and Section 5.2.3 of this SER cover primary water chemistry control.

DCA Part 2, Tier 2, Table 5.2-5, "Reactor Coolant Water Chemistry Controls," shows limits for primary water chemistry parameters in the RCS. DCA Part 2, Tier 2, Section 5.2.3.1, states that the primary water chemistry program is based on the EPRI Guidelines. Additionally, the applicant stated that it will follow both the "mandatory" and "shall" elements of the EPRI Guidelines. The NRC staff finds the parameters monitored in the primary water chemistry program to be acceptable because the monitored parameters will conform to the latest version of the EPRI Guidelines, and the primary water chemistry program specifically includes the "mandatory" and "shall" elements of the EPRI Guidelines. Additionally, the staff reviewed the proposed normal operating ranges for certain primary water chemistry parameters provided in Table 5.2-5 and found them to be within the recommended limits in the EPRI Guidelines.

DCA Part 2, Tier 2, Section 5.2.3.2.1, states the following:

> When measured water chemistry parameters are outside the specified range, corrective actions are taken to bring the parameter back within the acceptable range and within the time period specified in the EPRI water chemistry guidelines.

DCA Part 2, Tier 2, Section 9.3.4.3, states that "Action Level 2 and Action Level 1 conditions require correction within 24 hours and 7 days, respectively." Additionally, this section of DCA Part 2 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 the applicant. The staff finds the proposed corrective actions and associated timeframes acceptable because they will be in accordance with the EPRI Guidelines.

DCA Part 2, Tier 2, Section 5.2.3.1, states that the frequency for sampling primary water chemistry parameters is based on the EPRI Guidelines, as well as plant conditions. The staff finds it acceptable that the EPRI Guidelines will be used as the basis for selecting the frequency for sampling parameters in the primary water, as this provides reasonable assurance that the primary water chemistry parameters will be monitored at the appropriate frequency. Additionally, the staff reviewed the applicant's information that lists the parameters to be sampled, including the sampling frequency. The staff determined that the proposed samples and frequency are within the recommendations of the EPRI Guidelines and are therefore acceptable.

DCA Part 2, Tier 2, Section 9.3.4.2.3, "System Operation," describes the control of primary water pH. The applicant proposed to use lithium hydroxide additions, removing lithium ions with the cation bed ion exchanger in order to raise and lower pH. DCA Part 2, Tier 2, Section 5.2.3.1, COL Item 5.2-4, states that a COL applicant will implement a strategic water chemistry plan that follows the latest version of the EPRI Guidelines. Additionally, the applicant stated that the strategic water chemistry plan will implement, among other items, the "shall" requirements of the EPRI Guidelines. Included in the "shall" requirements of the EPRI Guidelines are certain requirements and recommendations on how to control primary water pH. The use of lithium hydroxide is common for controlling pH in operating commercial nuclear power plants. Operating experience has shown lithium hydroxide to be effective in controlling pH; therefore, the staff determined that the addition and removal of lithium is an acceptable method to control pH. The staff has also determined that the proposed plant-specific pH control program is appropriate because it will be developed in accordance with the EPRI Guidelines, and COL Item 5.2-4 directs a COL applicant to develop the program.

DCA Part 2, Tier 2, Section 9.3.4.2.2, describes the CVCS ion exchangers and other purification equipment. The CVCS contains four ion exchangers (two mixed bed, one auxiliary, and one cation bed). The mixed-bed ion exchangers are redundant (i.e., only one is needed in service to accomplish primary water purification) and can be operated in parallel but not in series. Each of the ion exchangers holds a minimum of 0.249 m^3 (8.8 cubic feet) of resin and can operate with a flow rate of 167 L/min (44 gpm) through the ion exchanger. DCA Part 2, Tier 2, Section 9.3.4.3, states that an 83-L/min (22-gpm) flow rate can correct primary water chemistry from Action Level 2 concentrations back to below the Action Level 2 threshold within 24 hours. Based on the RCS volume and flow rates, an 83-L/min (22-gpm) flow rate would be able to circulate the volume of the RCS through the ion exchangers in less than 24 hours. The staff reviewed the design of the ion exchangers and purification equipment to determine whether they are capable of maintaining primary water chemistry within the specified parameters. The staff determined

that there is reasonable assurance that the ion exchangers and purification equipment will be capable of maintaining primary water chemistry because there are redundant trains of ion exchangers and because the ion exchangers have an appropriate flow rate to maintain primary water chemistry.

In addition, the applicant stated that during refueling operations, the NPM will be transported to the RFP for disassembly and refueling. During this time, the NPM will be exposed to the pool water. The applicant stated that the PCUS will ensure that the impurity levels in the pool water meet the impurity levels specified for the RCS cold shutdown in the EPRI Guidelines. DCA Part 2, Tier 2, Table 9.1.3-2, shows the limits for certain parameters during refueling outages. These limits meet the limits specified for the RCS cold shutdown in the EPRI Guidelines.

At plant startup, hydrazine is injected into the RCS to scavenge oxygen. However, during normal operation, free oxygen is controlled by supplying hydrogen gas, which reacts with oxygen to form water. This reaction is catalyzed by radiolysis products, which are abundant in the coolant as it passes through the core region. It is important to note that the reverse reaction (i.e., the separation of water into hydrogen (H_2) and oxygen (O_2) molecules) is also catalyzed by radiation. Thus, the success of this method in reducing free oxygen depends on providing an excess amount of hydrogen. Hydrogen is supplied to the coolant through a compressed hydrogen source. By maintaining hydrogen concentrations as specified in DCA Part 2, Tier 2, Table 5.2-5, the applicant is consistent with the EPRI Guidelines, Revision 7. Therefore, there is reasonable assurance that the concentration of oxygen in the RCS will be sufficiently low to protect materials from oxidative degradation.

Boric acid is initially present in the RCS for reactivity control, and its concentration can be gradually altered over the course of an operating cycle. The CVCS monitors and maintains the inventory of boric acid. To increase concentration, the BAS part of the CVCS provides borated water after mixing demineralized water with dry boric acid to achieve the desired concentration. To decrease concentration, demineralized water is added to the RCS to dilute the concentration of boron. In addition, toward the end of the fuel cycle, borated water can be used to lower the boron concentration in the RCS through ion exchange. Borated water to be discharged will be sent to the LRWS.

Based on the NRC staff's review of the primary water chemistry and the ability of the CVCS to maintain primary water purity, the staff has reasonable assurance that GDC 14 will be met with respect to reducing the probability of abnormal leakage, rapidly propagating failure, or gross rupture of the RCPB. The staff finds COL Item 5.2-4, which directs a COL applicant to develop a strategic water chemistry plan that implements the "mandatory" and "shall" requirements of the EPRI Guidelines, acceptable because it is appropriate to develop this information as part of a site-specific application. Additionally, the staff has reasonable assurance that the CVCS equipment described in DCA Part 2 will be able to maintain primary water purity and be adequate to correct off-normal primary water chemistry conditions within the specified timeframe.

The staff finds that the NuScale 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.

9.3.4.4.5 GDC 29, "Protection against Anticipated Operational Occurrences"

The staff reviewed the NuScale 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 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 in the event of this anticipated operational occurrence. The staff further notes that the NuScale 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 the applicant designed the CVCS with redundant demineralized water isolation valves that will isolate the dilution source, and because the CVCS is not relied on for mitigating the effects of any anticipated operational occurrence or accident.

9.3.4.4.6 Exemption from GDC 33, "Reactor Coolant Makeup"

The applicant requested an exemption from GDC 33, which requires a system to supply reactor coolant makeup for protection against small breaks in the RCPB. The applicant stated that the NuScale Power Plant design does not require makeup to protect against breaks in the RCPB. The applicant stated that the RPV and CNV designs, in conjunction with the passive design and operation of the ECCS, ensure that the core is not uncovered, and adequate core cooling is maintained in the event of a break in the RCPB.

Regulatory Requirements

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

> The [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:
> …
>
> (3) The design of the facility including:
>
> (i) The principal design criteria for the facility. Appendix A to 10 CFR Part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for watercooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units;
>
> (ii) The design bases and the relation of the design bases to the principal design criteria…

- 10 CFR 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 finds that the exemption is authorized by law.

No Undue Risk to Public Health and Safety

This exemption does not affect the performance or reliability of power operations, does not impact the consequences of any DBE, and does not create new accident precursors. The NuScale Power Plant 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 finds that the exemption poses no undue risk to public health and safety.

Consistent with Common Defense and Security

The proposed exemption will not alter the design, function, or operation of any structures or plant equipment necessary to maintain a safe and secure plant status. In addition, the changes have no impact on plant security or safeguards. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the common defense and security are not impacted by this 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 CVCS design to determine the technical adequacy of the applicant's request for an exemption from the requirements of GDC 33. DCA Part 2, Tier 2, Section 9.3.4, and DCA Part 2, Tier 1, Section 2.2, state that the main function of the CVCS is to maintain adequate reactor coolant inventory during normal operation 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 finds that the NuScale design meets 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, DCA Part 2, Tier 2, Section 6.3, and DCA Part 2, Tier 1, Section 2.1, state that the ECCS's main function is to provide core cooling during and after anticipated operational occurrences and postulated accidents. The staff also notes that the applicant's 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 meets the underlying purpose for off-normal operation in two ways. The first is the applicant relies on the safety-related ECCS to provide SAFDL protection by maintaining core inventory and coolability. The second is, 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 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 NuScale Power Plant has an alternative means of maintaining reactor coolant inventory and coolability during off-normal transients. Therefore, based on the evaluation above, the staff finds that the NuScale CVCS meets 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 ECCS protects against the detrimental effects of off-normal operational coolant loss.

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

Conclusion

The staff finds that the applicant's design meets the underlying purpose of GDC 33, even though the applicant's design does not match the language of GDC 33 verbatim. The staff concludes that the technical basis for the applicant's request for exemption from GDC 33 is acceptable because it meets the conditions for an exemption in 10 CFR 50.12(a). Therefore, the staff approves granting NuScale's proposed exemption from the requirements of GDC 33.

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

The staff reviewed the NuScale 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 DCA Part 2, including drawings and diagrams, that the CVCS is equipped with appropriate ventlines and drainlines 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 SER provides the staff's review of the equipment and floor drainage system, and Chapter 11 of this SER contains the staff's review of the radioactive waste management system.

The staff reviewed the NuScale CVCS to determine whether it includes appropriate provisions for leakage control and detection. The staff confirmed through the review of information in DCA Part 2, that the CVCS design includes leak detection instrumentation that continuously compares the mass flow rates in and out of the RCS. The leak detection instrumentation will initiate an alarm in the MCR upon reaching a high setpoint. When the threshold for the high-high setpoint is reached, the CVCS will automatically close its module isolation valves to quickly isolate the unidentified CVCS leak; isolating the leak would take longer if the design used the safety-related containment isolation valves to provide isolation. The staff finds this leakage detection and control scheme appropriate because it reasonably minimizes leakage

from the CVCS. Furthermore, the staff confirmed that the applicant has identified COL Item 9.3-1, which directs a COL applicant that references the NuScale Power Plant design certification 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 CVCS meets the requirements of GDC 60 because the CVCS is equipped with ventlines and drainlines 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 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, which specifies that a COL applicant that references the NuScale Power Plant design certification will submit a leakage control program, including an ITP, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The staff did not review or approve these programmatic aspects of 10 CFR 50.34(f)(2)(xxvi) in connection with the NuScale DCA. A COL applicant referencing the NuScale standard design will be required to demonstrate compliance with 10 CFR 50.34(f)(2)(xxvi) by providing a program to address the remaining aspects of this requirement.

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

The staff reviewed the NuScale 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, DCA Part 2 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 SER 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 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.

9.3.4.4.9 10 CFR 20.1406, "Minimization of Contamination"

In Sections 12.3 and 12.4 of this SER, the staff reviewed the NuScale 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.

9.3.4.5 Initial Test Program

The ITP tests related to the CVCS, such as "Chemical and Volume Control System Test #38," and "Primary and Secondary System Chemistry Test #79," described in DCA Part 2, Tier 2,

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 DCA Part 2, Tier 2, Table 1.8-2.

Table 9.3.4-1 NuScale COL Information Items for Section 9.3.4

COL Item No.	Description	DCA Part 2, Tier 2, Section
5.2-4	A COL applicant that references the NuScale Power Plant design certification will develop and implement a Strategic Water Chemistry Plan. The Strategic Water Chemistry Plan will provide the optimization strategy for maintaining primary coolant chemistry and provide the basis for requirements for sampling and analysis frequencies, and corrective actions for control of primary water chemistry consistent with the latest version of the EPRI Pressurized Water Reactor Primary Water Chemistry Guidelines.	5.2.3
9.3-1	A COL applicant that references the NuScale Power Plant design certification will submit a leakage control program for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident (including systems and components used in post-accident hydrogen and oxygen monitoring of the containment atmosphere). 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.	9.3.2

The NRC staff finds NuScale COL Item 5.2-4 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 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 items 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 and therefore not applicable to the NuScale DCA.

9.3.6 Containment Evacuation System and Containment Flooding and Drain System

9.3.6.1 Introduction

The CES and the containment flooding and drain system (CFDS) transfer liquids and gases between the CNV free volume and other plant systems.

9.3.6.2 Summary of Application

DCA Part 2, Tier 1: The CES performs a function verified by ITAAC that is not safety related; namely, the CES supports the RCS by providing RCS leak-detection monitoring capability.

DCA Part 2, Tier 1, Table 2.3-1, lists the following ITAAC, which are evaluated in Section 14.3 of this SER:

- The CES level instrumentation supports RCS leakage detection.
- The CES pressure instrumentation supports RCS leakage detection.

The RCS leakage detection is a function that is not safety related, which the staff evaluates in Section 5.2.5 of this SER.

DCA Part 2, Tier 1, Table 3.17-1, shows that the CFDS containment drain separator gaseous discharge to the RBVS heating ventilation and air conditioning system is monitored for radiation to close the CFDS containment drain separator gaseous discharge isolation valve during high radiation.

DCA Part 2, Tier 2: A dedicated CES, which is not safety related, supports each NPM. This 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 the gases to the CES condenser.

Condensate from the CES condenser drains to a sample vessel before draining to the RWDS. Samples of the noncondensable gases flow to the PSS for analysis. Section 9.3.2 of this SER presents the staff's evaluation of the PSS.

The CES operates from the MCR using the module control system that provides both automatic and operator control of key CES functions, including valve alignment, vacuum pump speed, and purge gas flow. The module control system provides the following:

- indication, alarms, and interlocks for CES flow, temperature, pressure, radioactivity level, humidity, and valve position

- alarms and required automatic actuation for off-normal conditions

The CES and CFDS are not safety-related systems and are not assumed to operate during or after any DBA. However, the CFDS can be used to add additional borated coolant inventory to the CNV to remove decay heat during a beyond-design-basis event.

The CES and CFDS off-normal operations include the following:

- high-radiation level in gases discharged from the CES condenser
- high-radiation level in gases discharged from the containment drain separator tank

- equipment failure affecting one or both CES vacuum pumps
- addition of coolant inventory into a CNV during a beyond-design-basis event

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.

The CFDS, which is not safety related, is used to flood a CNV with borated reactor pool water after shutdown in preparation for NPM refueling and to drain water back to the reactor pool in preparation for NPM startup. The CFDS can also be used to add water to a CNV during a beyond-design-basis event.

Multiple NPMs share the CFDS because the system is used only as needed to prepare and recover an NPM from conditions needed for refueling. There are two independent CFDS subsystems, each servicing up to six NPMs. Each CFDS includes two pumps that can be aligned to either flood or drain any of the six supported NPMs. Each subsystem 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.

The MCR controls the CFDS operation using the plant control system, which provides the following:

- automatic and operator control of key CFDS functions
- CFDS valve alignment, except the NPM isolation valves, controlled by the module control system
- CFDS alarms and interlocks
- indication of CFDS flow, temperature, pressure, containment drain separator tank level, gaseous discharge process radiation level, and valve position

The EDNS provides electrical power for the CFDS I&C.

Technical Specifications: LCO 3.4.3 states, “RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the [pressure and temperature limits report (PTLR)],” has actions required for the condition of “[c]ontainment flooding initiated while RCS temperature greater than allowed by [PTLR].”

LCO 3.4.7 states that two of the following RCS leakage detection instrumentation methods shall be OPERABLE:

1. Two CES condensate channels.
2. Two CES inlet pressure channels.
3. One CES gaseous radioactivity monitor channel.

Technical Reports: No technical reports are associated with this section.

9.3.6.3 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 34, “Residual Heat Removal,” as it relates to the system’s capability to remove core decay heat
- 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,” DSRS Section 5.4.7, “Decay Heat Removal (DHR) System,” and Branch Technical Position (BTP) 5-4, Revision 4, “Design Requirements of the Residual Heat Removal System,” issued March 2007, 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 and CFDS against the other requirements listed in DSRS, Section 9.3.6. The staff evaluates GDC 5 and GDC 34, which are not listed 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.

9.3.6.4 Technical Evaluation

The staff reviewed NuScale DCA Part 2, Tier 2, Section 9.3.6, “Containment Evacuation System and Containment Flooding and Drain System,” using guidance provided in NuScale DSRS Section 9.3.6.

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 and 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. The applicant identified the quality group and seismic category of CES and CFDS components in DCA Part 2, Tier 2, Table 3.2-1. The staff finds that the quality group identification is consistent with RG 1.26. All SSCs of the CES and CFDS are designed to Quality Group D and seismic Category D. 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 12 NPMs have independent CESs, and thus, the CES is not shared among the NPMs. There are two independent CFDS subsystems, each supporting up to six NPMs.

None of the CFDS SSCs is safety related. Sharing of the CFDS SSCs among the NPMs does not impair their ability to perform their 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.

9.3.6.4.3 GDC 34, "Residual Heat Removal"

GDC 34 states the following:

> Suitable redundancy...shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

BTP 5-4 states the following:

> The system(s) that can be used to take the reactor from normal operating conditions to cold shutdown shall satisfy the following functional requirements:
> A. The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems.

SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994, allows an alternative to the BTP 5-4 position for passive systems based on the quality group classifications and the following:

> After the passive RHR system or main steam system effected the initial shutdown, a non-safety-grade reactor shutdown cooling system will be available to bring the plant to cold shutdown conditions for inspection and repair. EPRI stated that "these non-safety systems are required to be highly reliable...and there is no single failure of these systems or their support systems which would result in inability to terminate use of the passive safety grade system and achieve cold shutdown if desired."

DCA Part 2, Tier 2, Section 5.4.3.1, states the following:

> Water Reactor Utility Requirements Document (Reference 5.4-3) was determined to be acceptable by the Nuclear Regulatory Commission as documented in SECY-94-084. Per SECY-94-084 and NUREG-1242, Volume 3, Part 2, transition of a passive plant from safe shutdown conditions to cold shutdown conditions may be reached using nonsafety-related systems. The nonsafety-related containment flood and drain system is used to flood the containment to allow passive long-term decay heat removal via convection and conduction to the reactor pool via the RCS, RPV shell, flooded containment, and CNV shell.

DCA Part 2, Tier 2, Figure 9.3.6-2, "Containment Flooding and Drain System Diagram," shows that a single CFDS line to the module has valves in series (CFDS containment isolation valves, CFDS module isolation valve, and the valve upstream of the six module isolation valves in parallel), and thus, a failure of any of those valves may make the CFDS inoperable. By letter dated December 14, 2017 (ADAMS Accession No. ML17348B522), the applicant stated the following:

> The NuScale Power proposed technical specifications describe five operating modes: Operations, Hot Shutdown, Safe Shutdown, Transition and Refueling as shown in Part 4, Volume 1, Table 1.1-1. Of these modes, none are directly analogous to the legacy operating mode of "cold shutdown." If an accident

> scenario occurs, the reactor module will be brought from Mode 1, "Operations," to Mode 3, "Safe Shutdown," utilizing either the decay heat removal system, emergency core cooling system, or normal non-safety means such as the feedwater system and condenser....
>
> If additional coolant is needed to cool the nuclear power module (NPM) from normal operating conditions to conditions equivalent to cold shutdown in a conventional plant (i.e., RCS temperature less than 200 degrees F [93.3 degrees C]), inventory can be added to the NPM through either containment flooding and drain system (CFDS) or the chemical volume and control system (CVCS), as described in FSAR Sections 9.3.4 and 9.3.6. The CFDS and CVCS are separate systems and share no components, allowing the function of adding inventory to the NPM while meeting the intent of the single failure proof criteria.

The staff agrees with the applicant's statement because the applicant does not define "cold shutdown" as a "mode" in its proposed TS, which the staff evaluated and found acceptable in Chapter 16 of this SER, and because cold shutdown can be achieved using the CFDS and the CVCS, which are separate systems that share no components.

The staff finds that the information provided in the application is consistent with DSRS Section 5.4.7, "Decay Heat Removal (DHR) System," and therefore, concludes that the requirements of GDC 34 are satisfied.

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

DCA Part 2, Tier 2, Section 9.3.6.2.3, 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. DCA Part 2, Tier 1, Table 3.17-1, notes that the CFDS containment drain separator gaseous discharge to the RBVS heating ventilation and air conditioning system is monitored for radiation to close the CFDS containment drain separator gaseous discharge isolation valve during high radiation. DCA Part 2, Tier 2, Figure 9.3.6-2, shows the radiation monitor on the drain separator tank discharge line. In the event of high radiation levels, DCA Part 2, Tier 2, Section 9.3.6.2.3, states that "[w]ith the discharge line isolated, the radioactive gases are left in the system to decay until the level is below the limit for release through the RBVS plant exhaust stack." 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.

DCA Part 2, Tier 2, Section 9.3.6.5, states the following:

> [t]he CES and the CFDS have indication and alarms associated with system critical parameters to alert operators in the MCR of potentially adverse conditions. The types of parameters monitored are presented under the General Description headings of Section 9.3.6.2.1 and Section 9.3.6.2.2.

Each system monitors process variables, including pressure, temperature, tank level, and radioactivity.

DCA Part 2, Tier 2, Section 9.3.6.2.3, states the following:

> High-radiation levels in the gases removed from the CNV actuate an alarm in the MCR. If the radiation level in the CES gaseous process flow exceeds a specified limit, or upon monitor failure, the discharge path is transferred from the RBVS to the GRWS and the following automated functions are initiated:
>
> - service air to the CES isolates.
> - service air to the CES vacuum pumps isolates.

DCA Part 2, Tier 2, Figure 9.3.6-1, shows only a single line coming from the service air system to the CES. Therefore, isolating service air to the CES would automatically isolate the CES vacuum pumps. By letter dated September 18, 2017 (ADAMS Accession No. ML17261B286), the applicant explained that there is only one isolation function of service air to the CES triggered by the high-radiation signal or failure of the radiation monitor. The staff finds the applicant's explanation acceptable.

The staff noted that the applicant requested an exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the CES. DCA Part 2, Tier 2, Section 9.3.6, states the following:

> The NuScale design supports exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the CES. The CES will have radioactivity monitors on the discharge path of the system; however, they will not trigger a safety-related containment isolation signal in the event of a high radioactivity signal. Rather, upon a high radioactive effluent signal on the gaseous discharge path, the system automatically diverts the gaseous effluent from the [reactor building ventilation system (RBVS)] to the [gaseous radioactive waste system (GRWS)] using nonsafety-related signals and equipment. Refer to Part 7, Chapter 13 for further details.

Section 6.2.4 of this SER describes the staff's disposition of this exemption.

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

The staff reviewed GTS related to the CES and CFDS as presented in DCA Part 4, "General Technical Specifications." The staff evaluations of LCO 3.4.3 and 3.4.7 are described in Sections 5.3.2 and 5.2.5 of this SER. The acceptable value for containment vacuum is evaluated in Section 6.2.1.1 of this SER. Based on these evaluations, the staff determined that GTS 3.4.3 and 3.4.7 are consistent with DSRS Section 9.3.6 and the criteria for assigning an LCO in 10 CFR 50.36(c)(2). Therefore, the staff finds that the applicant identified acceptable GTS for the CES and CFDS.

9.3.6.7 Combined License Information Items

No COL information items are associated with this section.

9.3.6.8 Conclusion

The staff review found that the CES and the CFDS meet GDC 2, 5, 34, and 60, consistent with the guidance provided in DSRS Sections 5.4.7 and 9.3.6; therefore, the staff finds these systems acceptable.

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 control room ventilation system (CRVS) serves the entire control building (CRB) and the access tunnel between the CRB and the reactor building (RXB). The control room envelope (CRE) is isolated and air is provided by the control room habitability system (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 evacuated.

9.4.1.2 Summary of Application

DCA Part 2, Tier 1: The DCA Tier 1 information associated with this section is found in DCA Tier 1, Revision 3, Section 3.2, "Normal Control Room Heating Ventilation and Air Conditioning System."

DCA Part 2, Tier 2: The applicant has provided a system description in DCA Tier 2, Revision 3, Section 9.4.1, "Control Room Area Ventilation System," summarized here, in part, as follows:

> 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, CRVS maintains the CRE within the temperature and humidity limits needed to support personnel and to maintain equipment during all modes of operation, including normal, abnormal, station blackout, and toxic gas conditions.

ITAAC: The applicant gave the ITAAC associated with DCA Part 2, Section 9.4.1, in DCA Part 2, Tier 1, Section 3.1, "Control Room Habitability." These ITAAC are evaluated in Section 14.3 of this SER.

Technical Reports: The applicant did not reference any technical reports related to the CRVS.

Technical Specifications: There are no TS 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 for the CRVS. The following are the relevant requirements of NRC regulations for this area of review:

- GDC 2: 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 the following:

(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, quality, 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

(3) the importance of the safety functions to be performed

- GDC 4: 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 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.

- GDC 5: 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 19, "Control Room": A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 50 mSv (5 rem) whole body, or its equivalent to any part of the body, for the duration of the accident.

  The applicant has requested an exemption from GDC 19 to implement a design-specific Principal Design Criterion (PDC) 19 that maintains the reactor in a safe condition 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 exemption that supports PDC 19 is documented in Section 1.14 of this SER.

- GDC 60, as it relates to system capability to suitably control release of gaseous radioactive effluents to the environment.

- 10 CFR 20.1406: The facility 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.

  CRVS ventilation ducts should be designed to minimize the buildup of radioactive contamination within the ducts. The air filter units (AFUs) should be designed with features that minimize the time required for filter changes.

- 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 DCA and supporting Tier 2 information in accordance with the regulatory basis as provided in Section 9.4.1.3 of this SER.

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

DCA Tier 2, Table 3.2-1, shows the following:

- All CRVS SSCs are not safety related.
- Most CRVS SSCs are seismic Category III. However, note 5 to DCA Part 2, Tier 2, Table 3.2-1, states that where SSCs identified as seismic Category III in this table could, as the result of a seismic event, adversely affect a seismic Category I SSC or result in incapacitating injury to occupants of the control room, they are categorized as seismic Category II.
- Some SSCs are seismic Category I (CRE isolation dampers and position indicators, fire dampers, smoke dampers, radiation monitors (downstream of charcoal filter unit), air intake smoke detectors, outside air isolation damper position, and toxic gas detectors).
- Some SSCs are seismic Category II (outside air isolation dampers for CRV recirculation mode and ductwork and associated components associated with the outside air intake up to the radiation monitors downstream of the filter unit).

The CRE isolation dampers are located within the CRE of the CRB. The CRE is located within a portion of the CRB classified as seismic Category I. The CRE isolation dampers are also designed to seismic Category I and are protected from external events to the extent that the CRB is protected from such events.

The CRE isolation dampers are qualified to shut tight against CRE pressure in support of the CRHS for maintaining MCR habitability.

Radiation monitors downstream of charcoal filter unit and CRE isolation dampers are required to support CRHS operation. These radiation monitors and isolation dampers are not safety related because NuScale considers its control room habitability not a safety-related function. Control room habitability is reviewed in Section 6.4 of this SER.

The staff finds that the guidance of RG 1.29, Regulatory Position C.1.i, for nonsafety-related portions has been followed and therefore concludes that the CRVS complies with the requirements of GDC 2. A more detailed evaluation of seismic analysis of this system is contained in Chapter 3 of this SER.

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

nonsafety-related backup diesel generators (BDGs) are available. The CRVS has radiation monitors, toxic gas monitors, and smoke detectors located in the outside air intake and downstream ductwork, which allow the plant protection system to isolate the CRE and the outside air intake as needed in the event of fires, failures, malfunctions, toxic gas, or high radiation.

The CRB itself is a mild environment with no credible high-energy sources as the result of equipment failure. According to DCA Tier 2, Section 3.6.1.1.4, there are no high-energy lines in the CRB. There is no credible source of 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. 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, DCA Tier 2, Section 9.4.1.3, states, "the CRVS does not have a function relative to shutting down an NPM or maintaining it in a safe shutdown condition. Operation of the CRVS does not interfere with the ability to operate or shut down a unit."

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 make sure 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, 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 and CRHS is safety related. The analyses summarized in Chapter 15 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 design does not credit any operator actions to mitigate DBEs. Specifically, in DCA 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.

The NuScale 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"

Upon detection of smoke or toxic gas in the outside air duct, the outside air isolation dampers are closed 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, or if normal ac power is lost for 10 minutes, or if power is lost to all highly reliable dc power system-common (EDSS-C) battery chargers, the CRE is isolated and breathable air is supplied by the CRHS. The performance of the CRHS under DBA conditions is evaluated in Section 6.4 of this SER.

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 the requirements of PDC 19.

9.4.1.4.5 10 CFR 20.1406, "Minimization of Contamination"

DCA Tier 2, 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 has been 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 main control room (MCR) with respect to adjacent spaces. See Section 9.4.1 for 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. Areas served by the CRVS (MCR and technical support center) are designed to maintain operator doses within PDC 19 limits.
>
> If power is not available, or if a high radiation indication is received from the radiation monitors downstream of the CRVS filter unit, the control room envelope (CRE) isolation dampers close and the control room habitability system 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, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," 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 and ASME AG-1.

The staff finds that, at the design certification stage, the system is consistent with the guidance of RG 1.140 because 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 DCA Tier 2, Section 8.4, "Station Blackout," the applicant stated the following:

> The SBO duration for passive plant designs is 72 hours, which is consistent with Nuclear Regulatory Commission policy provided by SECY-94-084 and SECY-95-132 and the associated staff requirements memorandums. 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 relevant guidelines of Regulatory Guide (RG) 1.155 are applied as they pertain to compliance with 10 CFR 50.63 for the passive NuScale design.
>
> The analysis results show that a safe and stable shutdown is achieved, and that the reactor is cooled and containment integrity is maintained for the 72-hour duration with no operator actions. The core remains subcritical for the duration of the event. The reactor coolant inventory ensures that the core remains covered without the need for makeup systems.
>
> The environmental conditions in the main control room during the SBO were evaluated. 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 DCA Tier 2, Section 9.4.1.3, the applicant stated the following:

> In a station blackout event, the CRE isolation dampers close to form part of the CRE. The CRHS then provides bottled air to the CRE. Along with the CRHS,

> the CRE isolation dampers ensure that a suitable operating environment is maintained to support operators and equipment in the MCR.

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 SER. 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.

9.4.1.4.8 Technical Support Center

The TSC air conditioning, heating, cooling, and ventilation are discussed in DCA Part 2, Tier 2, Section 9.4.1.

9.4.1.4.8.1 Radiation Protection

DCA Part 2, Tier 2, Section 15.0.3.7.2, states the following:

> Accident analyses are performed for one emergency mode: that of uninterrupted power supply with continuous filtered airflow to the Technical Support Center (TSC) envelope for the event duration. In the event of immediate loss of power with control room habitability system (CRHS) activation, TSC personnel are evacuated and the TSC functions are transferred to an alternate site-specific location. With loss of power with CRHS activation, the TSC is evacuated since it is not serviced by the CRHS.

The staff's review of the DBA radiological consequences analyses, including the control room and TSC radiological habitability, is discussed in Section 15.0.3 of this SER.

9.4.1.4.8.2 Technical Support Center Location

NUREG-0696, "Functional Criteria for Emergency Response Facilities," Section 2.2, provides guidance on TSC location and states that the walking time from the TSC to the control room shall not exceed 2 minutes. Also, provisions shall be made to consider the effects of direct radiation and airborne radioactivity from implant sources on personnel traveling between the two facilities.

The NuScale DCA Tier 2, Section 13.3, "Emergency Planning," states that when using the shortest designed direct route, the walking time between the entrance of the control room and the entrance of the TSC does not exceed 2 minutes. Therefore, the staff finds this acceptable as it is consistent with applicable guidance.

9.4.1.4.8.3 Technical Support Center Habitability

NUREG-0696, Section 2.6, provides guidance on TSC habitability, stating that the TSC shall have the same radiological habitability as the control room under accident conditions and that TSC personnel shall be protected from radiological hazards, including direct radiation and airborne radioactivity from in-plant sources under accident

conditions, to the same degree as control room personnel. NUREG-0696, Section 2.6, also states that applicable criteria are specified in GDC 19 and SRP Section 6.4, "Control Room Habitability System."

Regarding the TSC ventilation system, NUREG-0696, Section 2.6, guidance states that the TSC ventilation system shall function in a manner comparable to the CRVS and that a TSC ventilation system that includes high-efficiency particulate air (HEPA) and charcoal filters is needed, as a minimum.

The CRVS serves both the CRE and the TSC, and the TSC has the same radiological and toxic gas habitability controls as the control room under accident conditions. The staff finds that the TSC design is acceptable because NUREG-0696 is satisfied.

9.4.1.5 Combined License Information Items

Table 9.4.1-1 lists the COL information item numbers and descriptions from Table 1.8-2 of DCA Part 2, Tier 2.

Table 9.4.1-1 Combined License Information Items

Item No.	Description	DCA Part 2, Tier 2, Section
9.4-1	A COL applicant that references the NuScale Power plant design certification will specify a periodic testing and inspection program for the normal control room heating ventilation and air conditioning system.	9.4.1

The staff determined that the list adequately describes actions for the COL applicant or holder. No additional COL information items need to be included in DCA Part 2, Tier 2, Table 1.8-2, for control room habitability considerations.

9.4.1.6 Conclusion

The staff reviewed information presented in the DCA on the design and operation of the CRVS. The staff concludes that the design of the CRVS complies with the requirements of 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

This section evaluates DCA Part 2, Tier 2, Section 9.4.2, "Reactor Building and Spent Fuel Area Ventilation System."

The reactor building HVAC system (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.

RBVS indoor design conditions are described in DCA Tier 2, Table 9.4.4-2. RXB harsh zones are identified in DCA Tier 2, Table 3C-1.

9.4.2.2 Summary of Application

DCA Part 2, Tier 1: In Section 3.3, "Reactor Building Heating and Air Conditioning System," the applicant stated the following:

> The RBVS is committed to maintain a negative pressure in the RXB and Radioactive Waste Building (RWB) relative to the outside environment and maintain the hydrogen levels in the RXB battery rooms below one percent by volume.

DCA Part 2, Tier 2: In Section 9.4.2.1, "Design Bases," the applicant stated, in part, the following:

> The RBVS serves no safety-related function and is not credited for mitigating any design basis events. The system is designed to remove radioactive contaminants from the exhaust streams of RXB general area, the radioactive waste building (RWB) general area, and Annex Building (ANB). 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.

ITAAC: The applicant gave the ITAAC associated with DCA Part 2, Section 9.4.2, in DCA Part 2, Tier 1, Section 3.3, "Reactor Building Heating and Air Ventilation System." These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: There are no proposed GTS associated with the RBVS.

Technical Reports: There are no technical reports associated with the RBVS.

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 states in part that SSCs 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 SSCs shall reflect the following:

  (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, quality, 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

  (3) the importance of the safety functions to be performed

- GDC 5 states in part 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 60 states in part that the nuclear power unit design shall include 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. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

- GDC 61 states in part that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity shall be designed to ensure adequate safety under normal and postulated accident conditions. The systems shall be designed as follows:

  (1) with a capability to permit appropriate periodic inspection and testing of components important to safety

  (2) with suitable shielding for radiation protection

  (3) with appropriate containment, confinement, and filtering systems

  (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal

  (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions

- GDC 64 states, in part, that 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 requires 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 RBVS comprises three subsystems to provide the following nonsafety-related functions:

(1) supply subsystem—Maintains the RXB area ambient conditions by AHUs.

(2) general area exhaust subsystem—Removes exhaust air from all areas of the RXB. Exhaust from the annex building HVAC system (ANBVS) joins exhaust from the RBVS and RWBVS. The combined flow enters the RBVS exhaust filter units and exits the plant through the plant exhaust stack.

(3) SFP exhaust subsystem—Removes exhaust air from around most of the SFP perimeter. The SFP exhaust filter units utilize HEPA filters and charcoal adsorbers to reduce radioactivity release to the environment.

During the review, the staff found that DCA Part 2, Tier 2, Section 9.4.2, has a brief description of the interaction of the ANBVS. There is no separate DCA section describing the ANBVS. According to DCA Part 2, Tier 2, Figure 1.2-2, the annex building is at conceptual stage. The staff acknowledges that CDI is outside the scope of the DCA.

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

SSCs of the RBVS whose structure failure could affect the operability of safety-related SSCs should be designed as seismic Category II, conforming to the guidance in RG 1.29.

DCA Tier 2, Section 9.4.2.1, states "components of the RBVS whose structural failure could adversely affect Seismic Category I SSC are designed as Seismic Category II. The remainder of the RBVS is Seismic Category III (non-seismic)."

As shown in DCA Tier 2, Table 3.2-1, all RBVS SSCs including ductwork and associated components (except for SFP exhaust components), and instrumentation are designed to seismic Category III. SFP ductwork and associated components and SFP filter units including fans are designed to seismic Category II.

The staff finds that the guidance of RG 1.29, Regulatory Position C.2, has been followed and therefore concludes that the requirements of GDC 2 are satisfied.

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

None of the RBVS SSCs is safety related. There are no functions required of the RBVS that would impact safety functions for any individual module. Sharing of RBVS SSCs in multiple-NPM plants does not 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). Because of this, the staff concludes that the requirements of GDC 5 are satisfied with respect to the RVBS.

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

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 DCA Part 2, Tier 2, Section 9.4.2.2.1, the general area exhaust filter units are designed, and will be constructed and tested, to meet the applicable performance requirements of ASME N509, N510, and AG-1 in conformance with RG 1.140. The same DCA section states that the SFP exhaust charcoal and filter units conform to RG 1.140. According to DCA Part 2,

Tier 2, Section 9.4.2.4, COL Item 9.4-2 directs COL applicants to specify periodic testing and inspection requirements for the RBVS in accordance with RG 1.140.

The staff reviewed conformance to RG 1.140 as discussed below.

9.4.2.4.3.1 RG 1.140, Regulatory Position C.2

According to RG 1.140, Regulatory Position C.2—

(1) The design of a normal atmosphere cleanup system should be based on the anticipated operating ranges for temperature, pressure, relative humidity, and radiation levels during normal plant operations and anticipated operational occurrences.

(2) If the normal atmosphere cleanup system is located in an area with high radiation during normal plant operation, then adequate shielding of components and personnel from the radiation source should be provided.

(3) The operation of any normal atmosphere cleanup system should not degrade the expected operation of any ESF system that is required to operate after a DBA.

(4) The design of a normal atmosphere cleanup system should consider any significant contaminants such as dust, chemicals, excessive moisture, or other particulate matter that could degrade the cleanup system's operation. Materials of construction and components should be selected and tested to limit the generation of combustibles and contaminants per ASME N509, Section 4.4, "Environmental Design Condition," and various ASME AG-1 sections.

RG 1.140 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 concludes that the applicant's commitment to following RG 1.140 at the design certification stage, 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.

9.4.2.4.3.2 RG 1.140, Regulatory Position C.3

According to RG 1.140, Regulatory Position C.3—

(1) Normal atmosphere cleanup systems should consist of at least the following components: (1) HEPA filters, (2) iodine adsorbers, (3) fans, and (4) interspersed ducts, dampers, and related instrumentation.

(2) To ensure reliable in-place testing, the volumetric airflow rate of a single cleanup unit should be limited to approximately 849.51 m^3/min (30,000 cubic feet per minute (cfm)).

(3) Each normal atmosphere cleanup system should be instrumented to monitor and alarm for pertinent pressure drops and flow rates.

(4) To maintain the radiation exposure to operating and maintenance personnel ALARA, normal atmosphere cleanup systems and components should be designed to control leakage and facilitate maintenance, inspection, and testing consistent with RG 8.8.

(5) Outdoor air intake openings should be equipped with louvers, grills, screens, or similar protective devices to minimize the effects of high winds, rain, snow, ice, trash, and other contaminants on the operation of the system.

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 design certification 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.

DCA Part 2, Tier 2, Section 1.2.2.1, states the following:

> The RXB is a Seismic Category I, reinforced concrete structure with design considerations for the effects of aircraft impact, environmental conditions, postulated design basis accidents (internal and external), and design basis threats. The RXB also provides radiation protection to plant operations and maintenance personnel.

Since the RBVS is located inside the RXB, this system is protected from missiles and designed to withstand the effects of natural phenomena.

DCA Part 2, Tier 2, Section 9.4.3.2, 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 design certification 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 diagram (Figure 9.4.2-1), major component (Table 9.4.2-3) and system instrumentation diagrams (Table 9.4.2-4), and inspection and testing (DCA Tier 2, Section 9.4.2.4) for the system.

Based on the above review, the staff finds that the in-place testing (RG 1.140), the general area exhaust filter unit flow rate (23,000 cfm), and SFP exhaust charcoal and filter unit flow rate (30,000 cfm) satisfy RG 1.140, Regulatory Position C.3.2; the instrumentation of each atmosphere cleanup system (Figure 9.4.2-1) satisfies RG 1.140, Regulatory Position C.3.3; and the missile protection installed for the RXB outdoor air opening (Figure 9.4.2-10) 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 design certification 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 DCA Part 2, Tier 2, Section 15.0.3.8.5, "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 DCA Part 2, Tier 2, Section 9.4.2.3, 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 DCA Tier 2, Section 15.0.3.7.3, "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 61 hours 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 5 mSv (0.5 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 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 GDC 64, "Monitoring Radioactivity Releases"

The SFP and NPMs are located in a leakage-controlled building. Radiation monitors are provided in the SFP exhaust ductwork up stream of the charcoal filter units and in the plant exhaust stack. Radiation monitors are also provided in the general exhaust ductwork from the RWB, RXB, and ANB as depicted in Figure 9.4.2-1.

The staff concludes that the functions of these monitors constitute compliance with GDC 64.

9.4.2.4.6 10 CFR 20.1406, "Minimization of Contamination"

DCA Tier 2, Section 9.4.2.1, states, "[c]onsistent with 10 CFR 20.1406, the design of the RBVS includes provisions to limit the spread of contamination and to facilitate eventual plant decommissioning."

DCA Tier 2, Section 9.4.2.3, states the following:

> [t]he RBVS is designed to move air from areas of typically lower radioactive contamination to areas that potentially more contaminated. The RBVS is designed with ducting runs as short as practical and that do not have sudden directional changes. Interior and exterior duct surfaces are relative smooth. The SFP area exhaust subsystem removes air from this potentially contaminated area and filters the exhaust.

After reviewing the configuration of the RBVS, the staff believes that the ventilation duct design is sufficient to minimize the buildup of radioactive contamination.

The staff notes that the location of RBVS filters allows for filter change to be performed in the RWB. Personnel will have enough space to wear appropriate personnel protection during testing and maintenance periods. This design feature minimizes the time required for filter changes.

Based on the above findings, the staff finds that the RBVS facility design is sufficient to 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 staff therefore concludes that the above described design considerations constitute compliance with 10 CFR 20.1406.

9.4.2.4.7 High-Energy Line Break Outside Containment

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. To provide protection against postulated piping failures in fluid systems outside containment to satisfy GDC 4, NuScale high-energy line break (HELB) calculations are performed to satisfy GDC 4: BTP 3-3, “Protection against Postulated Piping Failures in Fluid Systems Outside Containment”; BTP 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment”; and RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants.”

The essential equipment requiring protection includes the following:

(1) within the top of module (TOM)—containment isolation valves for the main steam, feedwater, CVCS and other isolation valves, as well as instrumentation and RPS electrical cables

(2) within the RXB pool room—refueling crane

As described in audit report “Regulatory Audit of Containment and Ventilation Systems,” dated November 9, 2018 (ADAMS Accession No. ML18291B228), the staff reviewed NuScale TOM and RXB pool room HELB calculations and found the following:

(1) It was assumed that the system is equipped with fire dampers that quickly isolated on high room temperatures consistent with typical industry practice. In other words, passive flow through the HVAC ducts is isolated by fire dampers, and no flow is conservatively assumed through the exhaust ventilation duct. The staff considers that passive building exhaust flow isolation is conservative for building pressure calculation.

(2) To maintain building pressure within “reasonable limits,” the ductwork of the RBVS SFP exhaust subsystem is credited in all GOTHIC TOM/pool room HELB cases as a passive ventilation path to ambient for the purpose of pressure relief within the pool room. This assumes that there are no dampers that close due to smoke, fire, or out-of-service fan for the duration of event. Because the SFP exhaust subsystem has no fire dampers, the staff considers this assumption consistent with typical industry practice.

NuScale calculations show that SSCs are compatible with environmental conditions documented in the applicant’s HELB calculations, which encompass the required event spectrum, including normal operations and postulated accidents.

According to DCA Tier 2, Section 9.2.5.2, to prevent overpressurization in the UHS area of the RXB during abnormal conditions, credit is taken for an overpressurization vent (OPV) included in the RXB system. Until the building pressure reaches the setpoint of the rupture disk, the RBVS will filter and control the release of airborne radioactive material from inside the RXB, including from pool water evaporation for loss of normal power supply.

As discussed above for HELB calculations and OPV design, the staff finds that the RBVS design concerning HELBs outside containment meets the requirements of GDC 4.

9.4.2.5 Combined License Information Items

The following is a list of item numbers and descriptions from Table 1.8-2 of the DCA.

Table 9.4.2-1 Combined License Information Items

Item No.	Description	DCA Tier 2 Section
9.4-2	A COL Applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the Reactor Building heating ventilation and air conditioning system in accordance with RG 1.140.	9.4.2

The staff determined that the list adequately describes actions for the COL applicant or holder. No additional COL information items need to be included in DCA Tier 2, Table 1.8-2, for RXB ventilation considerations.

9.4.2.6 Conclusion

The staff reviewed information presented in the DCA on the design and operation of the RBVS. The staff concludes that the design of the RBVS complies with the requirements of GDC 2, GDC 5, GDC 60, GDC 61, GDC 64, and 10 CFR 20.1406.

9.4.3 Radioactive Waste Building Ventilation System

9.4.3.1 Introduction

This section evaluates DCA Tier 2, Section 9.4.3, “Radioactive Waste Building Ventilation System (RWBVS).”

The RWBVS is designed to support personnel access and equipment functions by maintaining a suitable operating environment in the RWB, including the RWB control and monitoring room. The RWBVS also supports the control of radioactive contamination by maintaining airflow from areas of lesser potential contamination to areas of greater potential contamination, maintaining the RWB at a negative pressure with respect to the outside atmosphere, and collecting potentially contaminated discharges vented from equipment in the RWB.

The applicant stated that the RWBVS does not perform atmosphere cleanup functions (it does not filter exhaust). Portions of the RWBVS (the exhaust fans, exhaust ductwork, and associated components) are considered part of the atmospheric cleanup system because they conduct potentially contaminated air to the RBVS, which filters the exhaust before releasing it to the atmosphere. The staff evaluates the RBVS in Section 9.4.2 of this SER.

9.4.3.2 Summary of Application

DCA Tier 1: There are no Tier 1 entries for the RWBVS.

DCA Tier 2: DCA Part 2, Tier 2, Section 9.4.3.1, "Design Bases," states, in part, that the RWBVS serves no safety-related functions, is not credited for the mitigation of DBAs, and has no safe-shutdown functions. The RWBVS is not credited to operate during abnormal plant conditions. The effects of natural phenomena, such as earthquakes, on the RWBVS do not affect safety-related and risk-significant SSCs. The RWBVS is seismic Category III (nonseismic). The RWBVS is designed to limit hydrogen concentration in battery rooms below 1 percent by volume. The RWBVS does not have a function relative to shutting a unit down or maintaining a unit in a safe-shutdown condition.

The RBVS monitors and filters exhaust from the RWBVS. The design of the RWBVS minimizes contamination of the facility and the environment and therefore facilitates eventual plant decommissioning (10 CFR 20.1406).

DCA Part 2, Tier 2, Section 9.4.3.2, describes the RWBVS that serves the RWB. During normal operation, the RWBVS maintains temperature and humidity control within ranges suitable for the comfort of personnel and to prevent degradation of equipment, as indicated in DCA Tier 2, Tables 9.4.3-1 and 9.4.3-2. The system is designed to direct airflow from areas of lower potential contamination to areas of higher potential contamination. DCA Tier 2, Figure 9.4.3-1, shows the RWBVS flow diagram.

The RWBVS main supply subsystem provides filtered and heated or cooled air to various areas of the RWB, including the bay area, warehouse, pump and tank rooms, other systems in the RWB, and the liquid radioactive waste equipment and mechanical equipment rooms.

In addition to the main supply subsystem, the RWBVS has dedicated two 100-percent-capacity AHUs for each of the following areas: the RWB electrical room and the RWM control and monitoring room. Two 100-percent capacity fan coil units (FCUs) provide HVAC service to the telecommunications room; two additional 100-percent units provide service to the battery charger room.

The RWBVS air intake is located so that plant discharges are not near the intake. The RWBVS AHUs, FCUs, supply air fans, and unit heaters are located in dedicated low-radiation areas in the RWB to maintain exposures ALARA to personnel maintaining the equipment. The HVAC ducting is routed outside of pipe chases containing radioactive waste system piping and does not penetrate pipe chase walls, which could compromise shielding.

Exhaust from the RWBVS is combined with the RBVS exhaust in the mechanical room of the RWB where the RBVS general area exhaust fans and filter units are located. These exhaust fans and filter units are part of the RBVS even though the system's components are physically located within the RWB.

The main RWBVS supply subsystem consists of one AHU with two 100-percent-capacity external supply fans. The AHU housing consists of a prefilter bank, a high-efficiency filter bank, hot water heating coils, and a chilled water cooling coil bank. During winter conditions, a duct-mounted preheat coil preheats outdoor air. DCA Part 2, Tier 2, Table 9.4.3-3, lists the major RWBVS equipment, and DCA Part 2, Tier 2, Table 9.4.3-4, lists system instrumentation. As described in DCA Part 2, Tier 2, Table 9.4.3-2, occupied areas with light work will be kept to between 22.8 °C and 25.6 °C (73 °F and 78 °F), with humidity kept to between 35 and 50 percent; areas with moderate work will be kept to between 18.3 °C and 29.4 °C (65 °F and 85 °F), with humidity kept to between 35 and 60 percent; inaccessible areas without sensitive electronic equipment will be kept to between 10.0 °C and 54.4 °C (50 °F and 130 °F); areas with frequent inspection or maintenance but no sensitive equipment will be kept to between 10.0 °C and 40.6 °C (50 °F and 105 °F); and areas with electronic equipment will be kept to between 18.3 °C and 29.4 °C (65 °F and 85 °F), with humidity kept to between 30 and 55 percent or in accordance with the equipment manufacturer's humidity recommendations.

During normal operation, air enters the RWBVS through an intake located in the exterior wall of the RWB. The air intake and discharge will not be located near one another. Air proceeds through the main supply AHU and then through one of the two supply fans operating continuously to perform the following:

- Provide sufficient ventilation air to maintain the minimum air change rate of 2.0 times per hour throughout the main portions of the building.
- Pressurize designated rooms requiring positive pressure in relation to surrounding spaces within the RWB.
- Provide conditioned air to maintain necessary indoor design conditions to areas not served by local recirculation AHUs or FCUs.

The RWBVS main AHU supply airflow remains constant while the RBVS general exhaust fans modulate speed in response to pressure indicators, flow elements, or both, to maintain the RWB negative pressure at a set point. A duct-mounted temperature sensor located downstream of the AHU controls the temperature of the air supplied by the main RWBVS supply AHU.

The RWBVS is designed to limit hydrogen concentration levels in the battery rooms containing batteries below 1 percent by volume in accordance with RG 1.189, "Fire Protection for Nuclear Power Plants," Regulatory Position 6.1.7 (DCA Table 1.9.2), such that explosive levels of hydrogen are prevented from forming in the RWB battery rooms. As described in DCA Tier 2, Table 9.4.3-2, battery rooms will be kept to between 20.0 °C and 25.0 °C (68 °F and 77 °F), with humidity kept to between 30 and 55 percent or in accordance with the equipment manufacturer's humidity recommendations.

ITAAC: DCA Tier 1, Section 3.3, includes one ITAAC (Tier 1, Table 3.3-1, No. 2) that addresses verification that the RBVS can maintain a negative pressure in the RWB relative to the outside environment. These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: No GTS are associated with the RWBVS.

Technical Reports: No technical reports related to the RWBVS are referenced.

9.4.3.3 Regulatory Basis

SRP Section 9.4.3, Revision 3, "Auxiliary and Radwaste Area Ventilation System," issued March 2007, gives the relevant guidance to meet NRC regulations 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 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. For the RWBVS, guidance is provided in Regulatory Position C.3 in RG 1.52, Revision 4, issued September 2012, and Regulatory Positions C.2 and C.3 in RG 1.140, Revision 2, issued June 2001, as related to design, inspection, testing, and maintenance criteria for postaccident and normal atmosphere cleanup systems, ventilation exhaust systems, air filtration, and absorber units.

9.4.3.4 Technical Evaluation

The staff reviewed the RWBVS (DCA Tier 2, Section 9.4.3) for conformance with the requirements and acceptance criteria defined in SRP Section 9.4.3.

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

The guidance for GDC 2 is based on RG 1.29, Regulatory Positions C.1 and C.2, for safety-related SSCs and SSCs that are not safety-related if their failure could reduce the functioning of safety-related SSCs. Any SSCs not falling under Regulatory Positions C.1 and C.2 are considered seismic Category III.

In DCA Tier 2, Sections 9.4.3.1 and 9.4.3.3, the applicant stated that the RWBVS has no safety-related functions and that the RWB has no safety-related components. Failure of the RWBVS to operate does not prevent SSCs from performing their safety-related functions. The applicant considered GDC 2 in the design of the RWBVS, in that the RWBVS is fully contained in the RWB and the RWB has no safety related or seismic Category I equipment; therefore, the failure of the RWBVS will not affect the performance of safety-related functions. The RWBVS is seismic Category III (nonseismic).

SRP Section 3.2.1, Revision 3, "Seismic Classification," issued August 2016, provides guidance to the staff in reviewing seismic classification of SSCs for nuclear power plant applications. Accordingly, the staff reviewed DCA Tier 2, Table 3.2-1, and finds the all RWBVS SSCs and components are designed to seismic Category III and assigned an SSC classification as B2 (not safety related and not risk significant).

Based on these classifications, the staff finds that the guidance in RG 1.29 is satisfied and concludes that applicant met the requirements of GDC 2.

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

The guidance in GDC 5 governs the sharing of SSCs important to safety among nuclear power plant units to ensure that such sharing will not significantly impair the SSCs' ability to perform their safety functions.

The staff reviewed the RWBVS design to ensure that it meets the relevant requirements of GDC 5. DCA Tier 2, Sections 9.4.3.1 and 9.4.3.3, state that the RWBVS design considered GDC 5. The RWBVS does not have a function relative to shutting a unit down or maintaining a unit in a safe-shutdown condition. As the RWBVS does not affect the safe and orderly shutdown and cooldown of the NPMs, the staff finds that the RWBVS design meets 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 anticipated operational occurrences.

In DCA Tier 2, Sections 9.4.3.1 and 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 anticipated operational occurrences, is controlled. DCA Tier 2, Section 9.4.2, describes the RBVS.

Because the RBVS controls, monitors, and filters the RWBVS exhaust, the staff evaluates the RBVS in Section 9.4.2 of this SER, which considers the requirements of GDC 60. Based on the evaluation in 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"

DCA Tier 2, Section 9.4.3.2, states that the ductwork, supports, and accessories meet the design and construction requirements of Sheet Metal and Air Conditioning Contractor National Association Standard 1780, "HVAC Systems Testing, Adjusting and Balancing"; Standard 1520, "Round Industrial Duct Construction Standards"; and Standard 1966, "HVAC Duct Construction Standards—Metal and Flexible." Ducting interior and exterior surfaces have relatively smooth finishes to reduce localized collection of radioactive contamination.

The design will ensure that ducting runs are kept to a minimum and abrupt changes in ducting direction are avoided. Inner and outer surfaces of ducting have relatively smooth finishes. These design considerations facilitate the eventual decommissioning of the plant. Based on these design features, the staff finds that the RWBVS satisfies the requirements of 10 CFR 20.1406.

9.4.3.5 Initial Test Program

With regard to preoperation inspection and tests for this system, DCA Tier 2, Section 9.4.3.4, "Inspection and Testing," states that component fans, cooling coils, and electrical coils are

factory tested and certified. A system air balance test and adjustment to design conditions are conducted in the course of the plant preoperational test program, as indicated in DCA Tier 2, Table 14.2-21, "Radioactive Waste Building HVAC System Test #21," which lists component tests for the functional verification of RWBVS fans, dampers, and instrumentation to ensure compliance with RWBVS design requirements.

The ITP is evaluated in Section 14.2 of this SER.

9.4.3.6 Combined License Information Items

Table 9.4.3-1 lists the COL item number and description from DCA Tier 2, Table 1.8-2.

Table 9.4.3-1 NuScale COL Information Item for Section 9.4.3

COL Item No.	Description	DCA Tier 2 Section
9.4-3	A COL applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the Radioactive Waste Building heating, ventilation, and air conditioning.	9.4.3

9.4.3.7 Conclusion

The staff evaluated the NuScale RWBVS using the acceptance criteria of SRP Section 9.4.3. Based on this review, the staff concludes that DCA Tier 2, Section 9.4.3, and other related sections provide sufficient information and that the RWBVS will be designed to comply with GDC 2, GDC 5, and GDC 60.

9.4.4 Turbine Building Ventilation System

9.4.4.1 Introduction

DCA Part 2, Tier 2, Section 1.2.2.5.1, "Turbine Generator Building," states that the NuScale Power Plant contains two separate turbine generator buildings (TGBs), which are not safety related. Each building houses six turbine generator sets along with their auxiliaries, condensers, condensate systems, and the feedwater systems. A laydown area and overhead crane are provided for installation and maintenance activities.

Each TGB has a dedicated turbine building ventilation system (TBVS). The TBVS is designed to maintain a suitable environment for all equipment and personnel in the TGB during startup, shutdown, and normal plant operation and does not serve any safety-related functions.

9.4.4.2 Summary of Application

DCA Tier 1: DCA Part 2, Tier 1, does not mention the TBVS.

DCA Tier 2: DCA Part 2, Tier 2, Section 9.4.4.1, "Design Bases," states that the TBVS is designed to meet the following design bases:

- Provide ventilation for areas of the TGB to maintain a suitable environment for plant equipment and personnel.
- Maintain hydrogen levels in battery rooms below explosive concentrations.

- Prevent migration of smoke, hot gases, and fire suppressants from areas affected by fires to nonaffected areas.

DCA Part 2, Tier 2, Section 9.4.4.2, “System Description,” describes the TBVS that serves the turbine operating room, TGB battery room, TGB battery charging room, and turbine maintenance room to fulfill the design bases listed above, as summarized below.

Nine roof-mounted exhaust fans pull makeup intake air from exterior wall-mounted louvers with associated dampers to provide turbine operating room ventilation. Room-mounted electric unit heaters arranged around the perimeter of the operating room provide heating during the winter. Fire dampers located at duct penetrations through fire barriers maintain the fire-resistance ratings of the barriers. Smoke dampers provide smoke isolation of areas or AHUs. Combination fire and smoke dampers located at duct penetrations through fire barriers maintain the fire-resistance ratings of the barriers and provide smoke isolation. As described in DCA Part 2, Tier 2, Table 9.4.4-2, occupied areas are kept to between 18.3 °C and 29.4 °C (65 °F and 85 °F), with humidity kept to between 35 and 60 percent; areas with frequent inspection or maintenance and no sensitive equipment will be kept between 10.0 °C and 40.6 °C (50 °F and 105 °F); and areas with sensitive equipment will be kept between 18.3 °C and 29.4 °C (65 °F and 85 °F) and in accordance with the equipment manufacturer’s humidity recommendations.

Two split system air conditioning units provide cooling and heating for the TGB battery room. For extremely low temperatures, a room-mounted electric heater provides supplemental heating. A standalone humidifier provides humidification, and a standalone dehumidifier provides dehumidification. An exhaust fan in the battery room maintains the hydrogen concentration in the space to less than 1 percent by volume. Air exhausts directly to outside the TGB, and makeup air for the exhaust fan is pulled through a vent path in the door. The battery room temperature will be kept between 20.0 °C and 25.0 °C (68 °F and 77 °F) and in accordance with the equipment manufacturer’s humidity recommendations, as described in DCA Part 2, Tier 2, Table 9.4.4-2.

Two air conditioning units (one primary, one standby) cool and heat the TGB battery charging room. A room-mounted electric heater provides supplemental heating. A standalone humidifier provides humidification, and a standalone dehumidifier provides dehumidification.

In the turbine maintenance room, a dedicated air conditioning unit provides cooling, primary heating, and ventilation air. For extremely low temperatures, room-mounted electric heaters provide supplemental heating.

DCA Part 2, Tier 2, Section 9.4.4.5, indicates that duct-mounted smoke detectors are in HVAC ductwork and automatically shut down the respective HVAC equipment upon smoke detection.

DCA Part 2, Tier 2, Section 9.4.4.2, states that the turbine buildings are not directly connected to the RXB, the RWB, or any other areas that may contain radioactive contaminants, and that 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 DCA Part 2, Tier 2, Section 9.4.4, does not indicate that filtration and adsorption systems are present anywhere in the TBVS.

ITAAC: The applicant has not proposed any ITAAC for the TBVS.

Technical Specifications: No GTS are associated with the TBVS.

Technical Reports: No technical reports are related to the TBVS.

9.4.4.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this 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

The guidance in SRP Section 9.4.4, Revision 3, "Turbine Area Ventilation System," issued March 2007, lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

9.4.4.4 Technical Evaluation

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

The guidance for GDC 2 is based on RG 1.29, Revision 5, issued July 2016. Regulatory Position C.1 applies to safety-related SSCs, and Regulatory Position C.2 applies to SSCs that are not safety related if their failure could reduce the functioning of safety-related SSCs. Any SSCs that do not fall into Regulatory Positions C.1 or C.2 are considered seismic Category III.

DCA Part 2, Tier 2, Sections 9.4.4.2 and 9.4.4.3, state that no safety-related SSCs are in the TGB, failure of the TBVS will not affect safety-related SSCs, and the TBVS is seismic Category III (nonseismic). DCA Tier 2, Section 3.8.4.1.4, "Other Structures," states that the TGBs are in conceptual design but are separated from the RXB by approximately 21.3 m (70 ft). DCA Part 2, Tier 2, Section 9.4.4.2, states, "The turbine buildings are not directly connected to the Reactor Building, the Radwaste Building, or any other areas that may contain radioactive contaminants."

SRP Section 3.2.1 provides guidance to the NRC staff in reviewing the seismic classification of SSCs for nuclear power plant applications. Accordingly, the staff reviewed DCA Part 2, Tier 2, Table 3.2-1. The staff finds that the TGB and the turbine building cranes are SSCs classified as B2 (not safety related and not risk significant) and seismic classified as "Class III (nonseismic)".

After reviewing the safety classification of SSCs in the TGB serviced by the TBVS, the staff finds that there are no safety-related SSCs served by or that would be affected by failure of the TBVS. Therefore, the staff finds that DCA Part 2, Tier 2, Chapters 3 and 9, are consistent with regard to the safety and seismic classifications of the TBVS.

Accordingly, the staff finds that the TBVS conforms to the guidance in RG 1.29, and therefore, the requirements of GDC 2 have been met.

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

GDC 5 governs the sharing of SSCs important to safety among nuclear power plant units to ensure that such sharing will not significantly impair the SSCs' ability to perform their safety functions.

The staff reviewed the design of the TBVS to ensure that the relevant requirements of GDC 5 are met.

DCA Part 2, Tier 2, Section 9.4.4.3, states, "GDC 5 was considered in the design of the TBVS. The TBVS does not have any function relative to shutting a unit down or maintaining a unit in a safe shutdown condition." The staff's review of DCA Part 2, Tier 2, Chapters 3 and 9, shows that the TGB and the TBVS are independent of and physically separated from the RXB, which houses the multiunits (nuclear power modules).

Thus, even though NuScale is a multimodule power plant, operation of the TBVS does not affect the safe and orderly shutdown and cooldown of the NPMs. Accordingly, the staff finds that the relevant requirements of GDC 5 have been met.

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 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.

DCA Tier 2, Section 9.4.4.3, states, "GDC 60 was considered in the design of the TBVS. During normal operation, radioactive material is not expected to be present in the TGB; therefore, the TBVS does not include radioactivity monitoring and filtration."

The staff reviewed the TBVS to ensure that the relevant requirements of GDC 60 are met. Because the TGB is not expected to contain any radioactive materials during normal operation, the TBVS does not have radioactivity monitoring and air filtration and adsorption units. DCA Part 2, Tier 2, Section 9.4.4.3, states the following:

> The only potential source of radioactive material in the TGB is from a postulated steam generator tube failure. The main steam system and the Condensate and Feed-water System described in Chapter 10, have process radiation monitors to detect radioactive material introduced into the TGB.

RG 1.140 and RG 1.52 apply to filtration and adsorption systems, which are not present in the TGB. Therefore, these RGs are not applicable. Similarly, RG 1.140 and RG 1.52 would not be applicable to the TGB in any other design if they do not have filtration and adsorption systems. Based on these findings, the staff concludes that the requirements of GDC 60 are met for the TBVS.

9.4.4.5 Initial Test Program

With regard to preoperational inspection and tests for the TBVS, DCA Tier 2, Section 9.4.4.4, states the following:

Components are tested and inspected prior to installation and as an integrated system following installation. System airflows are measured and adjustments are made to ensure compliance with design requirements. Preoperational testing of the TBVS is performed in accordance with the requirements of [FSAR Tier 2] Section 14.2.

DCA Tier 2, Table 14.2-22, "Turbine Building Ventilation Test #22," lists the component tests for the functional verification of TBVS fans, dampers, and instrumentation. Based on this, the staff finds that the applicant included adequate preoperational inspections and tests for the TBVS.

9.4.4.6 Combined License Information Items

Table 9.4.4-1 lists the COL item number and description from DCA Tier 2, Table 1.8-2.

Table 9.4.4-1 NuScale COL Information Item for Section 9.4.4

COL Item No.	Description	DCA Tier 2 Section
9.4-4	A COL applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the turbine building heating, ventilation, and air conditioning.	9.4.4

9.4.4.7 Conclusion

The staff evaluated the TBVS for the NuScale standard 12-module plant design using the acceptance criteria guidance defined in SRP Section 9.4.4 and against the regulatory requirements described in Section 9.4.4.3 of this SER. Based on this review, the staff concludes that (1) DCA Part 2, Tier 2, Section 9.4.4, and other related sections contain sufficient information, and (2) the TBVS complies with the requirements of GDC 2, GDC 5, and GDC 60.

9.5 Other Process Auxiliaries

9.5.1 Fire Protection Program

9.5.1.1 Introduction

The fire protection program (FPP) comprises the integrated effort involving components, procedures, analysis, and personnel used in defining and carrying out all activities of fire protection. The fire protection system (FPS) is part of the overall FPP and includes the fire detection, notification, and suppression systems, as designed, installed, and maintained in accordance with applicable codes and standards. The applicant stated that the FPS is classified as a system that is not safety related. It does not perform safety-related functions, nor is the FPS credited with providing emergency backup functions that support operation of safe-shutdown systems.

9.5.1.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 3.7, "Fire Protection System," describes the FPS.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 9.5.1, "Fire Protection Program," describes the design basis and the FPP. DCA Part 2, Tier 2, Appendix 9A, "Fire Hazard Analysis," provides information on the fire hazards analysis (FHA) by evaluating the potential for the occurrence of fire within the plant 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.

ITAAC: The applicant gave the inspections, tests, analysis, and acceptance criteria (ITAAC) associated with DCA Part 2, Tier 2, Section 9.5.1, in DCA Part 2, Tier 1, Table 3.7-1, "Fire Protection System Inspections, Tests, Analyses, and Acceptance Criteria," items 1, 2, 3, and 4. These ITAAC are evaluated in Section 14.3 of this SER.

Initial Test Program: Preoperational tests related to the FPS being evaluated as part of the design certification are described in DCA Part 2, Tier 2, Table 14.2-25, "Fire Protection System Test #25," and DCA Part 2, Tier 2, Table 14.2-26, "Fire Detection System." The ITP is evaluated in Section 14.2 of this SER.

Technical Specifications: There are no proposed GTS associated with the FPS.

Technical Reports: There are no technical reports associated with the FPS.

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 certification 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 FPSs and potential fire impacts on shared SSCs important to safety.
- GDC 19, as it applies to providing the capability both inside and outside the control room to operate plant systems necessary to achieve and maintain safe-shutdown conditions.

The applicant has requested an exemption from GDC 19 to implement a design-specific Principal Design Criterion (PDC) 19 that maintains the reactor in a safe condition 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 exemption that supports PDC 19 is documented in Section 1.14 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 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.

RG 1.189 provides guidance and acceptance criteria for one acceptable approach for an FPP that meets regulatory requirements.

9.5.1.4 Technical Evaluation

9.5.1.4.1 GDC 3, "Fire Protection"

The staff reviewed the applicant's FPP against the four requirements described in GDC 3: 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.

9.5.1.4.1.1 Minimizing the Probability and Effect of Fires and Explosions

The staff reviewed DCA Part 2, Tier 2, Section 9.5.1, 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. DCA Part 2, Tier 2, Appendix 9A, provides a fire safe-shutdown plan that demonstrates that the NuScale 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 penetration seals to prevent the spread of fire between areas. The FHA defines the locations of fire areas and fire barriers.

The applicant provided an FHA for the RXB, CRB, and RWB. The NuScale DCA includes these three buildings. The applicant 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:

- 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.

The applicant 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.

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. National Fire Protection Association (NFPA) 80, “Standard for Fire Doors and Other Opening Protectives,” provides requirements for fire doors.

Fire dampers in 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, UL Standard 555, “Fire Dampers,” provides criteria for the design, fabrication, and testing of fire dampers.

Openings in fire barriers for pipes, conduits, and cable trays that separate fire areas are sealed 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 ASTM E-119, “Standard Test Methods for Fire Tests of Building Construction and Materials.” Additional guidance documents include ASTM E-814, “Standard Test Method for Fire Tests of Through-Penetration Fire Stops,” and Institute of Electrical and Electronic Engineers (IEEE) 634, “IEEE Standard Cable Penetration Fire Stop Qualification Test.”

As part of the FHA, the applicant 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.

The applicant 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, Regulatory Position 8.2, "Enhanced Fire Protection Criteria," describes the control room and the reactor containment building as such areas.

The applicant 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.

The applicant stated that the NPM containment is not a structural part of a building but rather a movable metal vessel. DCA Part 2, Tier 2, Section 9.3.6, states that the CES maintains the containment vessel (CNV) at a vacuum that cannot sustain fire. Electrical conductors within the CNV are of noncombustible construction or routed in a conduit, which results in no intervening combustible loading for an exposure fire impacting other cable or components in the containment. The vacuum is maintained until the containment is flooded with water to facilitate decay heat removal during shutdown and cooldown for refueling. Although the vacuum is eliminated as a result of this process, flooding the CNV acts to preclude the possibility of fire.

The applicant stated that the NuScale designs includes an area at the top of a module under the bioshield that is entirely enclosed by a 3-hour rated fire barrier or the spread of fire to or from the area is eliminated by other means that create a separate fire area enclosing the top of each module, thereby providing separation from other modules.

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 of 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, "Criteria for Independence of Electrical Safety Systems," issued 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 and identified safe-shutdown components for which fire-induced circuit faults could directly or indirectly prevent safe shutdown. The applicant stated that the FHA and fire safe-shutdown plan address possible fire-induced failures, including multiple spurious actuations. Consistent with RG 1.189, Regulatory Position 5.3.1.1, the applicant used the methodology described in Chapter 4 of 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. 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 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.

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.

9.5.1.4.1.2 Use of Noncombustible and Heat-Resistant Materials

The staff reviewed DCA Part 2, Tier 2, Section 9.5.1 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.

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.

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. Only metallic tubing is used for conduit. Thin-wall metallic tubing is not used. Flexible metallic tubing is used only in short lengths to connect components to equipment.

Where used in the control room, carpeting is tested in accordance with ASTM D2859, "Standard Test Method for Flammability of Finished Textile Floor Covering Materials," to determine the flammability characteristics of the material.

The liquid, gaseous, and solid radioactive waste processing and storage systems described in DCA Part 2, Tier 2, 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.

9.5.1.4.1.3 Fire Detection and Fighting Systems

The staff reviewed DCA Part 2, Tier 2, Section 9.5.1 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 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," and Class I circuits, as defined in NFPA 70, "National Electrical Code."

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 separate fresh water supplies are available.
- 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 100-percent-capacity tanks are installed and interconnected so that the 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.
- Fire water supplies are filtered and treated as necessary to prevent and control biofouling or microbiologically induced corrosion of the fire water systems.

Fire pump installations conform to NFPA 20, "Standard for the Installation of Stationary Pumps for Fire Protection." Consistent with NFPA 13, "Standard for the Installation of Sprinkler Systems," each pump is capable of delivering the demand from the largest sprinkler or deluge system plus an additional 1900 L/min (500 gpm) for fire hoses.

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. Automatic water spray systems are installed in accordance with NFPA 15, "Standard for Water Spray Fixed Systems for Fire Protection."

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.

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.

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.

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.

The staff finds that the proposed design conforms to GDC 3 in that the design provides fire detection and firefighting systems of appropriate capacity and capability to minimize the adverse effects of fires on SSCs.

9.5.1.4.1.4 Rupture and Inadvertent Operation of Firefighting Systems

The staff reviewed DCA Part 2, Tier 2, Section 9.5.1 and 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, DCA Part 2, Tier 2, Section 3.4, "Water Level (Flood) Design," evaluates the impact of inadvertent actuation or breaks in the FPS water supply piping. No credit was taken for the floor drains of the RWDS or the 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 module. Facility design ensures that fire water discharge in one area does not impact safety-related equipment in adjacent areas.

The staff finds that the proposed design conforms to GDC 3, in that the firefighting systems are designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of SSCs.

*9.5.1.4.2 GDC 5, "*Sharing of Structures, Systems, and Components"

The staff reviewed DCA Part 2, Tier 2, Section 9.5.1 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 are 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.

9.5.1.4.3 PDC 19, "Control Room"

The staff reviewed DCA Part 2, Tier 2, Section 9.5.1 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 control room, 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.

The NuScale design also incorporates a remote shutdown room that permits control and monitoring of systems related to safe shutdown when the MCR has to be abandoned because of fire. This alternative capability is physically and electrically independent of the control room. 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 8-hour battery backup power. Because the FPS protects the building and associated SSCs that house the control room and the control room itself and provides for a remote shutdown station, the staff finds that the design conforms to PDC 19.

9.5.1.4.4 GDC 23, "Protection System Failure Modes"

The staff reviewed DCA Part 2, Tier 2, Section 9.5.1 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. In DCA Part 2, Tier 2, Sections 3.11 and 4.6 and Chapter 7 evaluate the analysis of the MPS as it relates to GDC 23. Because the MPS fails into a safe state when exposed to adverse conditions such as heat from a fire, the staff finds that the design conforms to GDC 23.

9.5.1.4.5 Enhanced Fire Protection Criteria

The staff reviewed DCA Part 2, Tier 2, Section 9.5.1 and Appendix 9A, to ensure that it conforms to SECY-90-016. SECY-90-016, Section II.D, "Fire Protection," states the following:

> To minimize fire as a significant contributor to the likelihood of severe accidents for advanced plants, the staff concludes that current NRC guidance must be enhanced. Therefore the evolutionary ALWR designers must ensure that safe shutdown can be achieved, assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. Because of its physical configuration, the control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. Evolutionary ALWRs must provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, the evolutionary ALWR designers must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions.

The applicant stated that the RXB, CRB, and RWB are subdivided into 3-hour fire barriers to ensure adequate equipment and cable separation to meet the enhanced fire protection criteria. The control room, containment, and area under the bioshield are not included. During normal operation, the inside of the CNV is maintained at approximately 7 kilopascals (absolute) (1 pound per square inch, absolute), which cannot sustain fire. Additionally, the results of the FHA demonstrate that safe and stable conditions can be maintained without the need for repairs or local operator action.

The NuScale design provides for an alternative shutdown capability so that in the event of a fire in the control room, safe shutdown can be monitored from the RSS.

The applicant stated that the design of the plant and ventilation systems is such that smoke, hot gases, and fire suppressant in a single fire area will not migrate into other fire areas and adversely impact the safe-shutdown capability. Sections 9.4.1, 9.4.2, and 9.4.3 of this SER describe ventilation system performance.

Because the proposed design conforms to the criteria in SECY-90-016, the staff finds the design acceptable.

9.5.1.5 Initial Test Program

The preoperational test related to the FPS for design certification is Fire Protection Systems Test #25. This test is performed in accordance with DCA Part 2, Tier 2, Table 14.2-25. Section 14.2 of this SER provides the staff's evaluation of the plant's ITP.

9.5.1.6 Technical Specifications

There are no GTS requirements associated with the FPS. The system is not safety related and is not required for safe shutdown, and it does not meet a criterion in 10 CFR 50.36 that would require a TS. Therefore, the staff finds this acceptable.

9.5.1.7 Combined License Information Items

Table 9.5.1-1 lists COL information item numbers and descriptions related to fire protection.

Table 9.5.1-1 NuScale COL Information Items for Section 9.5.1

COL Item No.	Description	FSAR Tier 2 Section
13.1-2	A COL applicant that references the NuScale Power Plant design certification will provide a description of the proposed structure, functions, and responsibilities of the onsite organization necessary to operate and maintain the plant. The proposed operating staff shall be consistent with the minimum licensed operator staffing requirements in Section 18.5.	9.5.1
13.1-3	A COL applicant that references the NuScale Power Plant design certification will provide a description of the qualification requirements for each management, operating, technical, and maintenance position described in the operating organization.	9.5.1
13.4-1	A COL applicant that references the NuScale Power Plant design certification will provide site-specific information, including implementation schedule, for operational elements of the Fire Protection Program.	9.5.1
13.5-1	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific procedures that provide administrative control for activities that are important for the safe operation of the facility consistent with the guidance provided in RG 1.33, Revision 3.	9.5.1

9.5.1.8 Conclusion

Based on the review above, the staff concludes that the FPP for the NuScale design satisfies the relevant requirements for the FPP as described in the Regulatory Basis of this 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.

NuScale's COMS comprise the following systems:

- private branch exchange
- plant radio
- public address and general alarm
- sound-powered telephone
- distributed antenna
- security communications

9.5.2.2 Summary of Application

DCA Part 2, Tier 1: NuScale did not provide DCA Part 2, Tier 1, information associated with this section because there are no design commitments for the COMS.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 9.5.2, "Communication System," discusses the COMS.

ITAAC: No ITAAC are associated with DCA Part 2, Tier 2, Section 9.5.2, because the NuScale design adheres to the standardized ITAAC provided by the NRC, and there are no COMS-related standardized ITAAC.

Technical Specifications: No TS are associated with DCA Part 2, Tier 2, Section 9.5.2.

Technical Reports: No technical reports are associated with DCA Part 2, Tier 2, 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

- 10 CFR 50.55a, “Codes and Standards,” as it relates to the provision that SSCs are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed

- 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

  The applicant has requested an exemption from GDC 19 to implement a design-specific PDC 19 that maintains the reactor in a safe condition 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 exemption that supports PDC 19 is documented in Section 1.14 of this report.

- 10 CFR 73.45(e)(2)(iii), 10 CFR 73.45(g)(4)(i), 10 CFR 73.45(g)(4)(ii), 10 CFR 73.46(f), 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 provides the following acceptance criteria that confirm that the above requirements have been adequately addressed:

- Information on the requirements of 10 CFR Part 50, Appendix E, Part IV.E(9), will be found acceptable if adequate provisions for communications are made and described for emergency facilities and equipment, including, but not limited to, at least one onsite and one offsite COMS; each system shall have a backup power source.

- Information on the requirements of 10 CFR 50.34(f)(2)(xxv) and TMI Action Plan Item III.A.1.2 will be found acceptable if adequate provisions for communications are made to support an onsite TSC, an onsite OSC, and a near-site EOF.

- Information on the requirements of 10 CFR 50.47(b)(6) and 10 CFR 50.47(b)(8) will be found acceptable if adequate provisions for communications are provided and

maintained in the emergency facilities and control room to support the emergency response, including prompt communication among principal response organizations to emergency personnel and to the public.

- Information on the requirements of 10 CFR 50.55a will be found acceptable if SSCs are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

- Information on the requirements of GDC 1 will be found acceptable if 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. Where generally recognized codes and standards are used, the application shall identify and evaluate the codes and standards to determine their applicability, adequacy, and sufficiency; such codes and standards shall be supplemented or modified as necessary to ensure a quality product in keeping with the required safety function.

- Information on the requirements of GDC 2 will be found acceptable if 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 without the loss of capability to perform their safety functions and other requirements specifically identified in GDC 2.

- Information on the requirements of GDC 3 will be found acceptable if 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 and other requirements specifically identified in GDC 3.

- Information on the requirements of GDC 4 will be found acceptable if 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, and other requirements specifically identified in GDC 4.

- Information on the requirements of GDC 19 will be found acceptable if the application describes adequate communication equipment that is provided at appropriate locations inside the control room and designed with the capability to support all normal and emergency operations, including intraplant communications and plant-to-emergency facilities and offsite communication requirements even in the event of a single failure within a communication subsystem or the loss of the normal power source and other requirements specifically identified in GDC 19.

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

The NRC staff used a graded review approach, as described in Section 1.1 of this SER. This section describes how the staff applied the graded review approach for the COMS and identifies the safety-significant aspects of the review. The COMS functions are not safety related and not risk significant.

Sections 3.2.2, 17.4, and 19.1 of this SER 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. For the B2 (not safety related and not risk significant) functions of the COMS, the staff primarily focused on identifying performance-based activities (e.g., tests or inspections) within the operation program requirements that can be used to satisfy the acceptance criteria. The staff also focused its review to ensure that the COMS will not adversely impact any safety-related functions. Section 7.0.4.2 of this SER provides the 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. The applicant's public address and general alarm system, private branch exchange, sound-powered telephone system, and the plant radio system provide onsite communications. The applicant's private branch exchange and plant radio system provide offsite communications. A diverse EDNS that is not safety related supplies power to these systems. Four independent voice communications systems provide onsite communications. The failure of any or all of them does not affect safety-related equipment. DCA Part 2, Tier 2, Section 9.5.2, contains COL Information Item 9.5-1, which states the following:

> A COL applicant that references the NuScale Power Plant design certification 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 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 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 13.3 of this SER provides the staff's evaluation of the design details for the TSC and OSC. Because the design includes an onsite TSC and onsite OSC, the staff finds that the applicant has met the requirements of 10 CFR 50.34(f)(2)(xxv) with respect to COMS.

9.5.2.4.3 Compliance with 10 CFR 50.47(b)(6) and 10 CFR 50.47(b)(8)

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.

DCA Part 2, Tier 2, 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 control room to support the emergency response, including prompt communication among principal response organizations to emergency personnel and to the public.

The TSC and OSC provide prompt communications among principal response organizations. Section 13.1 of this SER provides the staff's evaluation of the design details for the TSC and OSC. Section 13.3 also states, in part, that the design of the TSC complies with NUREG-0696. DCA Part 2, Tier 2, Section 9.5.2.3, also states, in part, the following:

> TSC and OSC are equipped with voice communications such as private branch exchange, public address and general alarm system, plant radio, and sound powered telephone systems, which provide communications between the TSC and OSC 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 and an OSC that are 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 staff finds that the requirements of 10 CFR 50.47(b)(6) and 10 CFR 50.47(b)(8) have been met.

9.5.2.4.4 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. Nevertheless, DCA Part 2, Tier 2, Section 9.5.2.3, states, in part, the following:

> Consistent with GDCs 1 and 10 CFR 50.55a, 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. Recognized codes and standards are identified and evaluated to determine their applicability, adequacy, and sufficiency and supplemented or modified as necessary to assure a quality product in keeping with the required safety 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 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. DCA Part 2, Tier 2, Section 9.5.2.3, states, in part, the following:

> Consistent with GDC 2, portions of the COMS whose structural failure could adversely affect the function of Seismic Category I SSC are designed to Seismic Category II requirements in accordance with Section 3.2.1.2.

Section 3.2.1 of this SER provides the staff's evaluation of the SSCs' seismic classification. Since portions of the COMS whose structural failure could adversely affect the function of seismic Category I SSCs are designed to seismic Category II requirements, the staff finds that the requirements of GDC 2 have been met.

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. DCA Part 2, Tier 2, Section 9.5.2.3, states, in part, the following:

> Consistent with GDC 3, the COMS systems are designed and located to minimize consistent with other safety requirements, the probability and effect of fires and explosions.

Section 9.5.1 of this SER provides the staff's evaluation of the fire protection features. DCA Part 2, Tier 2, Section 9.5.2.2.1, states, in part, the following:

> Consistent with the requirements of Regulatory Guide 1.189 Position 4.1.7, the COMS is designed to provide effective communication 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 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. DCA Part 2, Tier 2, 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." Nevertheless, the COMS is designed to work in the environments in which it is located. DCA Part 2, Tier 2, Section 9.5.2.2.3, "System Operation," states, in part, the following:

> COMS equipment is designed to operate reliably within the environment in which it is installed including environmental conditions such as temperature, humidity, radiation, and noise. Furthermore the COMS is designed to operate taking into account placement of barriers such as shield walls. COMS equipment is accessible to personnel for operation, inspection, maintenance, and testing.

DCA Part 2, Tier 2, Section 9.5.2.2.3, also states, in part, that the COMS does not adversely impact other plant systems with EMI and RFI and that the system is designed to the guidelines of RG 1.180. However, because the COMS is not an important-to-safety or risk-significant SSC, the staff finds that the COMS does not need to be credited for evaluating compliance with GDC 4.

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. DCA Part 2, Tier 2, Section 9.5.2.3, states, in part, the following:

> A failure in a COMS subsystem will not significantly impair the ability of the other COMS subsystems to perform, including in the event of an accident in one NuScale Power Module and an orderly shutdown and cooldown of the remaining NuScale Power Modules.

DCA Part 2, Tier 2, Section 9.5.2.3, also states, in part, the following:

> …the public address and general alarm system, private branch exchange, and plant radio 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 three independent voice communications systems are designed and installed to provide assurance that any single event does not cause a complete loss of intra-plant communication.

Because the COMS is designed and shared among the different modules such that it does not significantly impair their ability to perform their safety functions, the staff finds that the requirements of GDC 5 have been met.

PDC 19 requires that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. 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 design provides a control room to operate the plant safely under normal and accident conditions. However, DCA Part 2, Tier 2, Section 9.5.2.3, states, in part, “The NuScale Power Plant 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. DCA Part 2, Tier 2, Section 9.5.2.3, also states the following:

> However the various independent and diverse communications systems located in the MCR significantly increase the overall command and control the reactor operators have over the plant by providing the ability to communicate and direct activities with operations, maintenance, health physics, firefighters, security, and rescue teams. The NuScale Power Plant has an independent plant radio system for security purposes. Other communications systems such as the public address and general alarm system and private branch exchange are available as alternate means, if necessary.

As the control room operators do not need the COMS to take actions for safe shutdown during normal and accident conditions, the staff finds that the COMS does not need to be credited for evaluating compliance with PDC 19.

9.5.2.4.5 Compliance with 10 CFR 73.45(e)(2)(iii), 10 CFR 73.45(g)(4)(i), 10 CFR 73.45(g)(4)(ii), 10 CFR 73.46(f), 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.

9.5.2.4.6 Electromagnetic Interference and Radiofrequency Interference Compatibility

In 10 CFR 52.47(a)(9), the NRC requires, in part, that applications for LWRs evaluate the standard plant design against the SRP revision in effect 6 months before the docket date of the application. The evaluation should identify and describe all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the SRP acceptance criteria. 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.

DCA Part 2, Tier 2, Section 9.5.2.2.3, states, in part, the following:

> The COMS does not adversely impact other plant systems with EMI and RFI and that the system is designed in consideration of the guidelines of Regulatory Guide 1.180, which identifies electromagnetic environment operating envelopes, design, installation, and test practices for addressing the effects of EMI, RFI, and power surges on instrumentation and controls systems and components.

DCA Part 2, Tier 2, Section 3.11, describes the equipment qualification program requirements for the COMS as well.

Because the applicant committed to conform to RG 1.180, the 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

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

9.5.2.6 Combined License Information Items

Table 9.5.2-1 lists the COL item number and description from DCA Part 2, Tier 2, Revision 2, Section 9.5.2.

Table 9.5.2-1 NuScale COL Information Item for Section 9.5.2

COL Item No.	Description	FSAR Tier 2 Section
9.5-1	A COL applicant that references the NuScale Power Plant design certification 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	9.5.2.1
9.5-2	A COL applicant that references the NuScale Power Plant design certification will determine the location for the security power equipment within a vital area in accordance with 10 CFR 73.55(e)(9)(vi)(B).	13.6.1

9.5.2.7 Conclusion

Based on the review above, and to the extent the application addressed the use of the COMS in intraplant 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

The plant lighting system (PLS) provides artificial illumination for buildings, rooms, spaces, and outdoor areas of the plant and must provide adequate lighting during all plant operating conditions such as (1) normal plant lighting, (2) emergency plant lighting, and (3) normal and emergency MCR lighting. The physical security system within the nuclear island and structures (RXB and CRB) relies on normal plant lighting and emergency plant lighting to support the successful implementation of security functions. This section of the SER provides the staff's evaluation of the normal plant lighting, the emergency plant lighting, and the normal and emergency MCR lighting systems for the NuScale design.

Chapter 13 of this SER provides the staff's evaluation of the lighting for physical security.

9.5.3.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 3.8, "Plant Lighting Systems," provides information associated with this section of the SER. The PLS supports up to 12 NPMs. The PLS provides artificial illumination for the entire plant: buildings (interior and exterior), rooms, spaces, and all outdoor areas of the plant. The PLS consists of normal and emergency lighting and includes miscellaneous nonlighting loads as required. The PLS supports the RXB by providing normal lighting, emergency lighting, and emergency lighting for the RSS. The PLS supports the CRB by providing normal lighting and emergency lighting in the MCR.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 9.5.3, "Lighting Systems," describes the lighting system, as summarized below.

The PLS is a non-Class 1E system that is not safety related. Normal plant lighting and emergency lighting do not perform safety-related functions during or after a DBA. DCA Part 2, Tier 2, Section 9.5.3.1, “Design Bases,” states, “Failure of the PLS does not compromise automatic actuation of nuclear safety-related systems, nor does it prevent safe shutdown of the reactor.” DCA Part 2, Tier 2, Section 9.5.3.1, also states that the PLS plant illumination levels are in accordance with the applicable lighting levels recommended in NUREG-0700, Revision 2, “Human-System Interface Design Review Guidelines,” issued May 2002. DCA Part 2, Tier 2, Section 9.5.3.1, further states the following:

> Lighting fixtures in the MCR, remote shutdown station, and in areas containing safety-related structures, systems, and components are mounted to meet Seismic Category II requirements to ensure that their failure does not reduce the functional reliability of safety-related equipment, result in incapacitating injury to control room occupants, or render the control room uninhabitable.

DCA Part 2, Tier 2, Section 9.5.3.2, discusses the normal and emergency plant lighting as well as MCR lighting.

Normal Lighting System

The normal plant lighting provides artificial illumination for outdoor areas on the plant site and for plant buildings, including the RXB, CRB, TGB, RWB, fire water building, diesel generator building (DGB), security buildings, administration and training building, annex building, central utilities building, cathodic protection system, and other miscellaneous loads that are not safety related. The normal plant lighting provides an independent electrical distribution system that is powered by the ELVS described in DCA Part 2, Tier 2, Section 8.3.1. PLS circuits are designed to ensure some lighting is maintained in the event of a circuit failure. In the event of loss of ac power to the normal PLS, the emergency lighting system provides plant lighting.

Emergency Lighting System

Emergency lighting outside the control room provides illumination upon loss of normal lighting in plant areas where emergency operations are performed, including the TSC, RSS, battery rooms, electrical distribution control panels, BDGs and their controls, and fire water building. DCA Part 2, Tier 2, Section 9.5.3.2, states, “Emergency lighting is sufficient to support required activities, such as fire suppression and safe shutdown, and to illuminate access and egress pathways to safe shutdown areas during a fire.” Emergency lighting located outside of the MCR consists of self-contained, battery-powered fixtures.

Normal and Emergency Main Control Room Lighting

Normal and emergency MCR lighting provides artificial illumination under all operating, maintenance, testing, and emergency conditions. Emergency lighting in the MCR is consistent with the minimum illumination levels in NUREG-0700, as discussed in DCA Part 2, Tier 2, Section 9.5.3.6, Reference 9.5.3-2. The MCR lighting fixtures are designed to operate on either 120-volt ac power or 125-volt dc power, which allows MCR lighting to automatically transfer from the normal ac power source to the emergency dc power source in the event of a failure of normal ac power. Normal and emergency lighting circuits are fed from their respective lighting panels, which are physically separated from each other. The MCR lighting circuits that automatically transfer from normal to emergency power are transferred to the normal power supply after power is restored. The MCR lighting is supported by two divisions of normal ac power that are not safety related and two divisions of emergency dc power that are not safety

related. The PLS supplies normal (ac) MCR lighting power, which is powered by the ELVS described in DCA Part 2, Tier 2, Section 8.3.1. The ELVS and PLS are not safety related and not risk significant. The not safety-related EDSS-C, described in DCA Part 2, Tier 2, Section 8.3.2, supplies emergency (dc) MCR lighting power. The EDSS-C batteries in either division are capable of maintaining MCR emergency lighting at a minimum illumination level of 110 lux (10 foot-candles) at work stations in the main operating area for a minimum of 72 hours following a DBE.

ITAAC: The ITAAC associated with lighting systems are provided in DCA Part 2, Tier 1, Section 3.8.2, Table 3.8-1, "Plant Lighting System ITAAC," and are discussed in SER Section 14.3. DCA Part 2, Tier 2, Chapter 14, "Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria," also includes initial testing and verification on the not safety-related, non-Class 1E electrical equipment. DCA Part 2, Tier 2, Table 14.2-60, "Plant Lighting System Test #60," describes these items.

Technical Specifications: No GTS are associated with this area of review.

9.5.3.3 Regulatory Basis

No specific GDC or other requirements directly apply to the performance of the lighting systems. However, 10 CFR 50.34(f)(2)(iii) states that an application shall provide a control room design that reflects state-of-the-art, human-factor principles before committing to the fabrication or revision of fabricated control room panels and layouts. A control room design includes lighting in order 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 2, applies as it relates to acceptable lighting levels.

- RG 1.75, Revision 3, 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.

9.5.3.4 Technical Evaluation

The staff reviewed the information in DCA Part 2, Tier 1, Section 3.8.1, "Lighting Systems," and DCA Part 2, Tier 2, Section 9.5.3, to determine whether the plant lighting levels are adequate during all plant operating conditions and whether the lighting systems can operate without adversely impacting the operation, control, and maintenance of SSCs.

SRP Section 9.5.3 states that the PLSs must provide adequate lighting during all plant operating conditions, including fire, transient, and accident conditions. DCA Part 2, Tier 2,

Section 9.5.3.1, states that the plant illumination levels provided by the PLS are in accordance with the applicable lighting levels recommended in NUREG-0700. DCA Part 2, Tier 2, Section 9.5.3.2, further states that the PLS is capable of delivering at least 1080 lux (100 footcandles) of illumination to the MCR and RSS seated operator stations and 540 lux (50 footcandles) of illumination to the MCR and the RSS primary operating areas and remote and auxiliary operating panels. The staff reviewed the illumination levels provided for (1) various tasks and work areas as specified in NUREG-0700, Table 12.1, "Nominal Illumination Levels for Various Tasks and Work Areas," (2) in-plant areas as specified in NUREG-0700, Table 12.10, "Range of Recommended Illuminances for Inspection/Assembly Activities," and (3) all other areas and rooms of the plant required for control and maintenance of equipment and plant access routes during normal plant operations. The staff finds the plant illumination levels provided by the PLS are in accordance with the applicable lighting levels specified in NUREG-0700.

SRP Section 9.5.3 states that the emergency lighting system should be capable of providing adequate lighting during all plant operating conditions. This evaluation should include consideration of fire, transient, and accident conditions. DCA Part 2, Tier 2, Section 9.5.3.4, states that the emergency lighting is inspected and tested periodically. The applicant provided clarification in a letter dated September 20, 2017 (ADAMS Accession No. ML17263B241), that testing and inspection programs for emergency lighting will be determined and developed by the licensee. The applicant further stated in a letter dated March 19, 2018 (ADAMS Accession No. ML18078B313), that the plant testing and inspection program is covered by the more broadly stated direction in COL Item 13.5-3. COL Item 13.5-3 directs an applicant that references the NuScale design certification to describe the site-specific maintenance and other operating procedures. The staff finds that NuScale conforms to the guidance in SRP Section 9.5.3 because the emergency lighting system design has adequate emergency lighting for all plant operating conditions, including fire, transient, and accident conditions, thus conforming to the guidance in NUREG-0700. In addition, maintenance of the emergency lighting system is adequately addressed since the emergency lighting system will be inspected and tested periodically in accordance with the program to be developed by the COL applicant.

Inspection and testing of normal lighting are addressed as part of preoperational testing. Section 14.2 of this SER provides the associated staff evaluation.

DCA Part 2, Tier 2, Section 9.5.3.2, states that emergency lighting, provided by lighting fixtures in areas where operators perform actions to support safe shutdown of the plant in the event of an SBO, is equipped with an 8-hour battery backup. In addition, the fixtures where operators perform actions required for a fire safe-shutdown scenario are also equipped with an 8-hour battery backup. In the ingress and egress areas, the fixtures are equipped with 1.5-hour battery backup. The staff finds that the applicant has addressed the effect of the loss of all ac power (i.e., during an SBO) on the emergency lighting system because battery backup lighting is provided in areas where operators perform actions critical to plant operation. Section 8.4 of this SER provides the staff's review and evaluation of an SBO.

In DCA Part 2, Tier 2, Section 9.5.3.1, the applicant stated that the guidance in RG 1.75 regarding physical separation between lighting circuits that are not safety related and safety-related circuits is not applicable because all onsite ac power systems are not safety related and non-Class 1E. However, the applicant has requested an exemption from GDC 17, "Electric Power Systems," as stated in DCA Part 7, "Exemptions," Section 4.2.1, such that neither onsite nor offsite power is required for the mitigation of Chapter 15 events. The staff

discusses the exemption request in Section 8.1.5 of this SER. In addition, the staff's evaluation of RG 1.75 is addressed in Section 8.3.1 of this SER.

The staff finds that the normal plant lighting, the emergency plant lighting, and the normal and emergency MCR lighting systems 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 NUREG-0700 and the SRP.

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 Combined License Information Items

No COL information items are associated with this section.

9.5.3.8 Conclusion

The staff reviewed the NuScale normal plant lighting, the emergency plant lighting, and the normal and emergency MCR lighting systems for conformance with the guidelines of SRP Section 9.5.3 and NUREG-0700, Chapter 12, "Workplace Design." Based on the above technical evaluation, the staff concludes that the normal plant lighting, the emergency plant lighting, and the normal and emergency MCR lighting systems design provides 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 NUREG-0700. Therefore, the staff concludes that the normal plant lighting, the emergency plant lighting, and the normal and emergency MCR lighting system designs are acceptable.

9.5.4 Backup Diesel Generator Auxiliary Systems

The applicant did not address in DCA Part 2, Tier 2, Chapter 9, the support systems associated with the operation of its backup power supply systems (BPSS). Instead, the applicant indicated in DCA Part 2, Tier 2, Section 8.3.1.1.2, that safety-related functions do not rely on ac electrical power from the BPSS, which includes the backup diesel generator (BDG) and the associated support systems. The BPSS is not safety related or risk significant.

The applicant stated in DCA Part 2, Tier 2, Section 8.3, that the BDGs are standalone, skid-based installations including support features. Therefore, auxiliary system support features are not assigned to discrete subsystems similar to those of traditional large LWR safety-related diesel applications, where each BDG is a standalone, skid-based installation that includes the following auxiliary equipment:

- diesel engine starting system
- combustion air intake and engine exhaust subsystem
- engine cooling subsystem
- engine lubricating oil subsystem
- engine fuel subsystem (including fuel storage and transfer)

- generator excitation, protective relaying, and I&C subsystem

DCA Part 2, Tier 2, Section 1.2.2.5.5, states that the NuScale design includes two DGBs, each housing a single BDG and associated auxiliary equipment. The DGBs house no safety-related systems and have no functional requirements that support the ESFs.

The staff reviewed Part 2 of the DCA to determine if the failure of the BPSS auxiliary systems could potentially have an adverse impact on important-to-safety SSCs, or on the plants' ability to achieve and maintain safe shutdown. The staff finds that, since the BPSS is not safety related and serves no safety-related functions, the requirements for diesel support systems operation, including GDC 17, 44, 45, and 46, specified in the discussion of diesel generator support systems in SRP Sections 9.5.4–9.5.8, are not applicable for the NuScale BPSS auxiliary systems. Furthermore, although the structural design of the DGBs is designated as CDI in DCA Part 2, all SSCs important to safety for the NuScale plant are located inside the RXB and CRB, both of which are adequately robust to protect the SSCs from the effects of BDGs failures. Therefore, the staff concludes that the requirements of GDC 2, 4, and 5 are not applicable to the BPSS, and the staff thus finds the BPSS auxiliary systems acceptable since the failure of the BPSS auxiliary systems will not have an adverse impact on important-to-safety SSCs or on the plants' ability to achieve and maintain safe shutdown.